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Organisation de Coopération Et de Développement Économiques (ocde)

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NUCLEAR
SAFETY RESEARCH INDEX

1977

VOLUME II
WATER REACTORS
(continued)

8. INSTRUMENTATION, CONTROL AND COMPUTERIZED
PROTECTION

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

Objectif général :

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

Objectifs particuliers :

Avant l'application aux composants des conditions d'environnement de l'accident de référence, il faut faire subir à ces derniers un vieillissement accéléré simulant le vieillissement normal en ambiance de centrale pendant 30 à 40 ans. La méthode de vieillissement accéléré pour être valable devra correspondre à une altération des caractéristiques spécifiques déterminées par rapport à la fonction que le composant doit assurer. Suivant l'utilisation qui en est fait, le ou les critères de vieillissement d'un élément peuvent être très variés : Par exemple, le critère déterminant la durée de vie d'un câble d'alimentation ne peut pas être le même que celui d'un câble véhiculant les impulsions issues d'un détecteur. Dans cette action, on se propose de travailler en collaboration avec les équipes chargées des études de vieillissement accéléré et des irradiations pour examiner si les méthodes proposées seront valables pour les matériels entrant dans le contrôle neutronique et la sécurité des réacteurs. On établira des méthodes d'essais après vieillissement accéléré. En dehors des mesures classiques (tenue mécanique, rigidité diélectrique, etc.) effectués sur certains composants soumis à l'ADR, il importe, en particulier pour les câbles, de faire des mesures particulières pour ceux entrant dans la composition des ensembles de sécurité. Ces mesures font appel

à des techniques spéciales ignorées pour les applications classiques. Il s'agit par exemple de mesurer dans les isolants l'apparition de microdécharges dont l'amplitude est de l'ordre de 10^{-13} C et qui peuvent être prises en compte au même titre que les impulsions dues aux neutrons venant du détecteur. Le vieillissement des alliages entrant dans la composition du blindage des câbles doit également être contrôlé (mesure d'impédance de transfert) etc. Des appareillages permettant d'effectuer les essais doivent être développés et les méthodes de mesure étudiées et mises au point.

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

Objectif général :

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

Objectifs particuliers :

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

Installations expérimentales et programme :

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

Etat de l'étude :

1) Avancement à ce jour :

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

2) Résultats essentiels :

Les méthodes sont définies.

Prochaines étapes :

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

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

Documents de référence :

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

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

Objectif général :

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

Objectifs particuliers :

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

Etat de l'étude :

1) Avancement à ce jour :

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

2) Résultats essentiels :

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

Prochaines étapes :

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

.../...

Documents de référence :

- "Méthode de mesure de l'immunité aux parasites d'un ensemble de mesure neutronique", J.BUISSON - Note SES/interne/SAI - 76/152.
- "Méthode d'essais et de mesure de la sensibilité aux perturbations des chaînes de contrôle neutronique et de détection de rupture de gaine", J.BUISSON - Note SES/interne/SAI - 76/263.
- "Contrôle de la susceptibilité aux perturbations des ensembles de démarrage de Fessenheim", J.BUISSON - Notes SES/interne/SAI - 76/99 et 76/211.

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

Objectif général :

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

Objectifs particuliers :

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

Etat de l'étude :

Avancement à ce jour :

L'étude débute actuellement.

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

Objectif général :

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

Objectifs particuliers :

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

Prochaines étapes :

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

Documents de référence :

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

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

Objectif général :

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

Objectifs particuliers :

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

Etat de l'étude :

Avancement à ce jour :

L'étude débute actuellement.

Prochaines étapes :

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

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

Objectif général :

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

Objectifs particuliers :

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

Etat de l'étude :

Avancement à ce jour :
 Etude bibliographique en cours.

Prochaines étapes :

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

Relation avec d'autres études :

Etude mission 4 : Comportement des matériaux aux ambiances nucléaires.

Documents de référence :

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

.../...

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

Objectif général :

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

Objectifs particuliers :

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

Avancement à ce jour :

Approvisionnement des compteurs en cours de réalisation.

Prochaines étapes :

Essais de qualification d'une année sur TRITON.

<u>Classification: 8</u>	
<u>Title 1 (Original Language):</u> Einsatz von Prozeßrechnern in Leistungsreaktoren zur Verbesserung der Betriebssicherheit (ATT O85 A - II.4.4., Jahresbericht A 76)	<u>COUNTRY:</u> BRD
	<u>SPONSOR:</u> BMI
	<u>ORGANIZATION:</u> LRA, Garching
<u>Title 2 (English):</u> The Use of Process Computers in Power Reactors for Improvement of Operation Safety	<u>Project Leader:</u> Dr. H. Hoermann Dr. W. Ehrenberger
<u>Initiated (Date):</u>	<u>Completed (Date):</u>
<u>Status:</u> continuing	<u>Last Updating (Date):</u> December 1976

1. General Aim

(a)

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

(b)

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

2. Particular Objectives

(a)

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

plant disturbances. Development of methods to implement new concepts for incipient failure detection using process computers.

(b)

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

3. Experimental Facilities and Program

(b)

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

4. Project Status

4.1 Progress to Date

(a)

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

(b)

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

Concerning the computerized protection system for the Brunsbüttel power plant (KKB) more than 10^5 additional test runs had been performed. It turned out that with regard to some points an obsolete version of the

specification had been used and that the employed reaction mode for reactor pressure changes worked differently to a normal two out of three selection. The computer self supervision programs were improved, now providing a better computer availability. Other investigations helped to clarify the causes of faults which occurred intermittently in the computers installed at the plant. ADCs of that system were remeasured, faults in the VVT (process interface) and concerning the IBM type writer must be corrected.

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

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

4.2 Essential Results

(a) -

(b)

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

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

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

5. Next Steps

(a) -

(b)

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

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

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

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

6. Relation with Other Projects

The project is related to the following activities:

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

7. Reference Documents

/1/

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

/2/

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

/3/

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

/4/

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

/5/

W.E. Büttner
Sicherheitstechnischer Vergleich eines festverdrahteten dynamischen Reak-
torschutzsystems. Teil I
MRR 161, Juli 1976

/6/

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

8. Degree of Availability

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

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

1. General Aim

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

2. Particular Objectives

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

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

3. Research Program

3.1 Safety valve tests

3.1.1 Pilot valve tests

3.1.2 Main and pilot valve tests

3.2 Analytical examination of the blowdown system

Compilation of conditions of loading of blowdown system

4. Experimental Facilities

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

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

5. Progress to Date

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

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

6. Results

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

7. Next Steps

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

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

8. Relation with Other Projects

9. References

10. Degree of Availability

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

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

3. Experimental facilities and programme

3.1. Experimental facilities

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

3.2. Programme

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

4. Project status

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

6. Reference documents

- 1) M. Luminari, G. Masini, F.A. Tacconi " Sviluppo del sistema a fluido bifase per la regolazione rapida della potenza neutronica del reattore prototipo CIRENE: impianto fuori pila BB-2" CISE R-300, 1970.
- 2) C.A. Marchiondelli, G. Masini, M. Perego, F.A. Tacconi "Sistema delle Barre Bifasi: impianto prototipo BB-3 per le prove di controllo di un reattore a potenza zero" CISE R-337, 1973.
- 3) F.A. Tacconi "Il programma BUANA '74 per la simulazione del sistema Barre Bifasi" CISE R-350, 1974
- 4) R. Granzini, G. Masini, A. Vanossi, A. Venturi, G. Zappellini "Prove di regolazione del reattore RB 3 col sistema delle barre Bifasi. Analisi dei risultati della seconda campagna sperimentale" CISE R-385, Dic. 1976.

<u>Title 1 (Original language)</u>	<u>Classification</u>
Sistema di controllo di reattori con barre bifasi	8

7. Degree of availability: to a limited extent

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

1. General aim.

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

2. Particular objectives

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

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

3. Experimental facilities

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

4. Project status

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

5. Next steps

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

6. Relation to other projects

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

<u>Title 1 (Original language)</u>	<u>Classification</u>
Diagnostica e Sicurezza tramite Analisi di Rumore (Parte A)	8 - 14

7. Reference documents

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

8. Degree of availability

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

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

1. General aim

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

2. Particular objectives

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

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

3. Experimental facilities

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

4. Project status

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

5. Next steps

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

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

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

<u>Title 1 (Original language)</u>	<u>Classification</u>
Diagnostica e Sicurezza tramite Analisi di Rumore (Parte B)	8 - 14

6. Relation to other projects

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

7. Degree of availability

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

PROJECT TITLE : Reactor Malfunction Diagnosis using Noise Analysis	CSNI INDEX NUMBER : 8 - 14 - 10
SPONSORING COUNTRY : Portugal	ORGANISATION : Junta de Energia Nuclear LFEN (Sacavém)
DATE INITIATED : January 1972 DATE COMPLETED : (in progress)	PROJECT LEADER : J.COSTA OLIVEIRA

Description :

Reactor noise signatures gained attractiveness as a useful aid to reactor control in detecting anomalous behaviour of reactor components. We try to apply noise analysis techniques, in order to isolate the major sources of neutron noise in a swimming-pool reactor. Our ultimate goal is to assess the correct performance of in-core and ex-core instrumentation to safety monitoring and early detection of abnormal operating conditions

Facilities

- 1 swimming-pool reactor (RPI).
- Transducers, amplifiers, sampling unit, magnetic recorder.
- 1 digital computer (PDP 15/30).
- Software for data processing.

Reference list

J.Costa Oliveira, C.Veiga, D.Trigueiros, J.Pombo Duarte, "Data Acquisition and Processing System for Reactor Noise Analysis", Annals of Nuclear Energy, 2, 149-159 (1975).

J.Varela, "Análise do Ruído Neutrónico do Reactor Português de Investigação", LFEN B-10 (1976).

OECD NUCLEAR ENERGY AGENCY - 38, Boulevard Suchet, 75016 PARIS
Telex : 630668 AEN NEA - Tel : 524 82-00

31.719

PROJECT TITLE : Noise analysis as a means for safety monitoring	CSNI INDEX NUMBER : 8.
SPONSORING COUNTRY : Sweden	ORGANISATION : AB Atomenergi
DATE INITIATED : July 1975 DATE COMPLETED : Phase 2 August 1976	PROJECT LEADER : Pehr E Blomberg

Description :

The aim of the project is to develop techniques for using noise measurements and analysis as a means for early detection of plant malfunctions and for improvement of operational reliability and safety.

Signals from the operational instrumentation of Swedish nuclear power plants are recorded and sampled. A multichannel taperecorder and a process computer are used for the data acquisition and the spectral analyses.

A state-of-the-art report has been published, a build-up of a noise data library has been initiated and the data collection and analysis work is continued.

Reference documents are reports to the Swedish Nuclear Power Inspectorate.

Reports on initial results has been published 1976.

PROJECT TITLE : Safety criteria for use of computer control of nuclear power plants	CSNI INDEX NUMBER : 8
SPONSORING COUNTRY : Sweden	ORGANISATION : AB Atomenergi
DATE INITIATED : 1976 DATE COMPLETED : 1977	PROJECT LEADER : Per-Gunnar Sjölin

Description :

The aim of the project is to establish and verify - in consideration of similar work going on in other countries - safety and reliability criteria for a system of computer control of nuclear power plants.

PROJECT TITLE : Supervision of reactor containment of PS-type by the aid of measurement of air-borne activity	CSNI INDEX NUMBER : 8
SPONSORING COUNTRY : Sweden	ORGANISATION : AB Atomenergi
DATE INITIATED : July 1976 DATE COMPLETED : September 1977	PROJECT LEADER : Lennart Svansson

Description :

Measurement of air-borne activity seems to be the fastest and most sensitive method to detect leaks of fission products into the PS-containment of a BWR. The objective is to investigate if the monitoring system should be based on direct measurement of the fission gases or measurement of aerosol activity.

	CLASSIFICATION: 8
TITLE: Computerized Protection Systems.	COUNTRY: Norway SPONSOR: OECD Halden Reactor Project ORGANIZATION: " " " "
	PROJECT LEADER:
INITIATED: 1975 STATUS: Planning	COMPLETED: LAST UPDATING: April 1977

1. *General Aim* Evaluation of methods for ensuring reliable software.
2. *Particular Objective* To obtain coding convention for high reliable software for small process computers.
3. *Experimental Facilities* Process computers.
4. *Project Status* A proposal for a work schedule has been made.
5. *Next Step* Further evaluation.
6. *Reference Documents* Annual reports to be published by the Nuclear Energy Agency of OECD.
7. *Degree of Availability* Upon request from the Nuclear Energy Agency.

	<p>CLASSIFICATION: 8, 15.1</p>
<p>TITLE: Operator - Process Communication.</p>	<p>COUNTRY: Norway</p> <p>SPONSOR: OECD Halden Reactor Project</p> <p>ORGANIZATION: " " " "</p>
	<p>PROJECT LEADER: M. Øvreide</p>
<p>INITIATED: 1970</p> <p>STATUS: In progress</p>	<p>COMPLETED:</p> <p>LAST UPDATING: April 1977</p>

1. *General Aim* To improve control room design through the application of computerized colour-graphic technique.
2. *Particular Objective* Investigate various ways of communication between the operator and the process, including methods for optimal presentation of information to the operator as well as the operator's interaction with the process.
3. *Experimental Facilities* Process computers, operator panels, and colour TV screens. Testing of the system-solutions in the Halden Reactor.
4. *Project Status* Experiments with a first prototype has been performed.
5. *Progress to Date* Experiments involving direct computer control and operation of the reactor plant have been conducted.
6. *Next Step* To develop a new prototype with special emphasis on high reliability and accessibility.
7. *Reference Document* Sixteenth Annual Report, OECD Halden Reactor Project, 1975, pages 21 - 23.
8. *Degree of Availability* Upon request from the Nuclear Energy Agency.

	<p>CLASSIFICATION: 8, 15.1</p>
<p>TITLE: Plant Disturbance Analysis.</p>	<p>COUNTRY: Norway</p> <p>SPONSOR: OECD Halden Reactor Project</p> <p>ORGANIZATION: " " " "</p>
	<p>PROJECT LEADER: M. Øvreide</p>
<p>INITIATED: 1974</p> <p>STATUS: In progress</p>	<p>COMPLETED:</p> <p>LAST UPDATING: April 1977</p>

1. *General Aim* Development of methods for on-line analysis of plant disturbances.
2. *Particular Objectives* Development of methods for a plant disturbance analysis system with emphasis on the interactive communication system for display of the fault propagation and on operator request, presentation of the consequences of the operator's intervention in the analyzing procedure. The causes of the disturbance, the possible consequences it may lead to, and the actions to take for the operator, will be presented to the operator on a colour display screen.
3. *Experimental Facilities* Process computer with colour display screen and keyboard.
4. *Project Status* A prototype has been made and an experiment has been performed at the Halden plant.
5. *Next Step* The system will be tested against simulated disturbances on large nuclear power plants. The form of presentation will be refined.
6. *Reference Documents* Annual reports to be published by the Nuclear Energy Agency of OECD. Sixteenth Annual Report, OECD Halden Reactor Project, 1975, pages 23 - 25.
7. *Degree of Availability* Upon request from the Nuclear Energy Agency.

PROJECT TITLE : Safety aspects related to control room, training and other human factors problems of nuclear power plants	CSNI INDEX NUMBER : 15.1 8. 15.2
SPONSORING COUNTRY : Denmark, Finland, Norway, Sweden	ORGANISATION : in Finland Technical Research Centre of Finland
DATE INITIATED : 1976 DATE COMPLETED : continues	PROJECT LEADER : Björn Wahlström

Description :

The project aims to the study of human factors problems in nuclear power plants. The areas to be covered are then

- evaluation of present and future control rooms
- evaluation and prediction of human performance
- training of staff

The project is intended as a joint project between institutions in the Scandinavian countries.

9. OTHER SAFEGUARDS

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

1. General Aim

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

2. Particular Objectives

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

3. Research Program

The investigation program encompassed the following works:

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

4. Experimental Facilities

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

5. Progress to Date

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

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

6. Results

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

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

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

7. Next Steps

The work has been completed.

8. Relation with Other Projects

RS 108 Fracture Safety Device for the Primary Circuit

9. References

10. Degree of Availability

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

1. General Aim and 2. Particular Objectives

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

3. Research Program

The investigation program encompassed the following work:

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

4. Experimental Facilities

No experimental facilities were necessary.

5. Progress to Date

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

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

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

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

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

6. Results

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

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

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

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

7. Next Steps

The next steps are in preparation.

8. Relation with Other Projects

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

9. References

10. Degree of Availability

10. CORE AND PRIMARY CIRCUIT IN STEADY STATE CONDITIONS
(no entries)

- 10.1. Physico chemical and material properties and their effects on fuel elements, core internals, control mechanisms and primary circuit components

PROJECT TITLE : NUCLEAR FUEL MATERIALS	CSNI INDEX NUMBER : 10.1, 11.1, 18
SPONSORING COUNTRY : FINLAND	ORGANISATION TECHNICAL RESEARCH CENTRE OF FINLAND REACTOR MATERIALS RESEARCH
DATE INITIATED : 1970 DATE COMPLETED : CONTINUES	PROJECT LEADER : MATTI OJANEN

Description :

1. GENERAL AIM

To generate experimental data of materials used in light water reactor fuel elements and to evaluate the influence of material properties on fuel behaviour.

2. PARTICULAR OBJECTIVES

- Basic properties of Zr-1%Nb and Zircaloy-2 cladding materials
- Fabrication methods (cladding tubes, sintered UO_2 pellets fuel assemblies)
- Quality control tests in fuel fabrication
- Generation and verification of fuel materials data for fuel element behaviour analysis under normal, off-normal and accident conditions

The main emphasis is on Zr-1%Nb material properties and on quality control tests and methods in fuel fabrication.

3. PROJECT STATUS

The microstructure and the mechanical properties of Zr-1%Nb cladding material have been investigated. Creep tests as well as corrosion tests will be initiated. The construction of suitable equipment to study cladding tube burst properties has been started. Evaluations of quality control programmes have been done. Quality control tests to verify manufacturers' data of fuel materials will be performed.

(cont.)

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14.560

PROJECT TITLE : NUCLEAR FUEL MATERIALS	CSNI INDEX NUMBER : 10.1, 11.1, 18
SPONSORING COUNTRY : FINLAND	ORGANISATION TECHNICAL RESEARCH CENTRE OF FINLAND REACTOR MATERIALS RESEARCH
DATE INITIATED : 1970 DATE COMPLETED : CONTINUES	PROJECT LEADER : MATTI OJANEN

Description : (cont.)

4. REFERENCE DOCUMENTS

Reactor Materials Research reports (partly confidential),
conference papers etc.

5. DEGREE OF AVAILABILITY

Upon request from the Technical Research Centre of Finland.

<u>Classification: 10.1</u>	
<u>Title 1 (Original Language):</u> Anwendung statistischer Analysenverfahren in Leistungsreaktoren mit dem sicherheitstechnischen Ziel der Früherkennung von Schäden (RS 68 A - II.4.1, Jahresbericht A 76)	COUNTRY: BRD
	SPONSOR: BMFT
	ORGANIZATION: TU Hannover
<u>Title 2 (English):</u> Application of Statistical Analysis Methods in Power Reactors under the Safety Oriented Aspects of Early Fault Detection	<u>Project Leader:</u> Prof.Dr.-Ing. D.Stegemann Institut für Kerntechnik
<u>Initiated (Date):</u> 1.3.1975 <u>Status:</u> Continuing	<u>Completed (Date):</u> 28.2.1977 <u>Last Updating (Date):</u> December 1976

1. General aim

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

2. Particular Objectives

Noise measurements in both Boiling- and Pressurized Water Reactors in order to gain extensive experience about space-dependent reactivity effects caused by pressure-, vibration- or coolant density feedback. Development of a new and reliable in-core measuring system, using "self-powered detectors", since in most cases a suitable instrumentation is not available.

Application of these improved devices for neutron flux distribution measurements with high spatial resolution and for reactor power noise measurements without disturbance of the normal plant operation.

3. Research Program

Continuous measurements at the PWR Obrigheim (KWO), concerning axial neutron flux distribution and local neutron noise, by means of a large number of self-powered neutron detectors.

Instrumentation of three new in-core assemblies for the 8th fuel cycle at KWO.

Development of improved self-powered-detectors and investigation of their signal components due to neutron and gamma fluxes.

Systematic investigations of the space dependent neutron flux fluctuations in the boiling water reactor Lingen (KWL), using different types of SPN-detectors, which could be inserted in a tube of the TIP-system and moved at any axial position in the core.

Computer analysis of the recorded stochastic signals and evaluation with respect to the identification of noise sources which might be relevant for safe reactor operation.

4. Experimental Facilities, Computer Codes

The data acquisition system consists of several axially and/or radially distributed in-core self-powered-detectors, connected to differential wide band amplifiers, analog frequency filters and an analog magnetic tape station.

In the KWO-PWR several instrumentation tubes, each with six Co-SPN-Detectors, are available.

For measurements in the KWL-BWR a three SPN-detector assembly was manufactured and inserted into a dry thimble of the TIP-System. This assembly could be vertically positioned in the core as desired.

Flux distribution measurements have been performed by using the DC-component of the detector signals while suppression of the stationary value and further amplification yields the fluctuating part which had been recorded on magnetic tape.

The collected data were processed later on to a digital computer via special frequency filters and analog-to-digital converter (ADC).

A sophisticated program system was developed for computer analysis of the stored signals in time and frequency domain.

5. Progress to Date

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

noise phenomena were carried out at the boiling water reactor Lingen (KWL), using a three-detector arrangement with Hafnium and Gadolinium detectors.

The data were stored on magnetic tape and prepared for an extensive computer analysis program.

6. Results

A large amount of data was obtained from the KWO-measurements in 4 different radial core positions and 6 vertical planes. The results of the computer analysis show remarkable structures in the neutron noise spectra, depending upon radial and vertical position of the detector, power distribution and thermohydraulic parameters. Up to 14 resonances could be found in the frequency range from 0 to 120 Hz.

For in-core measurements in the boiling water reactor Lingen a new detector type with Gadolinium emitter and two Hafnium detectors were applied. Because of the special design of the Hf-detectors extremely high neutron signals could be measured with an excellent spatial resolution of only 2,2 %. Also the Gd-SPN had the expected high signal level of similar magnitude. Neutron flux distribution and reactor noise measurements were performed in 30 vertical core positions.

7. Next Steps

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

8. Relation with other Projects

See IRS-F-24

9. References

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

10. Degree of Availability of the Reports

Reports are available through

Institut für Kerntechnik

Elbestr. 38 A

D-3000 Hannover 21

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

1. General Aim

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

2. Particular Objectives

The following objectives are envisaged:

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

3. Experimental Facilities and Program

Not relevant.

4. Project Status

4.1 Progress to Date

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

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

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

the pressure vessel are determined empirically. The measured data are used as a reference state of the reactor for the on-line monitoring.

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

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

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

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

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

4.2 Essential Results

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

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

5. Next Steps

With the proto-type vibration monitoring system at GKN long time behaviour, normal operational influences and signal variations during the first fuel cycle will be investigated systematically (e.g. increasing of the neutron noise PSDs corresponding to the increasing operation time). Studies are planned for the processing of the data and deriving of alarm criteria indicating an abnormal reactor state or malfunctions inside the primary system. It will be investigated whether methods of pattern recognition and learning codes for the reference functions can be involved to solve these problems. The experience in internal vibra-

tion monitoring as gained at PWRs will be applicated also to BWRs. In this sense the installed external accelerometers at the BRUNSBÜTTEL BWR will be used for loose parts detection as well as for longtime vibration control in the low frequency range. Accompanying model development will be done using finite element codes.

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

6. Relation with Other Projects

See annual report 1974.

7. Reference Documents

/1/

On-line-Überwachung zur Früherkennung von Fehlern und Schäden in Leichtwasserreaktoren

MRR 153, November 1975

/2/

B. Raible, J. Moravek, D. Wach

Ein Körperschall-Überwachungssystem zum Melden und Orten loser Teile
Compacts of Reaktortagung, Düsseldorf, März/April 1976

/3/

D. Wach

Korrelationsanalyse in der Meßtechnik von Kernkraftwerken

Lecture at Technische Akademie Esslingen, March 1976, published in:

Korrelationstechnik - ein neuer Zweig der industriellen Betriebsmeßtechnik, Lexika-Verlag, Grafenau, 1976

/4/

D. Wach

Ein neues Meßmodell der Neutronenflußkreuzleistungsdichte zur Identifizierung von Rauschquellen in Leistungsreaktoren

Thesis, Technische Universität München, Juli 1976

/5/

P. Panagopoulos

Korrelationsuntersuchungen der Rauschsignale von Neutronenfluß-Incore-detektoren in einem Druckwasserreaktor. Interner Bericht

MRR-I-72, August 1976

/6/

D. Wach

On-line-Schwingungsmessung am Reaktordruckbehälter des KKW Brunsbüttel, Referenzmessung während des Vollast-Probebetriebs am 22. Oktober 1976.

Vertraulicher Bericht

MRR-V-18, Oktober 1976

/7/

E. Sättler

Identifikation des Schwingungsverhaltens und Systemparameter-Estimation von Druck- und Kernbehälter des KKW Biblis A

MRR 165, Oktober 1976

/8/

R. Sunder, B. Österle

Modellentwicklung zum Schwingungsverhalten des Reaktordruckbehälters und seiner Einbauten am Kernkraftwerk Biblis A. Teil I: Analyse von Kennwertmessungen. Interner Bericht

MRR-I-61, Januar 1976

8. Degree of Availability

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

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

Ref. /4/ is available through

Technische Universität München, D-8000 München 2, Arcisstraße 21

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

1. General aim

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

2. Particular Objectives

Basing upon the experience in application of statistical analysis methods in power reactors (RS 68A), an on-line system is to be developed, that allows an early detection of certain faults during reactor operation and may support the operators to avoid dangerous operating conditions.

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

3. Research Program

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

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

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

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

4. Experimental Facilities, Computer Codes

Data acquisition system, consisting of in-core sensors (fission chambers, self-powered detectors), self-compensating low-noise amplifiers (developed at the Institut für Kerntechnik), signal distribution and multiplexing unit and analog magnet tape recorder with 14 tracks.

Analyzer which computes characteristic functions and values from the digitized input signals and decides about the storage on magnetic tape. Computer analysis of in-core data by Fourier transform, using partly the existing analysis methods and developing further program modules for special evaluation problems.

5. Progress to Date

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

Procuring of the following units:

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

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

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

Development of a driver for the ADC.

6. Results

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

7. Next Steps

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

8. Relation with other Projects

9. References

10. Degree of Availability of the Reports

10.2 Reactor physics
(no entries)

10.3. Thermohydraulics

PROJECT TITLE : Fuel Element Malfunction Detection & Identification	CSNI INDEX NUMBER : 10.3 LWR HWR
SPONSORING COUNTRY : Australia	ORGANISATION : University of N.S.W.
DATE INITIATED : 1974 DATE COMPLETED : Continuing	PROJECT LEADER : Mr. L.G. Kemeny Dept., of Nuc. Eng.,

Description :

Noise signals arising from hydroelastic vibration and temperature fluctuations in fuel elements and coolant are being utilised to identify normal and abnormal conditions for fuel channels.

A boiling simulation rig is being completed, and measurements of temperature noise on a thermocoupled fuel element are being made, using the HIFAR reactor of the Australian Atomic Energy Commission.

The work receives some support from the Australian Institute of Nuclear Science and Engineering.

OECD NUCLEAR ENERGY AGENCY - 38, Boulevard Suchet, 75016 PARIS
Telex : 630668 AEN NEA - Tel : 524 82-00

PROJECT TITLE : TURBULENT FLOW IN ROD CLUSTERS	CSNI INDEX NUMBER : 10.3 LWR HWR
SPONSORING COUNTRY : AUSTRALIA	ORGANISATION : School of Nuclear Engineering University of New South Wales
DATE INITIATED : March 1974	PROJECT LEADER :
DATE COMPLETED :	Prof. J.J. Thompson

Description :

Joint Project - School of Nuclear Engineering and Engineering Research
Division, Australian Atomic Energy Commission Research Establishment.

Work continues on experimental and theoretical research into the structure of fully developed turbulent flow in rod clusters, using a sin rod experimental channel. In addition to hydraulic and thermal aspects, the rig provides the opportunity for the determination of random pressure fluctuations, both near and far field.

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Telex : 630668 AEN NEA - Tel : 524 82-00

14.560

PROJECT TITLE : Critical Heat Flux in 37-Element Bundle Fuel	CSNI INDEX NUMBER : 10.3
SPONSORING COUNTRY : Canada (Atomic Energy Control Board)	ORGANISATION : Carleton University, Ottawa
DATE INITIATED : October 1975 DATE COMPLETED : October 1977	PROJECT LEADER : Prof. J.T. Rogers

Description :

1. General Aim

- 1) To establish a correlation(s) between critical heat flux (CHF) and the design and operating parameters for a 37-element nuclear fuel bundle.
- 2) To determine the uncertainties in the correlation(s) in order to permit a probabilistic analysis of CHF under normal operating conditions.
- 3) To recommend future experiments and analysis of CHF.

3. Experimental Facilities and Programme

A detailed study of all 37-element bundle data will be undertaken to establish a correlation or correlations which will follow forms which have been successful in the past. In addition, the study will survey all bundle CHF data obtained since 1970 to establish behaviour and trends with various design and operating parameters. Modifications deduced from these trends will be applied to the basic correlation, as required, to increase its range of applicability and confidence in its use.

4. Project Status

The researchers have developed a correlation based on all world data for sub-cooled, vertical, 37-element bundles which predicts CHF with an RMS deviation of $\pm 4.3\%$. The data has been taken from experiments covering a wide variety of conditions; the correlation can thus account for element clearance, heated perimeter, radial flux depression, sub-cooling, mass velocity and pressure. In progress now is the extension of the correlation to treat horizontal channels, via the Froude number, and to assess the effects of heated length, bundle junctions and non-uniform axial heat flux. The correlation is also to be tested further against more recent experimental data.

5. Reference Documents

Sadeghlou, M., "CHF in 37-Element Bundle Fuel"
Master's Degree Project Report
Carleton University, Ottawa, 1976.

9. Funding

\$9,500

OECD NUCLEAR ENERGY AGENCY - 38, Boulevard Suchet, 75016 PARIS
Telex : 63668 AEN NEA - Tel : 524 82-00

14.560

<u>Classification:</u> 10.3								
<u>Title 1 (Original Language):</u> Mischungseffekte bei parallel durchströmten Kanälen (RS 81 - II.1.3 , Jahresbericht A 76)	<u>COUNTRY:</u> BRD							
	<u>SPONSOR:</u> BMFT							
	<u>ORGANIZATION:</u> Euratom, Ispra							
<u>Title 2 (English):</u> Mixing Effects in Parallel Channels with Water Two-Phase Flow	<u>Project Leader:</u> H. Herkenrath W. Hufschmidt							
	<table border="0" style="width: 100%;"> <tr> <td style="width: 50%;"><u>Initiated (Date):</u></td> <td style="width: 50%;"><u>Completed (Date):</u></td> </tr> <tr> <td>1.1.1975</td> <td>31.12.1977</td> </tr> <tr> <td><u>Status:</u></td> <td><u>Last Updating (Date):</u></td> </tr> <tr> <td>Continuing</td> <td>December 1976</td> </tr> </table>	<u>Initiated (Date):</u>	<u>Completed (Date):</u>	1.1.1975	31.12.1977	<u>Status:</u>	<u>Last Updating (Date):</u>	Continuing
<u>Initiated (Date):</u>	<u>Completed (Date):</u>							
1.1.1975	31.12.1977							
<u>Status:</u>	<u>Last Updating (Date):</u>							
Continuing	December 1976							

1. General aim

The investigation of the interaction and mixing effects between adjacent subchannels of a fuel rod bundle cooled by boiling water is aimed at providing experimental data on the local distribution of mass-flow, enthalpy and void-fraction so as to achieve a better thermohydraulic description and more precise DNB margin for a BWR core. Furthermore, studies in steady-state conditions are a necessary preliminary to studies of the mixing process in transient conditions such as those occurring during power transients or blow-down. Therefore, the first phase of the programme will be devoted to steady-state and the transient conditions will be studied later.

2. Particular objectives

The mixing programme was started in the frame of collaboration contracts between CNEN (Rome) and BMFT (Bonn) on the one side and Euratom on the other. These contracts involved the studies of the boiling mixing phenomena using two different test sections: A two-channel test section for fundamental studies (pressure up to 150 bars; high-pressure water loop PRIL) and a 16-rod cluster test section with BWR geometry (pressure 70 bars; high-pressure water loop BOWAL). Later, the BOWAL studies will be extended to a pressure of 160 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 various preparations for the 16-rod bundle tests in BOWAL.

3. Research programme

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

The original programme provided the following studies:

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

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

The actual programme covers:

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

4. Experimental facilities

The high-pressure water loop BOWAL (3.6 MW power input) to be used for the boiling mixing experiments with the 16-rod cluster test-sections in BWR- and PWR-geometry and the high-pressure water loop PRIL (0.6 MW power input) to be used for the fundamental mixing studies and for calibration tests with void-meters have been described in detail in [1] and [2] and are schematically shown in figures 1 and 2.

5. Progress to date and results

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

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

foreseen for the signals from the pressure transducers, however, have to be replaced by more precise digital instruments. Furthermore, the motor regulation valves in the sampling pipes from the subchannels of the test-section were less sensitive than expected and the regulation spindles had to be replaced after different calibration tests (Fig. 3).

The mean outlet temperatures at the primary and secondary side of the condensers for the heat-balance measurements (for outlet-enthalpy determination in the sampled subchannels) showed remarkable deviations (about $\pm 5\%$). By means of the installation of appropriately developed flow mixing devices (Fig. 4) the deviations could be decreased to a value of about $\pm 1\%$ corresponding to the tolerances of the thermocouples used. These thermocouples are embedded in silver crosses as described in [2] and [3] to equilibrate temperature variations and fluctuations in the connection tubes of the condensers. As these silver crosses are manufactured by a cast-process small cavities (Fig. 5) caused tightness-troubles and had to be eliminated.

During 1976 the installation of the calibration device in PRIL loop (Fig. 2) has been completed and now an installation is available for the necessary continuous control and calibration of the X-ray void meters under real pressure and temperature conditions (up to 160 bars and 347°C). By means of two preheater tubes (300 kW power-input, each) and with an appropriate mixing chamber different qualities and flow-patterns of the steam-water mixture are adjustable. In a pressure housing after this mixing chamber the annular probe voidmeter [4] for the qualitative determination of the flow pattern will be tested. In this probe at the outlet a mixing device is incorporated providing an homogeneously distributed two-phase mixture over the tube cross-section (i.d. 16mm). After this device the X-ray voidmeters [5] are controlled and calibrated with the same arrangement used in the sampling pipes of BOWAL [3]. Due to administrative difficulties the originally foreseen X-ray tubes of TUBIX, Paris (30 kV 600 W) can not be used for the boiling mixing studies and new types (Siemens, Karlsruhe with an energy up to 60 kV and a power of 3000 W) have been ordered and have to be tested and calibrated.

7. Next steps

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

8. Relation with other projects

- RS 109 Experimental investigation of the influence of PWR loops on blowdown
- RS 163 Theoretical and experimental investigations on thermo- and fluiddynamic behaviour of the reactor core during the first blowdown phase.

9. References

- [1] Herkenrath, H., Mörk-Mörkenstein, P.
The pressurized and boiling water loop of the Technology Division at Ispra
EUR 4683 e (1971)
- [2] Herkenrath, H., Hufschmidt, W.
The pressurized and boiling water loops BOWAL and PRIL for boiling mixing studies of the Heat Transfer and Fluid Mechanics Division of JRC, Ispra
EUR (to be published in 1977)
- [3] 1975 Safety Programme Progress Report of the Ispra Establishment, Sheet 2.4., pp 73-83, ACS - 66 e
- [4] Herzberger, P., Hufschmidt, W.
Bestimmung des Dampfgehaltes und der Strömungsform eines Zweiphasengemisches in konzentrischen Ringspalten
EUR 5077 d (1974)
- [5] Herzberger, P., Hufschmidt, W., Wind, F.
Die Verwendung von Gamma- oder Röntgenstrahlen zur Messung des Dampfgehaltes in Zweiphasenströmungen
EUR 5415 d (1975)

10. Degree of availability of the reports

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

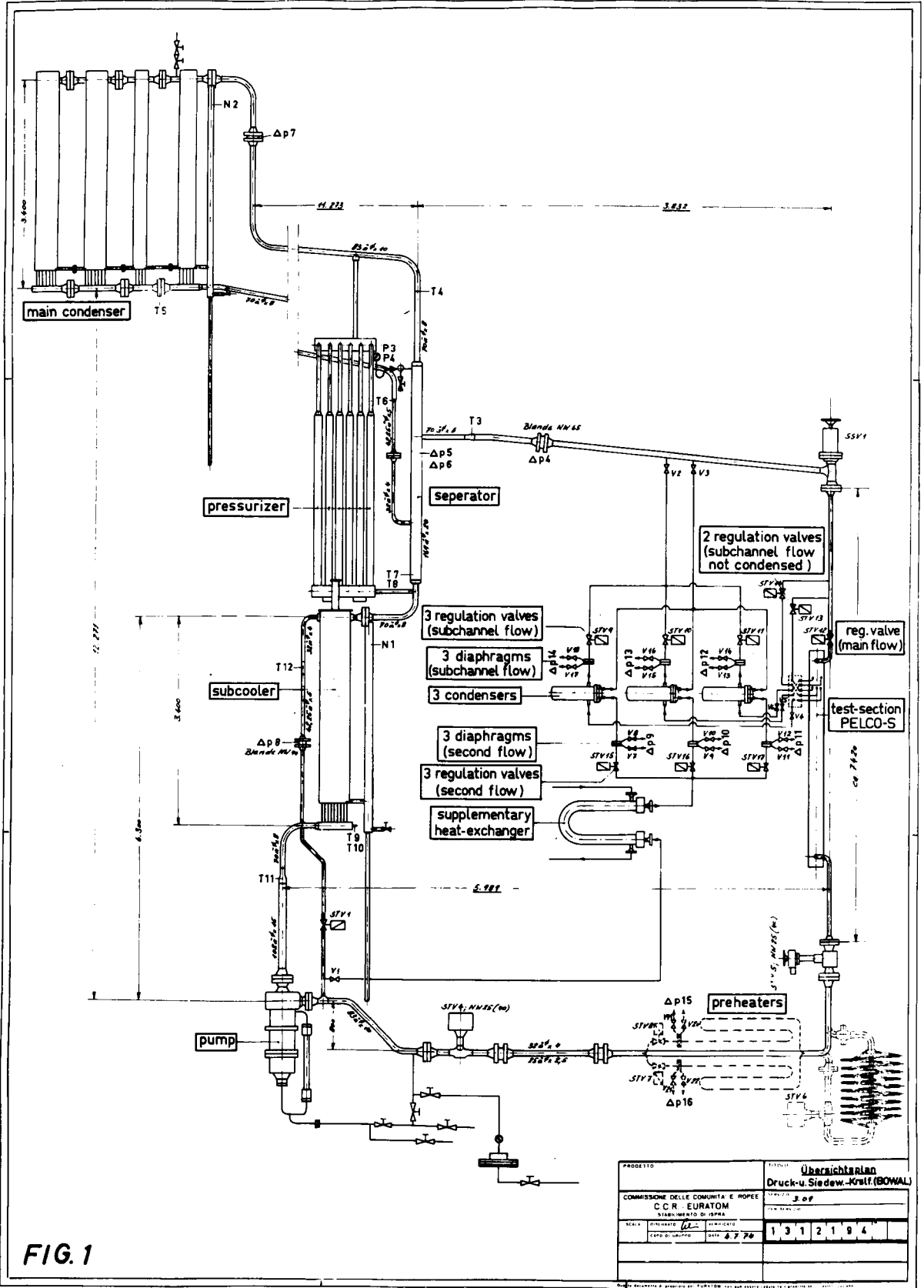
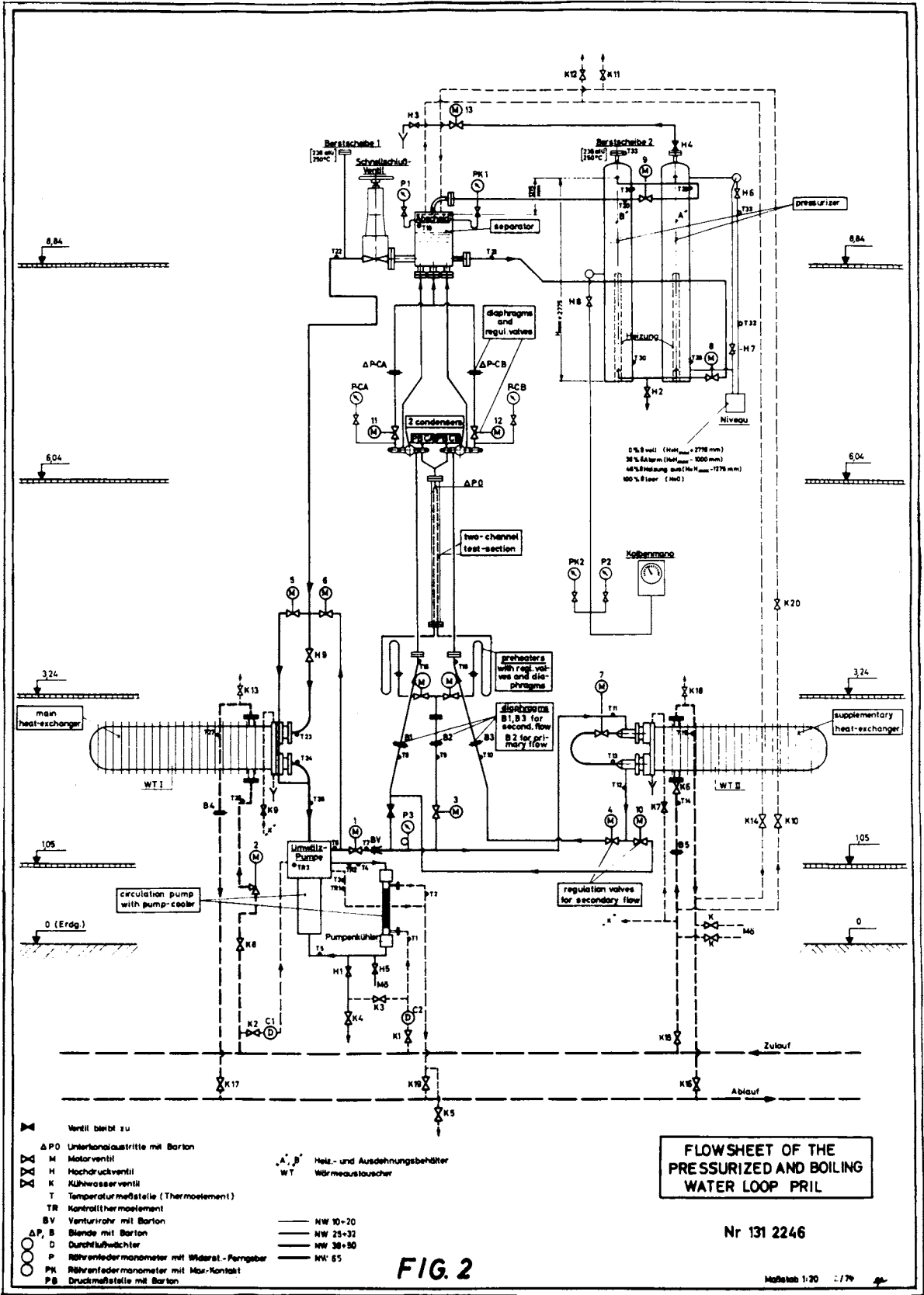


FIG. 1

PROGETTO	Übersichtsan
COMMISSIONE DELLE COMUNITA E ROPEE	Druck-u. Siedew.-Kwstf. (BOWAL)
CCR - EURATOM	3.09
STABILIMENTO DI ISPRA	
SCALA	
ELABORATO	
VERIFICATO	
CAPO D'OPERA	DATA 8.7.70
	1 3 1 2 1 9 4



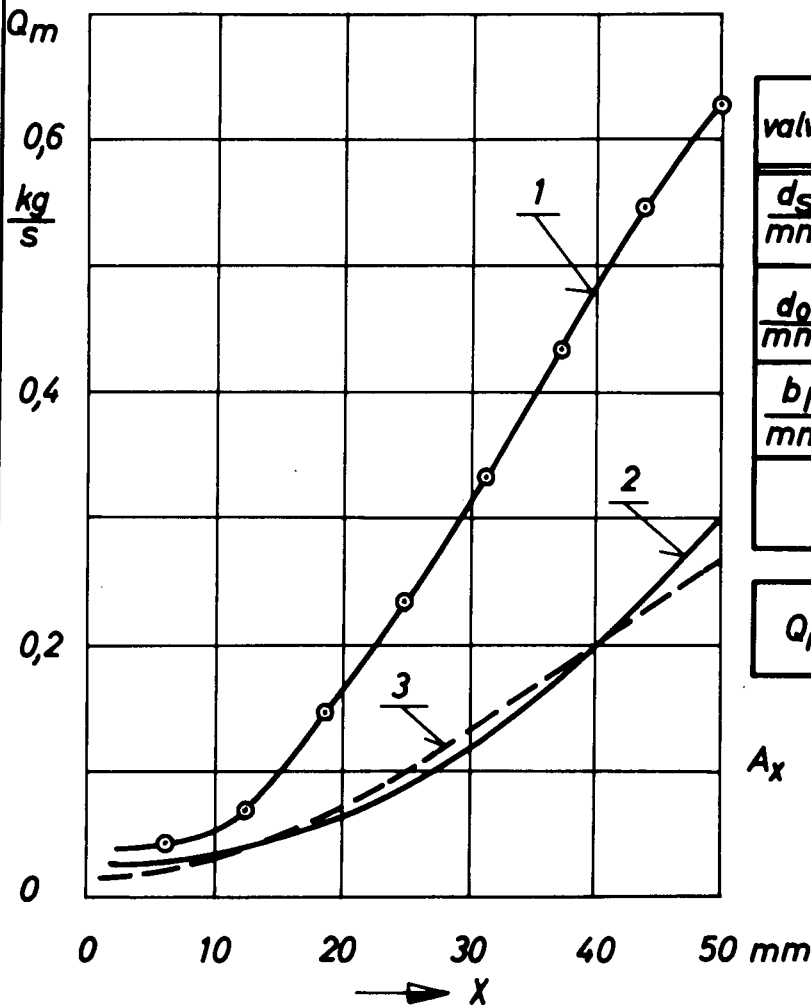
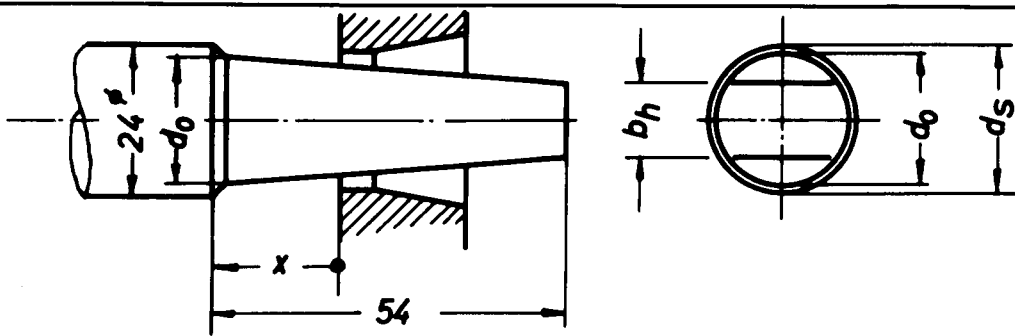
FLOWSHEET OF THE PRESSURIZED AND BOILING WATER LOOP PRIL

Nr 131 2246

FIG. 2

Maßstab 1:20

- ▼ Weill bleibt zu
- Δ PO Unterkanalaustritte mit Barton
- M Motorventil
- H Hochdruckventil
- K Kühlwasserventil
- T Temperaturmeßstelle (Thermoelement)
- TR Kontrollthermoelement
- BV Venturrohr mit Barton
- Δ P, B Blende mit Barton
- D Durchflußwächter
- P Röhrenfedermanometer mit Widerst.-Farngaber
- PH Röhrenfedermanometer mit Max.-Kontakt
- PB Druckmeßstelle mit Barton
- A, B Heiz- und Ausdehnungsbehälter
- WT Wärmeaustauscher
- NW 10+20
- NW 25+32
- NW 38+50
- NW 65



valve	1	2	3
$\frac{d_s}{mm}$	20,0	20,0	12,5
$\frac{d_0}{mm}$	19,8	19,8	12,3
$\frac{b_h}{mm}$	11,8	15,1	7,2
	measured	calculated	

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

A_x = flow-cross section

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

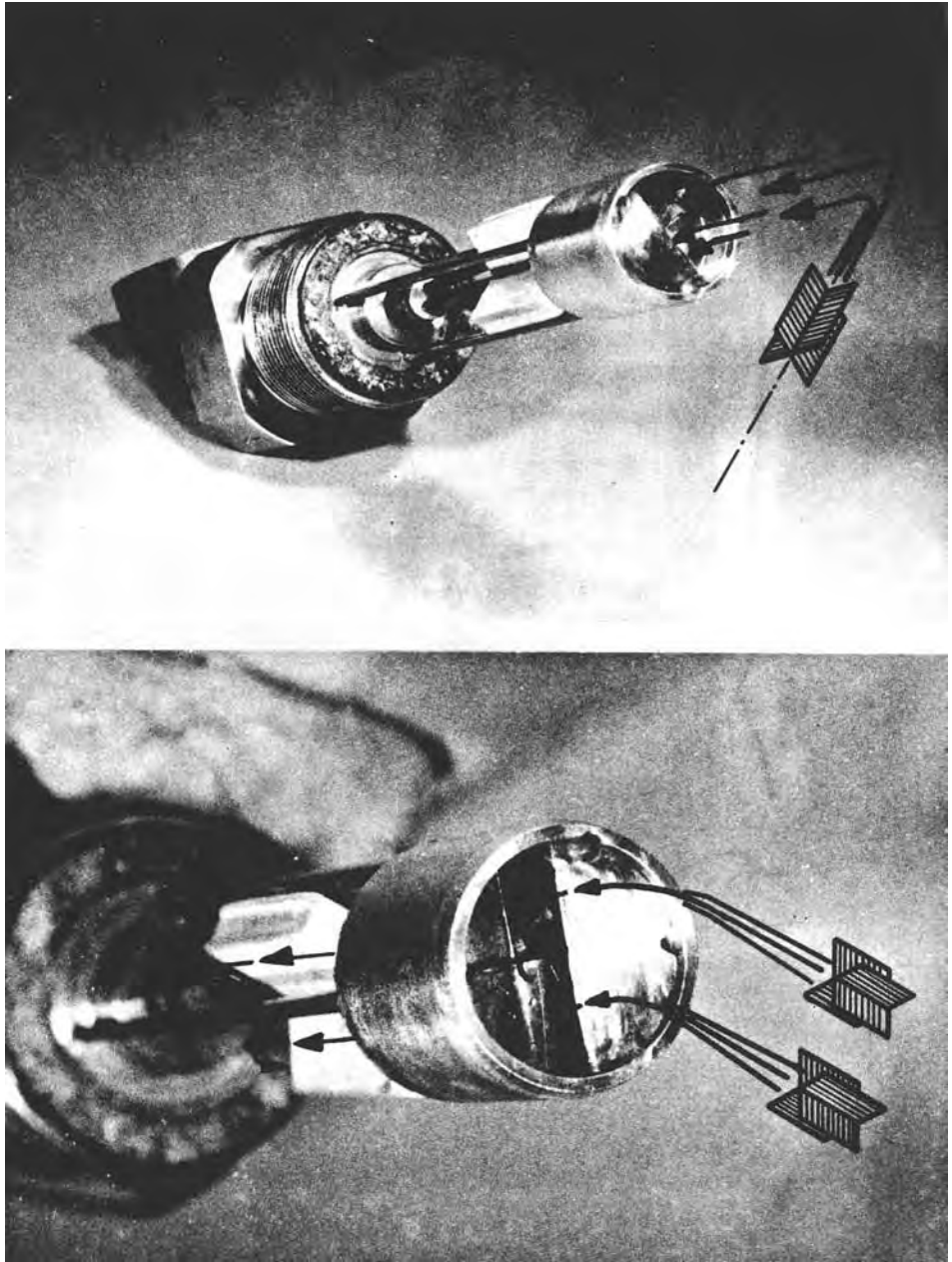


FIG. 4 *Flow mixing chamber and silver cross for the mean outlet temperatures of the condensers for the heat-balance measurements (outer diameter about 30mm, 4 turn-round sheets.)*

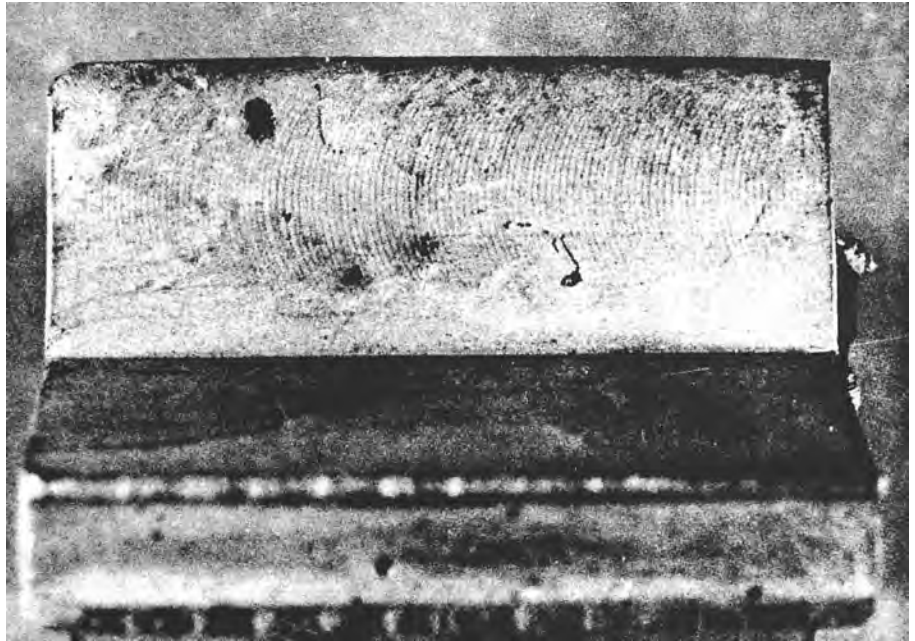


FIG. 5 *Cavity effects in the silver crosses for the mean temperatures of the condensers (length 30 mm, diameter about 30 mm, thickness of wing 6 mm.)*

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

1. General aim

Development of burnout correlations based on the critical quality-boiling length concept, utilizing both the bundle parameters and the subchannel parameters.

2. Particular objectives

Measurements of CHF data on BWR 16 rod test sections, 12 ft long, having different heat flux distributions.

3. Experimental facilities and programme

The experiments have been done in the CISE IETI-3 and IETI-4 loops of 8 MW, in the following range of parameters: pressure 70 ata, specific flowrate $G = 12-200 \text{ g/cm}^2 \text{ sec}$.

4. Project status

4.1. Progress to date

The measurements have been completed for radial power distribution: data for 6 different test sections were obtained. Measurements on a bundle having a cosine flux distribution are in progress and will be completed in mid 1977.

4.2. Essential results

Critical quality-boiling length parameter correlates well the data on a bundle parameters basis and a new calculation procedure, named ACHAB method, has been developed to predict the CHF margin in terms of power ratio.

The method, based on rod centered subchannel analysis predicts quite satisfactorily the influence of radial power distribution.

5. Next steps

It has been recognized the necessity of obtaining new round CHF data in order to optimize the rod-bundle CHF correlation. These CHF tests on round tubes, having the same heated diameters as the rod bundle subchannels, are now under way on the CISE IETI-1 facility.

<u>Title 1 (Original language)</u>	<u>Classification</u>
Esperienze di burnout su elementi di combustibile per BWR	10.3

6. Relation with other projects

CHF round tube experiments will provide data to be used in the theoretical developments of this programme.

7. Reference documents

- 1) G.P. Gaspari et al. " Experiments on the influence of Radial Heat Flux Distribution on Critical Heat Flux in BWR Bundle Geometry" Reactor Heat Transfer Meeting of Karlsruhe - October 9-11, 1973.
- 2) B. Marinelli et al. " Dryout Experiments in a 16 rod BWR Geometry with Six Different Radial Test Flux Distribution", 1975 Heat Transfer Conference S. Francisco, Cal. August 11-13, 1975.

8. Degree of availability

Open.

<u>Title 1 (Original language)</u> Esperienze di dryout con sezioni di prova 8x8 tipo BWR/6		<u>Classification</u> 10.3
<u>Title 2 (English)</u> Dryout (CHF) experiments in 8x8 BWR/6 test sections		<u>Country</u> ITALY <u>Sponsor</u> NUCLITAL <u>Organisation</u> NUCLITAL/CISE
<u>Date initiated</u> January 1977		<u>Project Leader</u> G.P. Gaspari (NUCLITAL)
<u>Date completed</u> June 1979		
<u>Last updating</u> May 1977		

1. General aim

Verification and development of project dryout (CHF) correlations applicable to steady state and transient analysis.

2. Particular objectives

Obtaining CHF experimental data by a test section simulating as close as possible the actual 8x8 fuel element as far as geometry, spacers, local and axial peaking pattern are concerned

3. Experimental facilities and programme

The experiments will be carried out on the CISE IETI-4 loop in the following ranges of conditions; pressure 40 - 100 bar , specific mass flowrate 50 - 2000 Kg/m² 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 1978 second half. The analysis completion is scheduled in June 1979.

<u>Title 1 (Original language)</u>	<u>Classification</u>
Esperienze di dryout con sezioni di prova 8x8 tipo BWR/6	10.3

6. Relation with other projects

These experiments follow a previous dryout programme carried out by a 4x4 rod bundle.

Results will be strictly related to a following programme, concerning operational and LOCA transient conditions, which is now under definition.

Void fraction measurements will be carried out in the same test section.

7. Reference documents

None

8. Degree of availability

Proprietary

<u>Title 1 (Original language)</u> Comportamento termoidraulico di un canale di potenza del reattore CIRENE nella fase di avviamento	<u>Classification</u> 10.3
<u>Title 2 (English)</u> Thermohydraulic behaviour of a CIRENE power channel during reactor start-up	<u>Country</u> : ITALY <u>Sponsor</u> : CNEN <u>Organisation</u> : CISE
<u>Date initiated</u> June 1975 <u>Date completed</u> December 1976 <u>Last updating</u> 1977	<u>Project Leader</u> UTM (CISE)

1. General aim: Thermohydraulic behaviour of a boiling power channel at positive inlet quality in the low total flowrate range both in steady state and transient conditions.
3. Experimental facilities and programme
 - 3.1. Experimental facility:
 - CIRCE: large scale facility simulating in a closed circuit 2 full-scale power channels; water flowrate: 22 kg/s, steam flowrate 3 kg/s; test section maximum power 12,5 MW (d.c.).
 - 3.2. Programme
 - 3.2.1. Steady state tests with positive quality at inlet to determine two-phase flow patterns and to measure critical power.
 - 3.2.2. Transient tests by reducing inlet steam flowrate from the steady state value to zero: measurements of flowrate, pressure, pressure drops, mass transients in the circuit.
4. Project status
 - 4.1. Progress to date: the experimental programme has been completed.
 - 4.2. Essential results: Defined information has been obtained as far as the minimum flowrate is allowed for a stable operation and the procedure to be followed during steam reduction phase.
7. Degree of availability: to a limited extent.

<u>Title 1 (Original language)</u> Esperienze di frazione di vuoto in sezioni di prova 8x8 tipo BWR/6		<u>Classification</u> 10.3
<u>Title 2 (English)</u> Void fraction experiments in 8x8 BWR/6 test sections		<u>Country</u> ITALY <u>Sponsor</u> NUCLITAL <u>Organisation</u> NUCLITAL/CISE
<u>Date initiated</u>	January 1977	<u>Project Leader</u> F. LUCCHINI (NUCLITAL)
<u>Date completed</u>	June 1979	
<u>Last updating</u>	May 1977	

1. General aim

Verification and development of design void fraction correlations applicable to steady state and transient analysis.

2. Particular objectives

Obtaining experimental data by a test section simulating as close as possible the actual 8x8 fuel element as far as geometry, spacers, local and axial peaking pattern are concerned.

3. Experimental facilities and programme

The experiments will be carried out in the CISE IETI-4 loop using the quick-closing valve technique.

The following test conditions will be investigated:

pressure 40 - 100 bar, specific mass flowrate 50 - 2000 Kg/m²s, outlet quality from 5% up to critical quality. Two local peaking patterns will be tested.

4. Project status

At present the 8x8 test section is under construction.

5. Next steps

The experiments are planned in the 1978 second half. The analysis completion is scheduled in June 1979.

<u>Titolo 1 (Original language)</u>	<u>Classification</u>
Esperienze di frazione di vuoto in sezioni di prova 8x8 tipo BWR/6	10.3

6. Relation with other projects

These experiments follow a previous void fraction programme carried out by a 4x4 rod bundle .

Results will be strictly related to a following programme concerning operational and LOCA transient conditions which is now under definition.

Dryout experiments will be carried out in the same test section.

7. Reference documents

None

8. Degree of availability

Proprietary

ENERGIEONDERZOEK CENTRUM NEDERLAND (ECN)		CLASSIFICATION: 10.3
TITLE: Burnout onderzoek aan 9-staafs bundels		COUNTRY: 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: April 1977	SCIENTISTS: H. Hoogland D.W. Middleton D. Werner
STATUS: nearly finished	COMPLETED: July 1977	

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 700 kW 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/m² 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

Completion of the study through a report, scheduled July 1977

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: 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: April 1977	SCIENTISTS: H. Hoogland A. Warmenhoven
STATUS: nearly finished	COMPLETED: December 1977	
<p><u>General aim</u></p> <p>Experimental investigation of the mixing phenomena in a bundle geometry.</p> <p><u>Particular objectives</u></p> <p>Experimental study of the heat and mass exchange between the subchannels in a bundle in which a nonhomogeneous heat distribution is generated.</p> <p><u>Experimental facilities and program</u></p> <p>Testloop for low-pressure experiments.</p> <p><u>Project status</u></p> <p>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.</p> <p><u>Next steps</u></p> <p>The obtained data show that for low mass flows mixing does not increase when a supporting grid is installed. A report of the measurement-results is planned in July 1977.</p> <p><u>Relation to other projects:</u> No</p> <p><u>Reference documents:</u> No</p> <p><u>Degree of availability:</u> Through ECN library</p> <p><u>Budget:</u> -</p> <p><u>Personnel:</u> -</p>		

PROJECT TITLE : DRYOUT IN BOILING CLUSTERS	CSNI INDEX NUMBER : 10.3
SPONSORING COUNTRY : UNITED KINGDOM	ORGANISATION : AEE WINFRITH
DATE INITIATED : 1966 DATE COMPLETED :	PROJECT LEADER : D H LEE

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, and heat transfer coefficients in post-dryout regime.

Reference Documents

Internal documents.

Obertelli and Owen AEEW-M935 (Unclassified)
AEEW-M1386 (1976) Unclassified

OECD NUCLEAR ENERGY AGENCY - 38, Boulevard Suchet, 75016 PARIS
Telex : 630668 AEN NEA - Tel : 524 82-00

31.719

PROJECT TITLE : <u>Basic studies of two phase mixing in fuel cluster geometries.</u>	CSNI INDEX NUMBER : LWR 10.3
SPONSORING COUNTRY : Commission of the European Communities	ORGANISATION : JRC - Ispra
DATE INITIATED : 1973 DATE COMPLETED : 1977	PROJECT LEADER : H. Herkenrath

Description :

1. General aim

Basic studies of thermohydraulic mixing between the subchannels of a boiling fuel cluster.

2. Particular objectives

To examine boiling mixing (steady state and transient condition in a 16 rod cluster (BOWAL loop). Validation of cluster thermo-hydraulics codes. Calibration of different void meters for steady state and transient conditions (PRIL loop).

3. Experimental facilities and programme

3.1 PRIL loop

This loop consists of :

- controllable forced circulation
- partial blowdown (~20 %) facility for the transient tests
- pressurizer, condensor, subcooler

The characteristics of PRIL are :

- Maximum power : 600 KW
- Maximum pressure : 200 bar
- Maximum temperature : 365°C
- Special Instrumentation : different void meters.

3.2 BOWAL loop

This loop consists of :

- controllable forced circulation
- partial blowdown (20 %) facility for the transient tests

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- pressurizer, condensor, subcooler and preheater (500 KW)

The characteristics of BOWAL are :

- Maximum power : 3,6 MW
- Maximum pressure : 250 bar
- Maximum temperature : 365°C
- Special Instrumentation : distribution of temperature and power along a number of the rod pressure distribution along the test section. Outlet enthalphy distribution among the 6 basic subchannels of the cluster.

4. Project status

4.1 Progress to date

The calibration tests showed the necessity of several modifications, mainly in the regulation and measuring device. The experimental studies under steady state conditions with the BWR-test section (70 bars) have been started in January 1977.

5. Next steps

Completion of the steady state measurements with the BWR test section.
Calibration and mounting of 6 x-ray void meters for studies under transient conditions.
Construction of a PWR test section (160 bars).

6. Reference documents

- JRC Safety programme progress report 1975 and 1976.
- Herkenrath, H. and W. Hufschmidt
"The pressurized and boiling water loops BOWAL and PRIL for boiling mixing studies of the Heat Transfer Division of JRC, Ispra".
EUR-Report (to be published).
- Herzberger, P., W. Hufschmidt and F. Wind,
"Die Verwendung von Gamma - oder Röntgenstrahlen zur Messung des Dampfgehaltes in Zweiphasenströmungen".
EUR 5415. d (1975).

10.4. Mechanical effects (e.g. vibration)

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

Objectif général :

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

Objectifs particuliers :

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

Installations expérimentales et programme :

Boucle à eau (Gascogne) DEMA

Etat de l'étude :

- 1) Avancement à ce jour :

ETUDES DE SINGULARITES D'ECOULEMENT -

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

ETUDE DE TUBES EN ECOULEMENT TRANSVERSAL -

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

2) Résultats essentiels :

a) Singularités :

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

b) Ecoulement transversal :

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

Prochaines étapes :

Les expériences sur deux barreaux seront effectuées courant 77.

Documents de référence :

Rapport EMT/76/40 : "Etude des fluctuations de pression dans les circuits parcourus par des fluides - sources de fluctuations engendrées par les singularités d'écoulement (Théorie Générale - Formulaire et abaques)".

- "Etude expérimentale de deux singularités d'un circuit" (Note C.E.A. 1735)
- "Sources acoustiques associées à la réunion de deux écoulements dans un circuit" (Rapport EMT/75/13).

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

1. General aim

The object of this project is the detection of impending damage to the core components and in the primary circuit of the cooling system at an early stage. An essential feature of this measuring system is that the transducers are located outside of the reactor pressure vessel and the remaining cooling system, and that information can thus be obtained on the behaviour of the core components and the primary circuit during operating, with simultaneous monitoring thereof.

2. Particular objectives

The particular objectives and main emphasis of the work are:

- a) Study of parameters on an experimental loop for a Leak Detection and Monitoring System based on acoustic emission.
- b) Development of an enlarged Loose Parts Monitoring System (LMPS), with which loose and loosened parts can be detected and located.

3. Research program

The research program for the study of parameters for the leak monitoring

system embraces a total of eighty experiments:

Variation of the following parameters was carried out:

Four types of leaks	leaking flange, leaking valve, crack, nozzle throat
Two leakage rates:	240 l/h, 24 l/h
Three enthalpies:	steam, water/steam, water
Underground noise:	yes/no

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

4.1 Experimental facilities

For the study of parameters, the Kraftwerk Union Erlangen has made available an experimental loop, which made it possible to simulate leaks under specific reactor conditions.

4.2 Computer codes

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

For analysing the noise of leakages, the power spectra are calculated, which are issued via display or X-Y recorder and are also stored on discs for subsequent recall.

5. Progress to date

In cooperation with the Kraftwerk Union Erlangen and the Battelle-Institut Frankfurt, the research program on leaks was implemented. The noises of different types of leakages were recorded on magnetic tapes and were then analysed in off-line procedure. During the course of the tests, television recordings were also made on magnetic video tapes for optical documentation of the water or steam jet.

6. Results

The analyses have not yet been completed.

7. Next steps

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

8. Relation with other projects

Battelle-Institut Frankfurt:

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

9. References

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

Quarterly research reports:

IRS-F-27

IRS-F-28

IRS-F-30

IRS F-31

IRS-F-33

Annual report 1975: IRS-F-29

10. Degree of availability of the reports

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

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

I. General aim.

Apparatus and plants using large quantities of fluids are often damaged by flow-included vibrations. It is not possible by theory to gain information concerning the best way of laying out a mechanical structure to reduce or eliminate vibrations. Under real conditions the flow pattern itself and the geometry defined by the mechanical structure are very complicated. It therefore remains for the experimental engineer to explain the relationship between the movement of the mechanical structure and the excitation caused by the flow. For a systematic investigation in the planning stage experiments making use of models can also be carried out complementary to measurements taken at the plant itself. Model technology is justified theoretically in mechanics of similarity.

II. Particular objectives

Under investigation are the flow-induced vibrations of the core components in the reactor pressure vessel of PWR caused by the flow of the main coolant. The measurements are taken of pressure vessel models (scale ~ 1:10) of normal construction (4-loop-plant). The main goal is to obtain criteria for a safe design of core components and to test if model experiments in the project stage of new plants are suited to predict damages caused by flow-induced vibrations. To solve this problem it is necessary to know the response of the

structure according to the dynamical load caused by the main coolant flow. For a safe design of a mechanical structure it is important to be sure that there is no dominant feedback-mechanism (fluid-resonant, fluid-dynamic, fluid-elastic) in the flow field; the effect of feedback is always a selective amplification of the load energy about dominant frequency bands associated with an increase of spatial correlation. Without feedback-mechanism the load energy in a turbulent flow, which causes the mechanical structure to vibrate, has a statistical distribution being unable to excite the structure systematically.

III. Research Program

Figure 1 gives a diagrammatic representation of the investigation program: Using only the hydrodynamic analogy of the flow, the structure of the model should be rigid, because it does not correspond in its elastic properties to the actual installation. There are dimensionless numbers enabling a transfer of measurement values taken at the model to the actual reactor-pressure-vessel. In this model it is possible to investigate fluid-dynamic and fluid-resonant feedback - whenever they exist in the hydrodynamic system.

If one takes into account not only the hydrodynamics but also the kinetics of the core components one obtains a hydroelastic similar model. This model allows the investigation of fluid-elastic feedback between structure and fluid. The difficulty of this model is that six dimensionless numbers and the boundary conditions of the flow and the structure must be fulfilled.

It is provided that the measurements taken at the model will be compared with those of the original reactor.

IV. Experimental facilities; computer codes

A.) Experiments conducted on the hydrodynamically similar model

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

B.) Experiments conducted on the hydroelastically similar model

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

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

C.) Computer codes

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

V. Progress to date

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

VI. Results

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

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

shown in figure 3, which shows a spectrum where the geometry of the gap between core barrel and pressure vessel has been changed by a ram ledge (Stauleiste). It is obvious that this variation has a great influence on the frequency content of the pressure signals.

During 1976 the core barrel and the pressure vessel with which the hydro-elastic model is to be carried out has been completed. In manufacturing the core assembly of the model very detailed designung of the components was necessary. The instrumentation has been initiated.

VII. Next steps

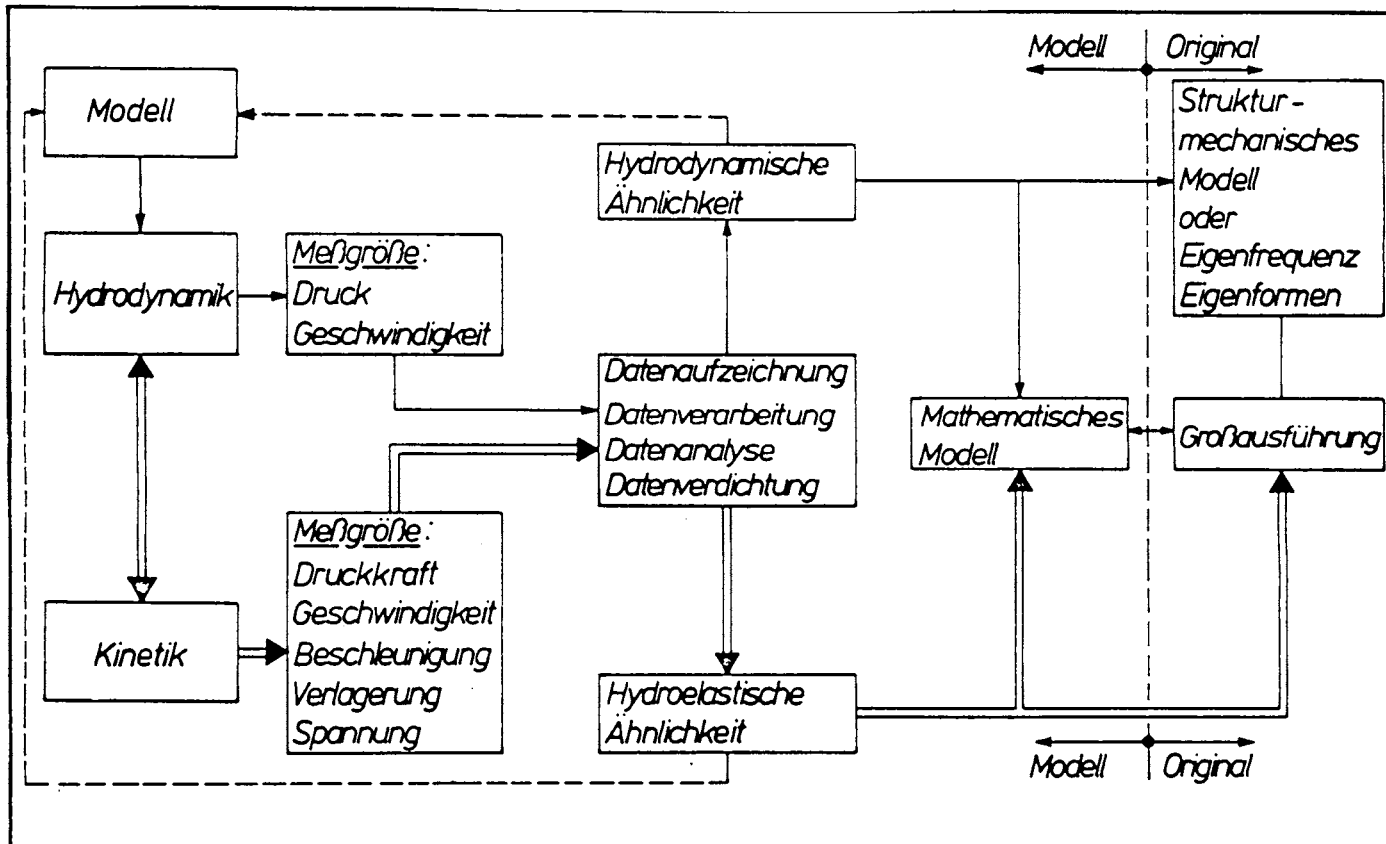
- a) Preliminary conclusion of the experiments with air as the flow medium
- b) Experimental determination of the mechanical properties of the hydro-elastic model, i.e. eigenfrequencies, damping etc.
- c) Measurements with the hydro-elastic model.
- d) Development of a semi-empirical model to calculate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

VIII. Relation with other projects

See this report for 1974

IX. References

Annual reports for the IRS/BMFT (German)



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

Abb. 1

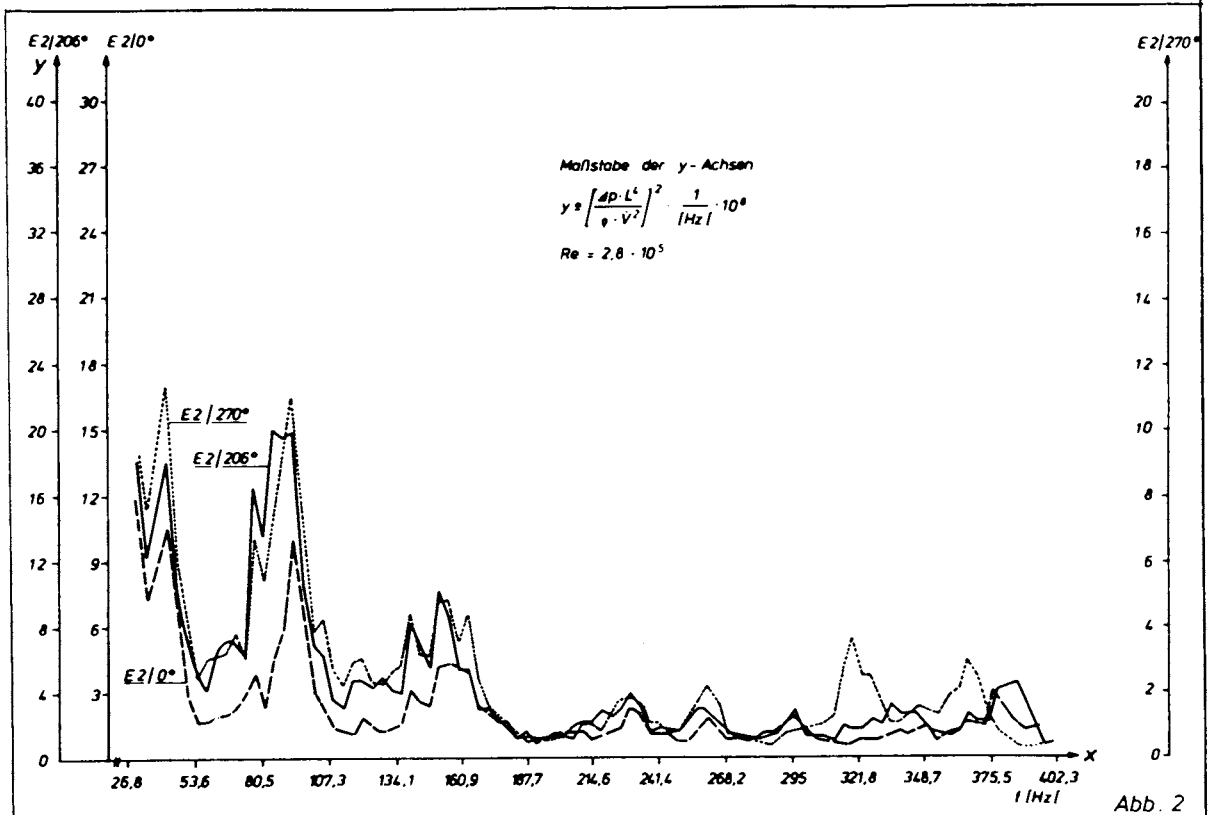


Abb. 2

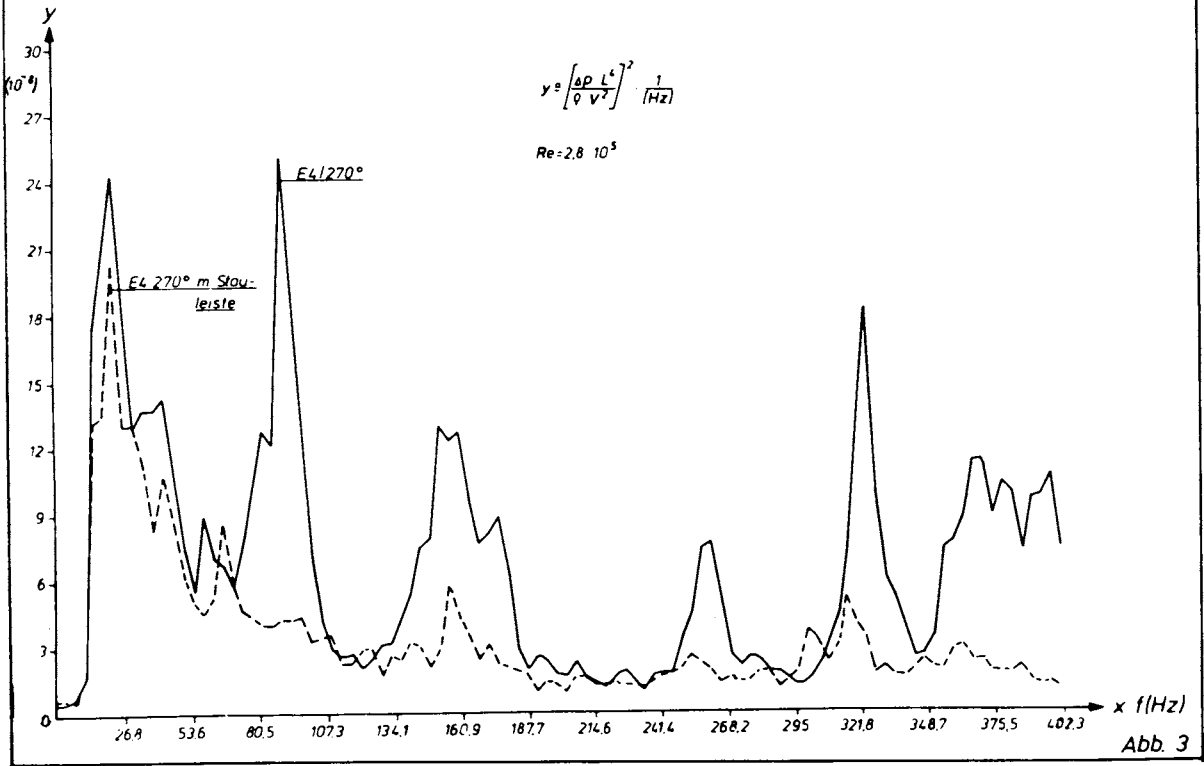


Abb. 3

<u>Classification:</u> 10.4								
<u>Title 1 (Original Language):</u> Experimentelle Untersuchungen zur Absicherung von Druckbehältereinbauten gegen strömungsinduzierte Schwingungen beim Siedewasserreaktor (RS 218 - II.4.1, Jahresbericht A 76)	COUNTRY: BRD							
	SPONSOR: BMFT							
	ORGANIZATION: KWU, Karlstein							
<u>Title 2 (English):</u> Experimental Investigation of Protection of Pressure Vessel Internals Against Flow-Induced Vibrations in BWRs	<u>Project Leader:</u> Dr. Simon							
	<table border="0" style="width: 100%;"> <tr> <td><u>Initiated (Date):</u></td> <td><u>Completed (Date):</u></td> </tr> <tr> <td>1. 9. 76</td> <td>31. 5. 77</td> </tr> <tr> <td><u>Status:</u></td> <td><u>Last Updating (Date):</u></td> </tr> <tr> <td>Continuing</td> <td>31. 12. 76</td> </tr> </table>	<u>Initiated (Date):</u>	<u>Completed (Date):</u>	1. 9. 76	31. 5. 77	<u>Status:</u>	<u>Last Updating (Date):</u>	Continuing
<u>Initiated (Date):</u>	<u>Completed (Date):</u>							
1. 9. 76	31. 5. 77							
<u>Status:</u>	<u>Last Updating (Date):</u>							
Continuing	31. 12. 76							

1. General Aim

The project consists of investigations of the effect of higher fluid velocities at the pump outlet on the vibration behaviour of RPV internals in the BWR core inlet plenum, in which the advanced internal BWR coolant recirculation pump has been installed.

2. Particular Objectives

Protection by experimental means of the currently constructed control rod guide tubes and in-core instrumentation tubes considering their vibrational behaviour.

3. Research Program

- 3.1 Overall Program
- 3.2 Pre-tests on a 1 : 5 scale model
- 3.3 Preparation for tests
- 3.4 Vibration measurements at the Karstein test stand
- 3.5 Evaluation of tests
- 3.6 Measurements in a 1300 MW standard plant

4. Experimental Facilities

Pre-tests on a BWR flow model (1 : 5 scale) will be performed in order to make sure that representative flow fields are present in the Karlstein test stand. It is planned to investigate in Karlstein 11 original control rod guide tubes including their supports and with dummy control rods, and 4 in-core instrumentation tubes, in the most adverse position with respect to vibrational effects, which are the 1st and 2nd control rod guide tube rows.

5. Progress to Date

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

6. Results

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

7. Next Steps

The following measurements are planned according to the test program:

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

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

8. Relation with Other Projects

9. References

10. Degree of Availability

<u>Title 1 (Original language)</u> Comportamento vibrazionale della colonna di combustibile Cirene - Fretting corrosion.	<u>Classification</u> 10.4. HWR
<u>Title 2 (English)</u> Dynamics of fuel strings in axial flow - Fretting corrosion. (HWR - Calandria tubes)	<u>Country</u> ITALY <u>Sponsor</u> CNEN <u>Organisation</u> CNEN-CISE
<u>Date initiated</u> 1974 <u>Date completed</u> In progress <u>Last updating</u> March 1977	<u>Project Leader</u> G. LELLI

1. General aim

Optimization of fuel bundles for HWRs in order to minimize fretting corrosion damages.

2. Particular objectives

Reduction of the vibrational amplitude of fuel strings under coolant flow.

3. Experimental facilities and programme

The experimental facility consists of a full scale adiabatic loop. The experimental programme consists in the analysis of the vibrational behaviour of fuel strings and of the fretting corrosion marks both on the bundles and on the pressure tubes. The experimental results will be useful also to qualify mathematical models developed in order to simulate the fuel strings behaviour.

4. Project status

A few preliminary results show a good vibrational behaviour of the fuel string and few minor fretting-corrosion marks. The above mentioned encouraging goal is especially due to the improvements obtained in the assembling of the fuel bundles.

5. Next steps

One more test in the adiabatic loop will be carried out. In pile experiments are planned.

6. Relation to other projects

Optimization of mechanical design of fuel bundles for HWR.

7. Reference documents

- G. Lelli "Computer modelling of the dynamics of a string of fuel bundles in axial flow. Part B: response to a generalized forcing function and further developments". July 1975. CRNL-1339.

<u>Title 1 (Original language)</u>	<u>Classification</u>
Comportamento vibrazionale della colonna di combustibile Cirene - Fretting corrosion	10.4. HWR

- G. Lelli "A comparison of experimental data, on the natural frequencies and mode shapes for Split-Spacer and LS-3 bundles, with the calculated data from an analytical model". Jan. 1975. Internal report.
- J. Fedoruk, G. Lelli "A comparison of Experimental Mobility data for Split Spacer and LS-3 Fuel bundles with calculated values from an analytical model". August 1975. Internal report.
- Z. Arcangeli, G. Colombo, G. Lelli, M. Vignolini "Misure comparate di deformazione a parallelogramma di fasci di combustibile Cirene". Aprile 1976. Internal report.
- Z. Arcangeli, G. Lelli "Studio del regime vibrazionale della colonna di combustibile Cirene. Frequenze proprie. Effetto della rigidità". Giugno 1976. Internal report.
- G. Lelli, M. Vignolini "Verifica di un valore massimo della spinta esercitata dalla molla di compressione della colonna di combustibile Cirene. Considerazioni relative alla fretting corrosion ed allo stato di sollecitazione delle piastre di estremità dei fasci". Febbraio 1977. Internal report.

8 - Degree of availability

To a limited extent.

N.V. KEMA

Arnhem, 25 februari 1976

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

1. General aim

On load surveillance of vital systems.

2. Particular objectives

Dynamic behaviour of a BWR.

3. Experimental facilities

Dodewaard nuclear power plant.

4. Project status

Still in progress.

5. Next steps

Not applicable.

6. Relation with other projects

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

7. Reference documents

None.

8. Degree of availability

Through the organizations KEMA and RCN.

11. MATERIAL AND MECHANICAL PROBLEMS UNDER NORMAL AND ACCIDENT CONDITIONS (e.g. load following, turbine trip, blowdown, etc.)
(no entries)

- 11.1 Fuel elements and core (e.g. fuel densification, fuel pin distortion, cladding ballooning, cladding oxidation, cladding embrittlement, cladding water reaction and rupture)

PROJECT TITLE : Deformation and Fracture Behaviour of Reactor Materials	CSNI INDEX NUMBER : 11.1
SPONSORING COUNTRY : Australia	ORGANISATION : Australian Atomic Energy Commission, Private Mail Bag, Sutherland, NSW, 2232
DATE INITIATED : July 1970 DATE COMPLETED : Continuing	PROJECT LEADER : Dr. K.U. Snowden Materials Division

Description :

Objective:

To study the effects of temperature, neutron irradiation and stress on zirconium alloys and steels used for structural purposes in reactors.

Proposed Investigations:

The programme consists of systematic studies of the effects of creep and fatigue stressing on various alloys. Particular attention will be given to fracture behaviour and the conditions required to induce unstable crack growth under static and cyclic stress conditions.

PROJECT TITLE : STOCHASTIC ANALYSIS OF THE TEMPERATURE DISTRIBUTION IN CANNED CYLINDRICAL FUEL ELEMENTS	CSNI INDEX NUMBER : 11.1 LWR HWR
SPONSORING COUNTRY : AUSTRALIA	ORGANISATION : School of Nuclear Engineering University of New South Wales
DATE INITIATED : July 1976 DATE COMPLETED :	PROJECT LEADER : Dr. P.R. Barrett

Description :

Theoretical analysis of the temperature distribution in canned cylindrical fuel elements, including stochastic effects of gap conductance, fuel composition, and fuel pellet axis location. This project is a continuation of work on stochastic modelling of fuel pellet and fuel element behaviour.

PROJECT TITLE : STOCHASTIC THERMAL STRESS ANALYSIS OF CANNED CYLINDRICAL FUEL ELEMENTS	CSNI INDEX NUMBER : 11.1 LWR HWR
SPONSORING COUNTRY : Australia	ORGANISATION : School of Nuclear Engineering University of New South Wales
DATE INITIATED : January 1974 DATE COMPLETED : June 1976	PROJECT LEADER : Dr. P.R. Barrett

Description :

Theoretical analysis of randomly cracked fuel pellets, involving temperature and stress fields in the pellet and the contacting fuel can.

PROJECT TITLE : Thermo-mechanical analysis of nuclear plant structures	CSNI INDEX NUMBER : 11.1, 11.5
SPONSORING COUNTRY : Australia	ORGANISATION : University of N.S.W.
DATE INITIATED : ? DATE COMPLETED : Continuing	PROJECT LEADER : Associate Professor Z. J. Holy

Description :

Research into the modelling and optimal numerical solution of linear and non-linear problems of thermal and mechanical response of nuclear power plant components and structures.

The project is of a theoretical nature. The programme for 1977 is a continuation of work on the response of fuel elements to pressure pulse loading.

The work involves interaction with AEC engineers, and receives some support from the Australian Institute of Nuclear Science & Engineering.

PROJECT TITLE : Zircaloy-Steam Reaction at High Temperatures	CSNI INDEX NUMBER : 11/1
SPONSORING COUNTRY : Australia	ORGANISATION : Australian Atomic Energy Commission, Private Mail Bag Sutherland, NSW, 2232
DATE INITIATED : January 1973 DATE COMPLETED : June 1976	PROJECT LEADER : Dr. D. Bradhurst Materials Division

Description :

Objective:

To study the Zircaloy-Steam reaction under conditions relevant to loss of-coolant conditions in water-cooled reactors, including the effects of deformation on the reaction.

Results:

The effect of deformation before and during the Zircaloy-steam reaction at 1060°C has been investigated. The reaction was also studied during simulated thermal transients, up to 1300°C. The results indicate that an allowance for deformation-enhanced oxidation should be included in loss-of-coolant-accident analysis where Zircaloy-2 cladding is involved.

Publications:

1. "The Effects of Deformation on the High Temperature Steam Oxidation of Zircaloy-2", D.H. Bradhurst and P.M. Heuer, J. Nucl. Mater. 55 (1975) 311-326.
2. "The Effects of Deformation on the High Temperature Oxidation of Zirconium Alloys", D.H. Bradhurst and P.M. Heuer, Paper 6-4, 6th. Int. Cong. Met. Corrosion, Sydney, Dec. 3-9, (1975).

OECD NUCLEAR ENERGY AGENCY - 38, Boulevard Suchet, 75016 PARIS
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14.560

		Classification 11.1
<u>Title 1</u> Mixed oxide fuel element behaviour under irradiation		<u>Country</u> : Belgium
<u>Title 2</u>		<u>Sponsor</u> : BELGONUCLEAIRE
<u>Initiated</u> : 1972 <u>Completed</u> : -		<u>Organisation</u> : BELGONUCLEAIRE-CEN/SCK
<u>Status</u> : in progress <u>Last updating</u> : June 1977		<u>Project leader</u> : Mr. H. Bairiot.
1.	<u>General Aim</u> Study of mixed oxide fuel element behaviour under irradiation. Post-irradiation examination.	
2.	<u>Particular Objectives</u> 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.	<u>Experimental facilities and programme</u> - In-pile (BR-2 reactor) and out-of-pile tests for mixed oxide fuels. Neutron radiography. - VENUS critical facility for study of local power peaking effects with simulated axial gaps. - Fuel assemblies irradiated in BR-3, SENA, DODEWAARD, GARIGLIANO, etc...	
4.	<u>Project status</u> Progress to date : VENUS tests, irradiations in BR-2, BR-3 and SENA. Essential results : Power peaking effects, burnable poison effects, fuel-clad interaction, fission products.	
5.	<u>Next steps</u> : -	
6.	<u>Relation with other projects</u> : -	
7.	<u>Reference documents</u> : -	
8.	<u>Degree of availability</u> : Proprietary.	

Titre Conséquence d'une rupture de gaine sur les combustibles CAMEL.	Pays : FRANCE
	Organisme directeur : CEA
Titre (anglais) Effect of a cladding rupture on CAMEL fuel element behaviour.	Organisme exécuteur : CEA/DMECN-SECS
	Responsable : G.LESTIBOUDOIS (SECS)
Date de démarrage : 01/01/77 Etat actuel : en cours	Date prévue d'achèvement 31/12/80 Dernière mise à jour : 01/03/77 Scientifiques : G.DECONTENSON B.HOUDAILLE

Objectif général :

Etudier les problèmes de détection des ruptures de gaines d'un élément combustible CAMEL.
Etudier l'évolution d'une rupture de gaine sur ce type d'élément.

Objectifs particuliers :

Vérifier comment réagissent les dispositifs courants de détection de rupture de gaine dans le cas des CAMEL.
Vérifier s'il y a dégradation du combustible ou du gainage.

Installations expérimentales et programme :

Les essais sont envisagés à SILOE et à OSIRIS.
Les dispositifs à utiliser seront les suivants : BOUILLEUR, IRENE, BOUFFON.
Le programme 1977 sera limité à 2 essais dans le dispositif BOUFFON.

Etat de l'étude :

1) Avancement à ce jour :

Préparation des irradiations en cours.

2) Résultats essentiels :

Néant.

.../...

Prochaines étapes :

Irradiation du premier crayon en février 1977.

Relation avec d'autres études :

Etude des limitations de fonctionnement des réacteurs à eau à combustibles CAMEL.

Titre Limitation du fonctionnement des éléments combustibles dues à l'existence de ruptures de gaine.	Pays : FRANCE Organisme directeur : CEA
Titre (anglais) Operational limits of fuel elements due to cladding defects.	Organisme exécuteur : CEA/DMECN-SECS (Saclay) Responsable : G.LESTIBOUDOIS
Date de démarrage : 01/01/72 Date prévue d'achèvement : 31/12/77 Etat actuel : en cours Dernière mise à jour : 01/03/77	Scientifiques : G.DE CONTENSON B.HOUDAILLE

Objectif général :

Déterminer le comportement d'un crayon combustible des PWR présentant une rupture de gaine.
 En déduire éventuellement des limitations à ses conditions de fonctionnement.

Objectifs particuliers :

Etudier les problèmes d'hydruration des zircaloy en présence de vapeur d'eau.
 Rassembler des résultats sur le dégagement des produits de fission à partir de crayons défectueux.

Installations expérimentales et programme :

Les essais d'irradiation sont effectués à SILOE.
 Les dispositifs expérimentaux sont appelés BOUFFON.
 Les diverses irradiations sont effectuées à des niveaux de puissance variable et pendant des temps différents.

Etat de l'étude :

1) Avancement à ce jour :

- De nombreuses difficultés de mise au point du dispositif ont retardé les programmes. Des essais ont été réalisés dans de bonnes conditions, avec des flux thermiques superficiels, sur la gaine, étagés entre 140 et 180 W/cm².

.../...

2) Résultats essentiels :

Les essais n'ont pas montré, contrairement à ce que l'on pouvait penser, qu'il se produisait une détérioration rapide de la gaine par attaque hydrurante lorsque le crayon fonctionnait avec une rupture de gaine ouverte.

Prochaines étapes :

Vérification, par examen en cellule chaude, des conditions dans lesquelles se développent les couches d'oxyde à l'intérieur des gaines en zircaloy, et la répartition des hydrures aussi bien à l'intérieur qu'à l'extérieur.

Relation avec d'autres études :

Etude des limitations des conditions de fonctionnement des centrales de puissance.

Documents de référence :

- "Examen irradiation de BOUFFON 08", DTECH-SECS, Compte-Rendu d'examen N° 45.
- "Examen après irradiation de BOUFFON 09", Note Technique SECS/SEECRE-704.
- "Emission de produits de fission par un crayon présentant une rupture en fabrication - BOUFFON 09J49", Compte-Rendu DMG 35/76.

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

1. General Aim

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

2. Particular Objectives

Influence exerted by the oxygen potential (O/U-ratio) of UO_2 on the strain and rupture behavior of short Zry-4-cladding tubes under LOCA-conditions.
 UO_2 /Zry-reaction experiments under PCM-(power cooling mismatch) conditions. Study of the chemical interactions between fission products and Zry.

3. Experimental Facilities

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

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

4. Project Status

4.1 Progress to Date

The tubular specimens were introduced in a furnace kept at a steady-state temperature

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

The UO₂/Zry-reaction experiments are performed at temperatures up to 1400°C. The UO₂/Zry-contact pressures varied between 0 and 70 bar.

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

4.2 Essential Results

Whereas at low internal pressures hyperstoichiometric UO₂ causes a Zry cladding strengthening due to the oxygen uptake, at high internal pressures (> 15 bar) the influence of the oxygen potential of the UO₂ (O/U-ratio) on the strain and rupture behavior of the Zry cladding tubes is clearly reduced. At large differential pressures the heating rate and the pressure plot have practically no influence on the burst temperature of the cladding tubes. In accordance with data from the literature the burst strains of the tubular specimens vary in argon atmosphere between 20 and 130% and show a minimum in the two-phase region of Zry at about 930°C.

Initial UO₂/Zry-reaction experiments performed under PCM-conditions show that the chemical interactions depend on the contact pressure at the UO₂/Zry-phase boundary. Under good contact conditions UO₂ is reduced above 1000°C by Zry with the simultaneous formation of α-Zr(O), uranium and a (U,Zr)-alloy.

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

5. Next Steps

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

Influence of fission products and thin oxide layers on the chemical interactions of UO₂ and Zry under PCM-conditions.

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

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

6. Relation with other Projects

PNS 4235.1 and 4235.2

7. Reference documents

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

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

8. Degree of availability

Unrestricted distribution

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

1. General Aim

Power ramp tests in HFR Petten are being carried out with PWR and BWR standard fuel rods of different production parameters, in order to investigate the reason of fuel rod defects, especially the fuel-cladding tube interaction.

2. Particular Objectives

The mechanism causing fuel rod defects by alternating power loads and positive power ramps is not yet well known. In practice fuel rod defects have been discovered after local power variations in power reactors and in power ramp tests at Halden, Studsvik and Risø.

In order to study this problem, the irradiation and load change behaviour of PWR and BWR fuel rods with oxide fuel (UO_2 and UO_2/PuO_2 respectively) is being investigated in HFR Petten. The fuel rods assembled to segmented rods have been or are still being preirradiated in the Obrigheim and Würzgassen power stations up to 5.000 - 24.000 MWd/t(U) burn-up.

3. Program

- 3.1 Extraction and sectioning of segmented PWR and BWR fuel rods, intermediate inspection of the fuel rod segments/test fuel rods.
- 3.2 Neutronradiografic investigations on the test fuel rods.
- 3.3 Installation of the test fuel rods into the irradiation capsules and performance of power ramp experiments in HFR Petten.
- 3.4 Post-irradiation examination of the test fuel rods.
- 3.5 Intermediate evaluation of the power ramp experiments and report to the results.

4. Experimental Facilities

The test fuel rods have been or are still being preirradiated in the Obrigheim and Würgassen power stations for 1 - 3 cycles. In the pool-side-facility of the HFR Petten power ramps will be carried out up to linear heat rates $q' \leq 620$ W/cm with ramp rates $\dot{q}' \leq 100$ W/cm min, and heat rate increases $\Delta q' \leq 350$ W/cm. The test capsules will be instrumented with inductive elongation detectors in order to measure changes of the fuel rod length. The tests will be performed as a joint project of KFA Jülich and Kraftwerk Union.

The following parameters will be varied:

fuel:	UO ₂ , UO ₂ /PuO ₂ density, pellet geometry
cladding tube:	production parameters (e.g. graphite on inner surfaces)
fuel rod:	pressurization, burn-up

5. Project Status/ Progress to Date

The segmented fuel rods No. 1902, 1201, 1202, 1203 and 1204 were sectioned and inspected in the hot cells at Karlstein. The segments were transported to Petten.

6 preirradiated segmented fuel rods with UO_2 and UO_2/PuO_2 pellets were unloaded from Obrigheim.

Several test fuel rods were ramped in in-situ and start-up ramp tests; the linear heat rates were increased from 300 W/cm to 460 - 608 W/cm with different ramp rates.

Finally the fuel rods were investigated by neutronradiography, γ -scanning and eddy current tests. The diameter was measured.

2 not irradiated fuel rods were transported to Petten, 49 segments preirradiated in KWO were transported from Obrigheim to the hot cells in Karlstein. The transport of 10 ramped fuel rods from Petten to Karlstein was prepared.

The evaluation of the first results was started.

6. Project Status/ Essential Results

No fuel rod defects were detected (no fission products were found in the coolant of the irradiation capsule) during the slow or fast start-up ramps. The eddy current inspection however analysed some influences of the ramps. The signals indicated circumferential ridges of the cladding tubes. No signals were found, which were due to cladding defects.

Only during the in-situ ramp IS 5 water has penetrated through the cladding and has filled half the bottom plenum.

The neutronradiography showed that at high heat rates the dishings of the UO_2 -pellets have nearly been closed and that some cracks existed in the pellets.

The γ -scanning (La-140) indicated that during the ramp tests more than 60 % of the active rod length have been in the ramp of 90 % - 100 % maximum load.

The length of the fuel rods was about 0,2 - 0,3 o/oo higher after the ramp test than before ramping.

The visuell inspection showed no failures, but on some fuel rods deposits caused by surface boiling were found. The diameter of the test fuel rods was not essentially changed, but circumferential ridges up to 4/ μ m height were found, which were concentrated on the pellet-pellet interfaces. Some ridges were placed near pellet cracks.

The start-up ramps up to 530 W/cm caused larger ridges than the in-situ ramps up to 460 W/cm.

7. Next Steps

Continuation of the specified irradiation program and post-irradiation-examination of the ramped test fuel rods.

Retransfer of 10 irradiated test fuel rods from Petten to Karlstein for post-irradiation examination.

Intermediate inspection of 28 preirradiated test fuel rods in hot cells at Karlstein.

Transport of 28 preirradiated test fuel rods to Petten.

Continuation of the evaluation of results.

Preparation of a publication of the first results.

8. Relation with Other Projects

The preirradiation of 34 segmented UO_2 - and UO_2/PuO_2 -fuel rods was sponsored by the BMFT (AtI 7384 and 7704).

9. References

10. Degree of Availability

<u>Title 1 (Original language)</u> Prove di scoppio su guaina	<u>Classification</u> 11.1
<u>Title 2 (English)</u> Burst tests	<u>Country</u> ITALY <u>Sponsor</u> CNEA <u>Organisation</u> CNEA
<u>Date initiated</u> January 1973 <u>Date completed</u> 1976 <u>Last updating</u> December 1976	<u>Project Leader</u> G.P. CALI'

1. General aim

Mechanical properties of irradiated cladding under rupture conditions.

2. Particular objectives

Measurement of 1) Ultimate burst strength 2) Wall thinning at fracture
 3) Total circumferential elongation in irradiated cladding internally pressurized up to rupture conditions.

3. Experimental facilities and programme

Risö and Studsvik Laboratories

4. Project status

Burst tests have been completed within the following range of conditions:
 temperature: 280°C and room temperature; burn-up: 1.500 to 24.000 MWD/T;
 integrated fast neutron dose: $6 \pm 30 \times 10^{20}$ n/cm²; cladding diameter and type
 15.06 mm ext.dia.(VDM), 14.26 mm ext.dia (Holwerin).

5. Next steps

None

6. Relation with other projects

None

7. Reference documents

Internal reports

8. Degree of availability

Open

<p><u>Title 1 (Original language)</u> Studio della conduttanza dell'intercapedine combustibile/guaina in barre di combustibile BWR</p>	<p><u>Classification</u> 11.1</p>
<p><u>Title 2 (English)</u> Gap conductance in BWR fuel rods</p>	<p><u>Country</u> ITALY <u>Sponsor</u> CNEN <u>Organisation</u> CNEN</p>
<p><u>Date initiated</u> 1968 <u>Date completed</u> in progress <u>Last updating</u> April 1977</p>	<p><u>Project Leader</u> A. Nobili</p>

1. General Aim

The aim of the experimental program is to study the effect of the different parameters (linear heath rating, cold gap, power cycling, etc.) on the gap conductance.

2. Experimental facilities

The irradiation has been performed utilizing instrumented rigs capables to measure directly the power generated in the fuel as well as the fuel and cladding temperatures (1).
Reactor utilized: Avogadro, Siloè, Essor.

3. Project status

A first series of irradiation concerning the thermal behaviour of different fuel types has been completed (2).

4. Next steps

An experimental program is in progress in order to improve the knowledge of the effects of the axial load, power cycling and cold gap on the gap conductance.
The irradiation will be performed in Essor and Siloè.

5. Relation to other projects

Preliminary studies to the ASCOT program.

6. Reference documents

The most important documents are:

- 6.1. A. Calza-Bini et al., "Esperienze di irraggiamento di capsule Cyranum per lo studio del comportamento termico della barra combustibile", RT/ING(70)15.

<u>Title 1 (Original language)</u> Studio della conduttanza dell'intercapedine combustibile/guaina in barre di combustibile BWR.	<u>Classification</u> 11.1
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6.2. A. Calza-Bini et al., "In-pile measurement of fuel cladding conductance for pelleted and vipac Zr-2 sheathed fuel pins", Nuclear Technology, Vol. 25, Jan. '75.

6.3. A. Calza-Bini et al., "Esperienze Gioconda: studio sperimentale del comportamento termico di una barra combustibile ad UO_2 : integrale di conducibilità e conduttanza dell'intercapedine tra combustibile e guaina", CNEN.

7. Budget, personnel involved

The budget for the second part of the program is 200 ML to irradiate 8 different fuel pins. This cost does not include the neutrons utilized in ESSOR reactor.

The personnel involved is 4 men for about 2 years.

8. Additional information

The results on gap conductance are utilized to calculate the stored energy in the fuel pin at the beginning of the LOCA accident.

<u>Title 1 (Original language)</u> Comportamento meccanico e chimico di guaine di Zr ad alta temperatura	<u>Classification</u> 11.1
<u>Title 2 (English)</u> High temperature mechanical and chemical behaviour of cladding tubes.	<u>Country</u> ITALY <u>Sponsor</u> CNEN <u>Organisation</u> CNEN
<u>Date initiated</u> 1975 <u>Date completed</u> in progress <u>Last updating</u> April 1977	<u>Project Leader</u> S. OMARINI

1. General aim

Development of stress strain and oxide growth rate correlations at high temperature for Zircaloy fuel cladding.

2. Experimental facilities and programme

a) Joule effect heating facility for testing of Zr cladding tubes with different heating rate or temperature transients up to 1100°C. Inner pressure range 20-60a ta.

b) Radiant heat facility for the high temperature Zr oxidation studies.

3. Project status

The first tests on CIRENE HWR reactor and on CAORSO LWR geometries already tested. Tests completed in 1977.

4. Next steps

a) Test of cladding tubes with a variable inner pressure to simulate the differential pressure of a LOCA transient.

b) Development of internal heater for fuel rod simulation and H-Gap studies.

5. Reference documents

Internal reports.

6. Degree of availability

To a limited extent.

7. Budget, Personnel involved

8 persons.

<u>Title 1 (Original language)</u> Misura in pila della velocità di creep di una guaina di Zircalloy 2 di tipo Caorso	<u>Classification</u> 11.1
<u>Title 2 (English)</u> In-pile creep rate measurements of Zircalloy 2 cladding of Caorso fuel pin	<u>Country</u> ITALY <u>Sponsor</u> CNEN <u>Organisation</u> CNEN
<u>Date initiated</u> 1977 <u>Date completed</u> 1977 <u>Last updating</u> April 1977	<u>Project Leader</u> Nobili

1. General Aim

Measurement of creep rate of a Zr-2 cladding under irradiation.

2. Experimental facilities

The cladding sample will be placed in a CHOUCA Device in which the temperature is constant ($\pm 1^\circ\text{C}$): the sample elongation is measured continuously by means of a resonant chamber system. The irradiation will be performed at Siloè Reactor.

3. Project status

The sample fabrication is in progress; the irradiation will take place at the beginning of July.

4. The next step will be to set-up an experimental program in order to investigate the behaviour of the cladding of a fuel pin under irradiation.

5. Relation to the other projects

Connection to the program performed by NUCLITAL.

6. Reference documents

The most important reference documents are:

- 6.1. M. Masson, P. Millies, R. Wallop "Mesure en continue de deformation pendant l'irradiation en pile", CEA-CENG, Service des Piles P₁(NT) 461-103/68.

<u>Title 1 (Original language)</u> Misura in pila della velocità di creep di una guaina di Zircalloy 2 di tipo Caorso	<u>Classification</u> 11.1
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6.2 R. Warlop "Ensemble de mesure de deformation", CEA-CENG/G, Service des Piles, NT/P_i 293-05-124/70.

6.3 A. Calza-Bini et al. "Misura in pila di velocità di creep su tubi di guaina in acciaio inox in geometria veloce", CNEN/RT/ING(75)16.

7. Budget, personnel involved

The cost of the experiment will be about 50 ML for 2000^h of irradiation.

The personnel involved is one man/year.

8. Additional information

The results will be utilized to calculate the permanent deformation of the cladding under internal pressure.

PROJECT TITLE: Creep and Plastic behaviours of Zircaloy Cladding Tubes - Phase I	CSNI INDEX NUMBER: L. W. R. 11.1
SPONSORING COUNTRY: Japan	ORGANISATION: Expert Committee on Nuclear Fuel Safety of Nuclear Safety Research Association
DATE INITIATED: May 1976 DATE COMPLETED: Oct. 1977	PROJECT LEADER: Y. MISHIMA

Description

1. Objectives

To obtain the creep and plastic equations of zircaloy cladding tubes, out-of-pile experiments are conducted, and in-pile and out-of-pile data from various sources are compiled.

2. Project Status

2.1 Program Description

(a) External pressure creep experiments creep deformation of fully annealed and stress relieved zircaloy cladding tubes is measured by profilometry after exposure to external pressures at high temperatures for various time periods. Some of the specimens used contain stainless steel pellets which simulate the hourglass deformation of UO_2 pellets to examine the effect of internal supports during creep down.

(b) Plastic deformation study

The Stress-strain behaviors of the zircaloy cladding tubes under a wide range of strain rate and under cyclic loading conditions are measured.

2.2 Progress

In the phase I of the program, the experimental apparatus were installed, and some experiments under rather narrow range of parameter were conducted so far.

3. Essential Results

Results will be discussed after the completion of more experiments.

4. Projection

In the Phase-II of the program, experiments will be continued to obtain data under a wide range of parameter, and the internal pressure creep experiment will be started.

5. Report

The report will appear soon.

PROJECT TITLE: Effect of pellet geometry on the thermal deformation of UO ₂ pellet	CSNI INDEX NUMBER: L. W. R. 11.1
SPONSERING COUNTRY Japan	ORGANIZATION: Expert Committee on Nuclear Fuel Safety of Nuclear Safety Research Association
DATE INITIATED: May 1975 DATA COMPLETED: Oct. 1976	PROJECT LEADER Y. MISHIMA

Description

1. Objectives

The objectives of the program are to examine the thermal deformation and cracking of UO₂ pellets of various shapes and sizes, and to improve our understanding of the pellet-clad mechanical interaction in LWR fuel.

2. Project Status

2.1 Program Description

Four fuel pins, contained in NaK filled capsules and irradiated in JMTR, were examined in the hot cell of JMTR. The parameters of fuel pins are the following. The cladding is of fully annealed Zircaloy of 0.3mm thickness. An extremely narrow gap was selected to cause fuel-clad interaction as early as possible. Thus, the geometry of pellets during irradiation could be retained and measured in post-irradiation profilometry. The pellets consist of two types: ordinary cylindrical pellets of various length to diameter ratio, L/D, and divided pellets

which were cut into eight octants of cylindrical shape before inserting into the cladding.

2.2 Progress

The post-irradiation examinations in the hot cell were completed, and results were analyzed. The examinations include visual examinations, profilometry, length measurement, and metallographic examination of crack pattern and restructuring of pellets. The analysis includes the estimation of temperature and stresses in pellets during irradiation.

3. Essential Results

The crack pattern in pellets appears to be explained from stresses in the pellets calculated by the thermo-elastic finite element method. The size of segments in the cracked pellets appears to have a minimum size which is closely near to the short devided pellets ($L/D=0.2$). The measured ridges on the profilometry recordings are attributed not only to the hourglass shaping of the pellets but also to the zigzagging of pellet stack.

4. Projection

The Program completed.

5. Reports

Final Report on "Effect of Pellet Geometry on the Thermal Deformation of UO_2 Pellet. May 1976, Nuclear Safety Research Association.

PROJECT TITLE : Study on the fuel behavior under irradiation	CSNI INDEX NUMBER : 11.1
SPONSORING COUNTRY : Japan	ORGANISATION : JAERI, Tokai
DATE INITIATED : DATE COMPLETED : Progressing (March 1977)	PROJECT LEADER : A. Morishima

Description :

1. Objectives

This program aims at contributing to the better understanding of fuel failure behavior and getting data for the verification of fuel behavior prediction codes.

2. Project Status

2.1 Program Description

(1) Fuel irradiation in HBWR, Halden Project

A total of eight instrumented fuel assemblies were irradiated in HBWR by JAERI and seven others were irradiated under joint research programs between JAERI and the corporations in Japan. The research subjects extend over pellet-clad mechanical interaction (PCMI), fuel center temperature, densification and behavior of moisture and hydrogen in fuel and cladding, respectively.

(2) Fuel irradiation in R2, Inter Ramp Project

Information will be published by Inter Ramp Project.

(3) Fuel irradiation in JMTR, JAERI

Two kinds of irradiation tests of fuel rods for a BWR are carried out in JMTR; one is to study densification behavior, and the other to investigate PCMI (especially relocation) behavior.

2.2 Progress

(1) HBWR irradiation

Results from JAERI's irradiation experiments IFA-106, 107, 149,

(Continued from the preceding page.)

150, 209, 225 were reported in the Halden Project Report series. The irradiation tests on IFA-411 and -424 are conducted and irradiation of IFA-410, 421 and 423 were terminated. Furthermore, for the study of elongation change, irradiation program due to 3 rods diametral rigs is in progress.

(2) R2 irradiation

Progress of the Inter Ramp Project will be published by the Project.

(3) JMTR irradiation

Irradiations were successfully finished on four fuel rods (73F-6A, -8A, -9A and -11A). Afterwards, for both 73F-8A and -11A post irradiation tests on density measurement and metallography in pre-and post-irradiations have been conducted. While, irradiations will be carried out in JMTR on three fuel rods (76F-1A, -2A and -3A) for the study of relocation behavior.

3. Essential Results

From metallographic data, both grain size and pore size distribution were measured.

Analysis of PCI behavior was conducted from IFA-411 and 411 data. Influence of several essential factors on PCI was studied against fuel rod parameters and operational conditions.

4. Projection

All of the post irradiation tests in HBWR and JMTR are continued. Three rods diametral rigs will be irradiated in 1977. Also, three capsules for relocation study will be irradiated in 1977.

PROJECT TITLE : The Study of Fuel Cladding Behavior	CSNI INDEX NUMBER : 11.1
SPONSORING COUNTRY : Japan	ORGANISATION : JAERI, Tokai
DATE INITIATED : May 1974 DATE COMPLETED : Progressing (March 1977)	PROJECT LEADER : S. Kawasaki

Description :

1. Objectives

In order to confirm the regulatory guides for fuel safety under a LOCA for LWRs, it is required to estimate the behaviour of the fuel cladding, such as zircaloy-steam reaction rate, ballooning, wall thinning and embrittlement.

2. Project Status

2.1 Program Description

- 1) To evaluate the oxidation for cladding, the reaction between zircaloy-4 and steam, and then the related embrittlement for cladding are examined.
- 2) To estimate the ballooning, perforation and inner surface oxidation of cladding, sham fuel rods are inductively or externally heated in steam atmosphere under a simulated LOCA condition.
- 3) To simulate inside oxidation, zircaloy-4 tubes are oxidized in stagnant steam and flowing hydrogen/steam mixed gases, and the related embrittlement of tubes is studied.
- 4) To develop sufficient information on the ductilities of zircaloy oxidized in a LOCA condition, zircaloy-4 tubes are oxidized in oxygen atmosphere to produce various oxygen concentration (gradients). The effects of oxygen on embrittlement for cladding are examined.
- 5) To evaluate the interference with emergency cooling effectiveness, fuel assemblies are heated in steam under a simulated LOCA condition.

(Continued from the preceding page.)

2.2 Progress

- 1) The reaction rate was measured between 900°C and 1330°C. The influence of the presence of pre-oxidized film under a normal reactor operation was examined too. The amount of oxidation (weight gain), the thicknesses of the oxide and the stabilized α -layer were measured. The ductility of oxidized specimens was estimated by ring compression test at 100°C.
- 2) Six kinds of pre-pressurized sham fuel rods, which were filled with Al_2O_3 pellets and Ar gas to simulate fuel pellet volume and to provide the volume of gas present in an actual fuel rod, were heated in steam or vacuum atmosphere to distinguish the factors influencing the deformation. The factors are included with initial internal pressure at room temperature, heating method, heating rate and presence of pre-oxidized film.
- 3) Zircaloy tubes were oxidized in stagnant steam and flowing hydrogen/steam mixed gas at temperatures between 950 and 1200°C. The ductility of the reacted specimens were measured by the ring compression test at 100°C.
- 4) Zircaloy-4 tubes were prepared up to 11700ppm (chemically analyzed) in oxygen atmosphere at 800°C and annealed in vacuum at 850°C to produce various oxygen concentration (gradients). Ring compression tests at temperatures between 100 and 700°C and metallographic observations were performed.
- 5) An assembly was designed, which consists of 7x7 rods, each of which contains a center line heater, Al_2O_3 pellets and pressurized Ar gas. The dimensions of the assembly are those of PWR except axial length. The burst experiment of the assembly in steam atmosphere started.

3. Essential Results

- 1) Above 1000°C, the amount of oxidation is represented as the following equation;

weight gain (W)

$$W^2 = k_w \times t \quad k_w = 0.468 \exp(-40710/RT) \text{ (g/cm}^2\text{)}^2\text{/sec,}$$

thickness of oxide

$$\delta^2 = k_\delta \times t \quad k_\delta = 0.0215 \exp(-35860/RT) \text{ (cm}^2\text{/sec),}$$

(Continued from the preceding page.)

thickness of oxide and α -layer

$$\xi^2 = k_{\xi} \times t \quad k_{\xi} = 0.396 \exp(-39420/RT) \text{ (cm}^2\text{/sec).}$$

within 60 minutes, the reaction showed a logarithmic type and the reaction rates were retarded by the presence of pre-oxide film produced under a normal reactor operation. The ductility of oxidized specimens decreased with increasing weight gain, but the deformation behaviour depends on the reaction temperature. In comparing with the equal weight gain, the ductility loss of the specimens oxidized at low temperature was less than that at high temperature.

- 2) The maximum circumferential strain of sham fuel rods was related to the burst temperature, and influence of the pre-oxidized film produced under a normal reactor operation and atmosphere become apparent. The average wall thickness of cladding varying with the deformation depends on the heating method and atmosphere. The average wall thickness of fuel rods by induction heating was smaller oxide layer on the inner surface of the tube was thicker than that of the outer when amount of the supplied steam to the inner surface of the tube wall was less and the rupture opening size was small. The thicker coarse inner oxide layer consists of monoclinic and tetragonal zirconia, while the thinner outer oxide layer is of monoclinic zirconia alone. The longitudinal inner surface oxidation was observed within 76mm from the center of rupture opening.
- 3) Though the weight gain of the specimen reacted with stagnant steam is less than that reacted with flowing steam for the equal temperature and time, the ductility loss is upside down. The weight gain and the ductility of the specimen oxidized in flowing mixed gas containing hydrogen of less than 20% (by volume) are almost the same as those oxidized in pure flowing steam. At about 20% of hydrogen, the value of the weight gain is scattered, and the weight gain and the ductility are decreased in the mixed gas containing hydrogen of higher than 20%. Considerable amount of hydrogen is absorbed in the specimens with less ductility in both experiments.

(Continued from the preceding page.)

- 4) The growth of the crystal grains in the oxygen-enriched specimens corresponds with the penetration of oxygen diffusion. A ductility minimum appeared for the specimens containing 3000ppm of oxygen at 100°C. It occurred for the specimens containing over 5300ppm of oxygen at 300°C and 11700ppm at 500°C.

4. Projection

The simulation experiment for inner surface oxidation will finish by May 1977. Assembly test will continue till 1980.

5. Reports

- 1) S. Kawasaki, T. Furuta and M. Hashimoto : "Zircaloy-Steam Reaction under a Simulated Loss-of-Coolant Accident", JAERI-M-6181 (July 1975) (in Japanese).
- 2) T. Furuta, M. Hashimoto and S. Kawasaki : "Effect of Oxygen Content and Distribution on Embrittlement for the Zircaloy Cladding", JAERI-M-6182 (July 1975) (in Japanese)
- 3) T. Furuta, M. Hashimoto, T. Otomo, S. Kawasaki and K. Homma : "Deformation and Inner Oxidation of the Fuel Rod in a Loss-of-Coolant Accident Condition", JAERI-M-6339 (Dec. 1975) (in Japanese)
- 4) T. Furuta, S. Kawasaki, M. Hashimoto and T. Otomo : "Factors Influencing Deformation and Wall Thickness of Fuel Rods under a Loss-of-Coolant Accident", JAERI-M-6542 (May 1976) (in Japanese)
- 5) K. Homma, M. Hashimoto, T. Furuta, T. Otomo and S. Kawasaki : "Behaviour of Inner Surface Oxidation of Zircaloy Cladding during a Loss-of-Coolant Accident", JAERI-M-6602 (July 1976) (in Japanese)
- 6) T. Furuta, S. Kawasaki, M. Hashimoto and T. Otomo : "Zircaloy-Steam Reaction and Related Embrittlement for Zircaloy Cladding under a simulated Loss-of-Coolant Accident", JAERI-M-6601 (July 1976) (in Japanese) (Revised edition of JAERI-M-6181)
- 7) S. Kawasaki, T. Furuta, K. Homma : "Inner Surface Oxidation of a Fuel cladding for LWR under a LOCA", CSNI Specialist Meeting at Spatind, (Sept. 1976)

(Continued from the preceding page.)

- 8) M. Suzuki, S. Kawasaki, T. Furuta : "Zircaloy-Steam Reaction and Embrittlement of the Oxidized Zircaloy Tube under Postulated Loss of Coolant Accident Conditions. (Oxidation Kinetics and Embrittlement of Zircaloy above 1200°C)", JAERI-M-6879 (Jan. 1977)

N.V. KEMA

Arnhem, 25 februari 1976

Classification 11.1	
<u>Title 1</u> Splijtstofverlenging tijdens bedrijf	<u>Country</u> The Netherlands <u>Organization</u> KEMA
<u>Title 2</u> Fuel elongation during power rating	<u>Scientists</u> W.R. van Engen J. Hoekstra W. Slegers

1. General aim

Pellet-clad interactions.

2. Particular objectives

To detect possible pellet-clad interaction situations in the core of a power reactor by means of

- a) absolute elongation of the fuel rods or whole assemblies
- b) change in time constant for heat transfer of the fission heat to the coolant

3. Experimental facilities

Dedewaard nuclear power plant.

4. Project status

Still in progress.

5. Next steps

To come to an additional method for incore measurements to have better knowledge over actual core conditions.

6. Relations with other projects

Measuring technique is based on the Halden incore instrumentation experience with differential transformers.

7. Reference documents

None.

8. Degree of availability

Through the organization KEMA.

PROJECT TITLE : Deformation and rupture of zircaloy cladding owing to LOCA	CSNI INDEX NUMBER : 11.1
SPONSORING COUNTRY : Sweden	ORGANISATION : AB Atomenergi
DATE INITIATED : 1976-12-29	PROJECT LEADER :
DATE COMPLETED : 1978-02-01	Peeter Tarkpea

Description :

The objective of the project is to investigate and evaluate the creep characteristics of zircaloy cladding material in oxidizing atmosphere and to develop and verify models for deformation and rupture of zircaloy cladding in oxidizing atmosphere under LOCA conditions.

PROJECT TITLE : Fission Product volatility in intact fuel rods	CSNI INDEX NUMBER : 11.1
SPONSORING COUNTRY : Sweden	ORGANISATION : AB Atomenergi
DATE INITIATED : 1974-07-01 DATE COMPLETED : 1977-06-30	PROJECT LEADER : Ronald S Forsyth

Description :

Great attention is paid during consideration of LOCA incidents to the increase in internal fuel rods gas pressures with risk of subsequent ballooning, rupture and propagation. However, little is known of the possible contribution to internal pressures of volatile fission products other than Kr and Xe.

In this project, intact irradiated fuel rods are heated in a furnace to various temperatures and the possible release of volatile fission products such as Te, I and Cs to the gas plenum is followed by means of gamma spectrometry. Study of the short-lived nuclides (such as Te and I) is made possible by re-irradiating the rods for a few days in the R2 reactor and performing the experiments shortly after removal from the reactor. Initially, fuel and clad temperatures have been restricted to 800°C and lower and under these circumstances no traces of volatile fission products have been detected in the fuel rod gas plenum. In a second stage, temperatures up to 1 200°C will be examined.

OECD NUCLEAR ENERGY AGENCY - 38, Boulevard Suchet, 75016 PARIS
Telex : 630668 AEN NEA - Tel : 524 82-00

14.560

PROJECT TITLE : Fuel cladding behaviour under Loss-of-coolant accident	CSNI INDEX NUMBER : 11.1
SPONSORING COUNTRY : Sweden	ORGANISATION : AB Atomenergi
DATE INITIATED : 1974 DATE COMPLETED : 1977	PROJECT LEADER : Peeter Tarkpea

Description :

The objective is to study the behaviour of zircaloy-2 fuel pin cladding under simulated BWR LOCA conditions. The studies are concentrated on a hypothetical LOCA obtained in a BWR with external circulation pumps.

As the LOCA temperature transient in a BWR with external circulation pumps is relatively slow the uptake of oxygen will reduce the creep rate. Thus the oxidation, the change in mechanical properties and the types of deformation of unirradiated zircaloy-2 cladding is studied. A part of these studies have been done but more information is needed about the creep behaviour of zircaloy-2 under temperature transient conditions. These data will be used to verify actual fuel cladding behaviour models, which are under study and to complete input data to codes, which are of interest.

Single pin tests were performed in steam atmosphere under transient conditions. The temperature transient put on the test specimens was a computer calculated BWR LOCA temperature transient of the hottest pins in a fuel bundle. Resistance heating and high frequency heating was used and the pins were pressurized. The internal pressure was varied between 0.53 and 0.88 MPa at 1 204°C and the pressure was either constant or increasing with the temperature transient. This increasing pressure was obtained in prepressurized tight-welded specimens.

The following observations and conclusions has been made.

Uniform expansion along the constant temperature length was 10 - 20 % before the specimens ruptured. Almost no or very little localized deformation was observed at the rupture area except for specimens run at constant internal pressures at and above 0.78 MPa. These specimens deformed locally and the local deformation was about 70 %.

Variations were observed in the oxidation parameter F_w , which is the ratio of transformed β -phase remaining to as-oxidized wall thickness. The oxidation parameter F_w was determined to values between 0.7 and 0.8 with the used temperature transient.

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14.560

PROJECT TITLE : FUEL CLAD DISTORTION UNDER ACCIDENT CONDITIONS	CSNI INDEX NUMBER : 11.1
SPONSORING COUNTRY : UNITED KINGDOM	ORGANISATION : RFL SPRINGFIELDS
DATE INITIATED : 1971 DATE COMPLETED :	PROJECT LEADER : DR K M ROSE

Description :

1. General Aim

To secure acceptable core conditions following a loss-of-pressure accident (LWR) and SGHWR.

2. Particular Objectives

To discover whether internal pressure across Zircaloy clad pins causes deformation which obstructs further cooling.

3. Experimental Facilities and Programme

Fuel bundles up to 2 m length are heated in a furnace, through defined temperature and pressure suequences. The bundles are dismantled and distortions measured.

4. Project Status

Many such tests have been performed. Extended blockage can be produced particularly by low temperature (850°C) suequences.

A 4 m directly heated rig is under construction.

Reference Documents

SNI Report SNI/1/14 Oct 1973

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31.719

PROJECT TITLE : HIGH TEMPERATURE CREEP OF ZIRCALOY	CSNI INDEX NUMBER : 11.1
SPONSORING COUNTRY : UNITED KINGDOM	ORGANISATION : RFL SPRINGFIELDS
DATE INITIATED : 1971 DATE COMPLETED :	PROJECT LEADER : DR K M ROSE

Description :

1. General Aim

To secure acceptable core conditions following a loss of pressure accident (LWR) and SGHWR.

2. Particular Objectives

To define limits of temperature and stress for which the Zircaloy can will not creep significantly.

3. Experimental Facilities and Programme

Short lengths of fuel can are taken through a defined sequence of temperature and internal pressure. The deformation of the can is measured throughout the transient.

4. Project Status

Many tests have been performed with Zircaloy cladding, originally with inert gas atmosphere, currently with a steam atmosphere as this significantly affects the results. The test is specific to the reactor temperature transient, so the tests are repeated from time to time as changes are made to the reactor design. Irradiated cladding has been examined.

Reference Document

SNI Report SNI/1/14 Oct 1973

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Telex : 630668 AEN NEA - Tel : 524 82-00

31.719

PROJECT TITLE : Mechanical Properties of Zircaloy Containing Oxygen	INDEX NUMBER : 11.1
SPONSORING COUNTRY : USA (NRC)	ORGANISATION : Argonne National Lab.
DATE INITIATED : July 1974 DATE COMPLETED : Continuing	PROJECT LEADER : T. F. Kassner

Description :

1. General aim: Development of embrittlement criteria for Zircaloy fuel sheaths during LOCA.
2. Particular objectives: Determination of stress-strain and impact properties of Zircaloy as functions of temperature, oxygen content, oxygen gradient, and microstructure in both uniaxial and biaxial stress.
3. Experimental facilities and programs:
 - a. Facilities: High vacuum and inert atmosphere, high temperature specimen homogenization furnaces. Special high vacuum and inert atmosphere, high temperature tensile testing apparatus for temperatures up to 1400°C.
 - b. Programs:
 1. Preparation and characterization of test specimens
 2. Mechanical properties of Zircaloy-oxygen alloys
 3. Development of a failure criteria for oxygen embrittled Zircaloy fuel cladding.
4. Project status:
 - a. Progress to date: Methods have been developed for producing satisfactory specimens and characterizing them; testing equipment has been obtained and development of test methods is underway.
 - b. Essential results: Superplasticity has been observed in Zircaloy at temperatures from 800 to 1300°C (test range) at strain rates from 10^{-4} to $> 10^{-2}$ sec⁻¹ (testing range), by both uniaxial and tube burst tests. End restraint and increased oxygen content both

decrease maximum rupture strain in burst tests and increase temperature for the lower superplasticity peak (3 peaks observed).

4. Next steps: Continue research program.
6. Relation with other projects: Related to Multirod Burst Test Program (ORNL), Zirconium Metal-Water Oxidation Kinetics (ORNL), PBF Experimental Engineering (ANC).
7. Reference documents: ANL-75-28 Quarterly Progress Report on Light Water Reactor Safety Research for Period Ending March 31, 1975.

Light Water Reactor Safety Research Programs: Quarterly Progress Report: Part III. T. F. Kassner, et al.

Jan-March, 1975, ANL-75-28

April-June, 1975, ANL-75-58

July-September, 1975, ANL-75-72

Oct-Dec., 1975, ANL-76-15

January-March 1976 ANL-76-49

April-June 1976 ANL-76-87

July-September 1976 ANL-76-121

October-December 1976 ANL-77-10

8. Degree of availability: NTIS

PROJECT TITLE : Multirod Burst Tests	INDEX NUMBER : 11.1
SPONSORING COUNTRY : USA (NRC)	ORGANISATION : Oak Ridge National Laboratory
DATE INITIATED : July 1974 DATE COMPLETED : Continuing	PROJECT LEADER : R. H. Chapman

Description :

1. General aim: Examine rod-to-rod interaction and develop correlation for predicting burst behavior in large clusters (simulated fuel assemblies) during transient temperature burst testing in low pressure steam.
2. Particular objectives:
 - a. determine minimum number of rods in a cluster which must be used to allow prediction of behavior of clusters containing many rods.
 - b. determine correlation of single rod behavior of multirod behavior if possible.
 - c. determine influence of one rod on its neighbors during multirod burst testing, with ballooning.
 - d. determine flow-blockage produced in clusters of burst rods as function of burst geometries and elongations.
 - e. determine rupture temperature-pressure relationships for rods having small gas volumes and small diametral gaps.
3. Experimental facilities and programs:
 - a. Facilities: Low-voltage megawatt DC power supplies, flow test loops, multirod burst test loop.
 - b. Programs:
 1. single rod tests
 2. 4 x 4 bundle tests
 3. 8 x 8 bundle tests
 4. 11 x 11 bundle tests

4. Project status:
 - a. Progress to date: Single rod test facility designed and constructed, master lot of test Zircaloy tubing obtained, prototype heaters being tested, multirod test facility designed and construction started. Single rod tests completed, data being analyzed.
 - b. Essential results: rupture temperature-pressure data agree with prior reports, rupture strain less than 40% maximum of any temperature. No data obtained yet on multirod tests.
5. Next steps: Conduct planned research program.
6. Relation with other projects: Zirconium Metal-Water Oxidation Kinetics (ORNL), Mechanical Properties of Zircaloy Containing Oxygen (ANL), Strength and Ductility of Irradiated Zircaloy (BMI), PBF Experimental Engineering (ANC), Fuel Model Development (ANC).
7. Reference documents: ORNL-TM-----Quarterly Progress Report on Reactor Safety Programs Sponsored by the Division of Reactor Safety Research, Vol. 1. Light Water Reactor Safety (report series).
ORNL-TM-----Quarterly Progress Report on Reactor Safety Programs Sponsored by the Division of Reactor Safety Research, Vol. 1. Light Water Reactor Safety (report series).
 - 7.1 Quarterly Progress Report on Reactor Safety Programs Sponsored by the Division of Reactor Safety Research for July-September 1974, ORNL/TM-4729, Vol. I, pp. 70-72.
 - 7.2 Quarterly Progress Report on Reactor Safety Programs Sponsored by the NRC Division of Reactor Safety Research for October-December 1974, ORNL/TM-4805, Vol. I, pp. 102-10.
 - 7.3 Quarterly Progress Report on Reactor Safety Programs Sponsored by the NRC Division of Reactor Safety Research for January-March 1975, ORNL/TM-4914, Vol. I, pp. 78-104.
 - 7.4 Quarterly Progress Report on Reactor Safety Programs Sponsored by the NRC Division of Reactor Safety Research for April-June 1975, ORNL/TM-5021, Vol. I, pp. 76-98.
 - 7.5 R. H. Chapman, Multirod Burst Test Program Quarterly Progress Report for July-September 1975, ORNL/TM-5154.

PROJECT TITLE: PBF Experimental Engineering	INDEX NUMBER: 11.1
SPONSORING COUNTRY: USA (NRC)	ORGANIZATION: EG&G Idaho Inc.
DATE INITIATED: 1964 DATE COMPLETED: Continuing	PROJECT LEADER: J. Zane/J. Crocker

Description:

1. General aim: To provide information on the behavior of reactor fuels under normal, off-normal and accident conditions.
2. Particular objectives: Testing of single fuel rods and fuel rod clusters under:
 - (a) Power-Cooling Mismatch (PCM) conditions
 - (b) Loss of Coolant (LOCA) conditions
 - (c) Reactivity Initiated Accident (RIA) conditions
 - (d) Inlet Flow Blockage conditions
3. Experimental facilities and programs:

The Power Burst Facility is a light water cooled and moderated reactor contained in an open top steel vessel. The present reactor core is designed for both steady state operation (to 30 MW) and pulsed mode operation (to 1500 MW sec). The PBF currently operates on a three shift basis, but four shift operation during the next few years is probable. At present reactor tests are scheduled at 7 day to 30 day intervals, with 7 to 10 tests scheduled per 8-month operating year. The program for the four year period, June 1975 - June 1979 encompasses tests in each of the following areas: (a) Power-cooling mismatch (PCM), 10 reactor tests (FY76, early FY77), (b) Irradiation effects, 6 reactor tests (FY76-77), (c) Loss of Coolant Accident (LOCA) 11 reactor tests (late FY77, 78, 79), (d) Inlet Flow Blockage, 1 reactor test (late FY78), (e) Reactivity Initiated Accident (RIA), 7 to 8 reactor tests (FY78-79), (f) Gap Conductance, 7 reactor tests (FY76-79) plus gap conductance measurements on most other PBF tests.

4. Project status: Program description documents (PDD) have been prepared for each of the test programs. Core lead rod tests have been performed. Single rod (4 at a time) tests in the PCM, irradiation effects and Gap Conductance test series have been performed. The initial tests in the LOCA and RIA series are scheduled for performance in late FY77 and early FY78, respectively. Advance preparations for current and all future test series are under way. (By 7/77, 54 fuel rods tested in 18 reactor tests.)
5. Next steps: Efforts will be made to test small clusters of irradiated and unirradiated fuel rods under planned test conditions.
6. Relation to other projects: This is part of an overall program sponsored by NRC in support of in-reactor study of properties of fuel rod materials and assemblies under normal, off-normal and accident conditions and in support of fuel element behavior model development and evaluation programs.
7. Reference documents:
 - a. System Safety Research Division, Aerojet Nuclear Company, "Fuel Behavior Requirements Project Description for Fuel Behavior Branch, AEC (NRC)/RSR," Appendices A through H, November 1973 through January 1975. TFBP-TR-100.
 - b. Quarterly Technical Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, January-March 1975, ANCR-1254 (September 1975).
 - c. Quarterly Technical Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, April-June 1975, ANCR-1254 (November 1975).
 - d. Quarterly Technical Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, July-September 1975, ANCR-1296 (February 1976).
 - e. Quarterly Technical Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, October-December 1975, ANCR-1301 (May 1976).
 - f. Quarterly Technical Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, January-March 1976, ANCR-1315 (June 1976).
 - g. Quarterly Technical Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, April-June 1976, TREE-NUREG-1004 (October 1976).
 - h. Quarterly Technical Report on Water Reactor Safety Programs, Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, July-September 1976, TREE-NUREG-1017 (January 1977).

- i. Quarterly Technical Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, October-December 1976, TREE-NUREG-1070 (April 1977).
 - j. R. K. McCardell et al, "TFBP Test Results Report", 'PCM Series Power Calibration Test', ANCR-1248 (July 1975).
 - k. G. W. Cawood et al, "TFBP Test Results Report", "PCM Series Test PCM-2A," ANCR-NUREG-1347 (September 1976).
 - l. S. L. Seiffert & G. R. Smolick, "Post-Irradiation Examination Results for PCM Series Test PCM-2A," TREE-NUREG-1029 (February 1977).
 - m. Z. R. Martinson & R. K. McCardell, "TFBP Test Results Report, "PCM Series Test PCM-2," TREE-NUREG-1038 (February 1977).
 - n. S. L. Seiffert, "Post-Irradiation Examination Results for PCM Series Test PCM-2," TREE-NUREG-1069 (March 1977).
 - o. G. B. Peeler et al, "TFBP Test Results Report, PCM Series Test PCM-4," TFBP-TR-190 (June 1977).
 - p. W. J. Quapp et al, "TFBP Test Results Report, Irradiation Effects Scoping Test 1," TFBP-TR-110 (January 1976).
 - q. W. J. Quapp et al, "Post-Irradiation Examination Results for the IES Series Test IES-1," ANCR-NUREG-1336 (September 1976).
 - r. W. J. Quapp et al, "TFBP Test Results Report, Irradiation Effects Scoping Test 2," TFBP-TR-137 (September 1976).
 - s. A. S. Mehner, "Post-Irradiation Examination Results for IE Series Test IE-1," TREE-NUREG-1022 (January 1977).
 - t. W. J. Quapp et al, "TFBP Test Results Report, Irradiation Effects Test IE-1," TREE-NUREG-1046 (February 1977).
 - u. R. W. Garner et al, "TFBP Test Results Report, Gap Conductance Test GC 1-3, and Summary of Piggy-Back Tests," TFBP-TR-186 (March 1977).
 - v. M. E. Yancey, E. H. Porter, H. R. Hansen, Inspection Systems for Zircaloy Clad Fuel Rods, ANCR-1259 (October 1975).
 - w. W. C. Francis, M. R. Martin, W. J. Quapp, G. W. Gibson, Nondestructive Examination of Irradiated Fuel Rods by Pulsed Eddy Current Techniques, ANCR-1282 (February 1976).
8. Availability: NTIS.

PROJECT TITLE : Strength and Ductility of Irradiated Zircaloy Cladding	INDEX NUMBER : 11.1
SPONSORING COUNTRY : USA (NRC)	ORGANISATION : Battelle Memorial Institute
DATE INITIATED : July 1974 DATE COMPLETED : Continuing	PROJECT LEADER : A. A. Bauer

Description :

1. General aim: Development of a correlation relating the mechanical properties of irradiated Zircaloy fuel sheaths to transient conditions encountered in postulated accident conditions.
2. Particular objectives: determine stress-strain behavior of irradiated Zircaloy fuel sheaths at temperatures from 25-1000°C by tensile, bend, impact, expanding mandrel, and burst tests.
3. Experimental facilities and programs:
 - a. Facilities: testing and metallographic hot cells with associated testing equipment.
 - b. Programs:
 1. Characterization of spent fuel rods, preparation of test specimens.
 2. Annealing of irradiation damage-isothermal and transient heating by hardness, tensile, and bend tests.
 3. Transient burst tests.
 4. Development of correlation and analysis.
4. Project status:
 - a. Two lots of spent fuel rods have been obtained. Master lot has been received. Test specimens have been prepared, and tests are underway.

b. Annealing kinetics of irradiation damage in Zircaloy vary with the test method used for evaluation. Data determined by measurement of hardness changes do not predict data measured by tensile and yield strengths which, in turn, do not predict data measured by uniform and total elongations in uniaxial tension. Ductilities are recovered less quickly during isothermal anneals than are the yield and ultimate strengths, and on transient heating, an "age-hardening" type of behavior is observed in a significant loss of ductility at temperature from 600 to 900°C occurs at the same time that the yield and ultimate strengths decrease to fully annealed values. Isothermal burst tests confirm the "aging" behavior. Ovalization and average creepdown vary from point to point along the axis of the fuel rod, as does the orientation of the major axis of the ovalization. Generally, the ovalization is greatest and the average creepdown is the least at the top and bottom of the fuel rod, while the reverse is true in the central region.

5. Next steps: Continuation of study with master lot of material.

6. Relation to other projects: Related to Multirod Burst Test (ORNL), Mechanical Properties of Zircaloy Containing Oxygen (ANL), PBF Experimental Engineering (ANC), Zircaloy Creepdown and Collapse (ORNL), Fuel Model Development (ANC).

7. Reference documents: Fuel Behavior Branch Program Plan, July 1974.

Progress on Evaluating Strength and Ductility of Irradiated Zircaloy, A. A. Bauer, et al

April-June 1975,	BMI-1935
July-September 1975,	BMI-1938
October-December 1975,	BMI-1942
January-March 1976	BMI-NUREG-1948
April-June 1976	BMI-NUREG-1956
July-September 1976	BMI-NUREG-1961
October-December 1976	BMI-NUREG-1967

8. Degree of availability: NTIS

PROJECT TITLE : Zirconium Metal-Water Oxidation Kinetics	INDEX NUMBER : 11.1
SPONSORING COUNTRY : USA (NRC)	ORGANISATION : Oak Ridge National Lab.
DATE INITIATED : 1974 DATE COMPLETED : Continuing	PROJECT LEADER : J. V. Cathcart

Description :

1. General aim: determine a reliable set of quantitative data on the reaction kinetics of Zircaloy with steam for use in evaluating the conservatism of the present ECCS criteria.
2. Particular objectives:
 - a. determine accuracies of temperature measurement required and methods for achieving them.
 - b. develop computer code for modeling both heat and mass flow during oxidation of Zircaloy by steam during transient heating.
 - c. determine diffusion coefficients of oxygen in beta Zircaloy.
 - d. determine rate of reaction of Zircaloy with steam at temperatures from 900 to 1500°C.
 - e. develop correlation of isothermal rate data and diffusivity measurements with results obtained during transient temperature oxidation.
3. Experimental facilities and programs:
 - a. Oxidation apparatuses for isothermal and transient temperature oxidation measurements.
 - b. Programs - see 2. Particular Objectives.
4. Project status:
 - a. Progress to date: All isothermal and transient oxidation rate data have been determined and reported; diffusion rate measurements have been made by two methods and reported; high pressure oxidation rate measurements have been made and are being analyzed. Two computer programs, SIMTRAN I and BILD-5, have been written and debugged.

b. Essential results: The isothermal oxidation rate measurements agree with some of the prior literature (Hobson), agree very closely with the (JAERI) experimental data points reported by Kawasaki/and by Leistikow (GfK-PNS), are considerably less than Baker-Just at all temperatures (max 1500 °C), and are less than Baker-Just equation. ~~xxxx~~ disagrees with the data reported by Biederman, et al (WPI)/The data have been statistically evaluated. Oxidation during simulated LOCAs have isothermal oxidation rate data in been used to verify the use of/SIMTRAN-I and BILD-5 ~~with the existing data~~ ~~xxxxxxx~~ for predicting oxidation during transient temperature conditions. No effects of pressures ~~xxxx~~ observed up to 500 seconds at 900 or 1000 °C or at 500 psia, but data at higher pressures ~~xxx~~/longer times show too much scatter to permit conclusions to be drawn at this time.

5. Next steps: complete final reports, program terminating.
6. Relation with other projects: Related to Multirod Burst Test (ORNL), Mechanical Properties of Zircaloy Containing Oxygen (ANL), PBF Experimental Engineering (ANC), and Fuel Model Development (ANC).
7. Reference documents: Quarterly Progress Report on Reactor Safety Programs Sponsored by the Division of Reactor Safety Research, Vol. 1, Light Water Reactor Safety, ORNL-TM-----, Vol. 1 (several reports, report series).

Quarterly Progress Report on the Zirconium Metal-Water Oxidation Kinetics Program Sponsored by the NRC Division of Reactor Safety Research:
J. V. Cathcart:

July-September 1975, ORNL-TM-5148

Oct.-December 1975, ORNL-TM-5248

Jan.-March 1976, ORNL/NUREG/TM-17

April-June 1976, ORNL/NUREG/TM-41

July-September 1976, ORNL/NUREG/TM-62

October-December 1976, ORNL/NUREG/TM-87

January-March 1977, ORNL/NUREG/TM-110

Zirconium Metal-Water Oxidation Kinetics. I. Thermometry, J. V. Cathcart, et al, ORNL-5102.

Zirconium Metal-Water Oxidation Kinetics . II. Oxygen-18 Diffusion in Beta Zircaloy, R.A.Perkins, ORNL/NUREG/TM- 19, July 1976

Zirconium Metal-Water Oxidation Kinetics. III. Oxygen Diffusion in Oxide and Alpha Zircaloy Phases, R.E.Pawel, ORNL/NUREG-5, October 1976

8. Degree of availability: NTIS

PROJECT TITLE : Zircaloy Fuel Cladding Collapse Studies	INDEX NUMBER : 11.1
SPONSORING COUNTRY : USA (NRC)	ORGANISATION : Oak Ridge National Lab.
DATE INITIATED : July 1974 DATE COMPLETED : Continuing	PROJECT LEADER : D. O. Hobson

Description :

1. General aim: Develop a valid data base for computer codes to be used for predicting collapse and creepdown of fuel cladding under external pressure during service in-pile.
2. Particular objectives:
 - a. determine ex-pile collapse and creepdown behavior of Zircaloy-4 tubing under external pressure as function of gap dimensions, pressurization rate, pressure differential at several temperatures.
 - b. determine in-pile collapse and creepdown behavior of Zircaloy-4 tubing under external pressure as function of gap dimensions, pressure differential, time, and flux at several temperatures.
3. Experimental facilities and programs:
 - a. Facilities: Ex-pile and in-pile high pressure autoclaves, new instrumentation for measuring clad surface displacement at many points, ORR test facility.
 - b. Programs:
 1. Ex-pile collapse behavior of tubing under external pressure as function of diametral gap, pellet-to-pellet gap, pressure differential, pressurization rate.
 2. Ex-pile creepdown of tubing under external pressure as function of diametral gap, pellet-to-pellet gap, pressure differential, time, at several temperatures.
 3. Same studies conducted in-pile to determine effect of flux.
4. Project status:
 - a. Progress to date: Instrumentation has been developed for determining displacement of cladding surface under external pressure. Ex-pile

collapse tests have been completed. Data are being analyzed.
Design of in-pile test equipment and instrumentation is underway.

- b. Essential results: collapse behavior follows von Mises theory of inelastic collapse, but pressures required are not predictable from tensile yield stress values.
- 5. Next steps: Conduct research program planned.
- 6. Relation with other projects: Fuel Model Development (ANC).
- 7. Reference documents:

ORNL-TM-----Quarterly Progress Reports on Reactor Safety Programs
Sponsored by the Division of Reactor Safety Research, Vol. 1, Light
Water Reactor Safety (report series).

HNL-TM-----Quarterly Progress Report on Reactor Safety Programs
Sponsored by the Division of Reactor Safety Research, Vol. 1., Light
Water Reactor Safety (report series).

Quarterly Progress Report on the Creepdown and Collapse of Zircaloy
Fuel Cladding Program Sponsored by the NRC Division of Reactor Safety
Research: by D. O. Hobson:

July-September 1975, ORNL-TM-5149

October-December 1975, ORNL-TM-5279

April-June 1976, ORNL/NUREG/TM-52

Zircaloy Fuel Cladding Collapse Test Plan, D.O. Hobson, ORNL/NUREG/TM-45, Aug 1976

Interim Report on the Creepdown of Zircaloy Fuel Cladding, D.O. Hobson and
C.V. Dodd, ORNL/NUREG/TM-103, May 1977

- 8. Degree of availability: NTIS

7.6 R.H.Chapman, Multirod Burst Test Program Quarterly Progress Report:

October-December 1975	ORNL/NUREG/TM-10
January-March 1976	ORNL/NUREG/TM-36
April-June 1976	ORNL/NUREG/TM-74
July-September 1976	ORNL/NUREG/TM-77
October-December 1976	ORNL/NUREG/TM-95
January-March 1977	ORNL/NUREG/TM-108

8. Degree of availability: NTIS

PROJECT TITLE : RELIABILITY OF CLASSICAL NDT	CSNI INDEX NUMBER : 12.3 11.2.3 LWR 11.1
SPONSORING COUNTRY : Commission of the European Communities	ORGANISATION : JRC Ispra
DATE INITIATED : 1977 DATE COMPLETED : 1980	PROJECT LEADER : S.J. CRUTZEN

Description :

1. GENERAL AIM

Among the actual non-destructive testing techniques on nuclear reactors components, ultrasounds and radiography are principally used. The experience however teaches that often dubious results are obtained when NDT techniques are applied. Development and assesment of ultrasonic inpection methods require more precision, resolution and reproductibility. Besides global detection and localization methods, sizing techniques are required. Classical ultrasonic testing could give better reliability if known correctly in its parameters. Therefore, characterisation studies leading to simple methods to be used or on the spot or in maintenance laboratories or in research laboratory are proposed.

2. PARTICULAR OBJECTIVES

1. Critical analysis of the transfer function of standard ultrasonic testing chain in order to assess the influence of the various components and of the operational conditions, on the overall reliability.
2. Study of the correlations between U.S. signals and reference artificial defects. This study will involve the use of the NDT Techniques, destructive techniques and the fabrication of well defined reproducible artificial defects.
3. Practical applications
 - a) characterisation of ultrasonic apparatus (in particular transducers) to be used for quality control or in service inspection, Up to now, requests were received from : ENEL (Lab.Centrale,Piacenza) CNEN,CEA-Cadarache, IRSID, NERATOM, BREDA, DOEL Reactor, NUKEM. Collaboration is foreseen with : AERE-Harwell, CEA(Saclay and Fontenay-aux-Roses) NUKEM, DANISH WELDING INSTITUTE, ASSOCIATION VINCOTTE and transducer manufacturers (up to now: AEROTECH, BALTEAU, NUKEM, CEA, ISTITUTO CORBINO)
 - b) HSSTP (PISC).Ispra will receive the test pieces in Summer 1977 and wil try to answer the specifications but also take the opportunity to demonstrate the necessity of characterisation techniques for ultrasonic transducers, for reproducibility in defects detection and localisation A collaboration will be set-up between ENEL(Lab.Centrale di Piacenza) and EURATOM (NDT Section) as the final report has to be unique for Italy. Ispra will take care mainly of what concerns characterisation and calibration as well as original methods other than the one's

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specified by the ASME XI (foçal probes).

- c) Identification of the failed fuel pins in an irradiated BWR fuel bundle using ultrasonic without any dismantling of the bundle. This activity was initiated in the frame of the Technical Support to Power Stations Programme and will be performed in collaboration with CNEN-ENEL on the practical case of the GARIGLIANO-CAORSO fuel bundles (with the already received agreement from the ACPM).
- d) In the frame of a collaboration contract (EUR-CEA-CNEN-203/75/PIHOIF) included in the Safety Programme, some work will be persued for the quantitative direction of cracks (in quantity and in depth), in sodium ducts submitted to thermal shocks. The feasibility study demonstrated the reliability of the X-ray techniques combined with the use of well defined reference defects and of a densitometer.

3. PROJECT STATUS

This study is in fact the partial continuation of a study initiated under the name "Quality Control" in the frame of the "Technical Support to Power Stations" Programme of the CEC. Laboratories are well equiped for:

- rather fundamental studies and results are available for what concern characterisation of transducers and materials
- systematic characterisation of ultrasonic equipment
 - Schieren bench (high resolution)
 - point by point precise examination
 - correction of transducers
 - liquid crystal characterisation bench
- fabrication of well shaped and reproducible reference defects using
 - punch method
 - electric discharge method.

4. NEXT STEPS

- 4.1. Rather fundamental studies concerning the characterisation of the transducer : intrinseque characteristics;
- 4.2. Characterisation of some materials using
 - echo phase shift analyses
 - spectrum analyses
- 4.3. Adaptation of the knowhow of the punch and electric discharge techniques for the fabrication of reference defects in the frame of Inservice Inspection of reactor pressure Vessels.
Utilisation of forming and brazing techniques for fabrication of reference defects in reference blocks.

- 4.4. Characterisation of the Ultrasonic equipments of ENEL (Laboratorio Centrale di Piacenza)
Characterisation of some probes for NERATOOM.
- 4.5. Participation at the HSST(Pisc)programme (September-October 77)
- 4.6. Definition and fabrication of the prototype apparatus for identification of the failed fuel pins in the GARIGLIANO BWR irradiated fuel bundles (CNEN-ENEL-ALCI)
- 4.7. Definition of the resolution of the "X-rays-densitometry" method for the quantitative (depth) detection of cracks in sodium tubing submitted to thermal shocks.

5. REFERENCE DOCUMENTS

- (1) Proceedings of "Information Meeting on Characterisation of Ultrasonic Equipment" S.Crutzen and all, Ispra, Novembre 25th and 26th, 1976
Euratom J.R.C.-Ispra NDT Section (to be issued)
- (2) E.E.BORLOO, P. JEHENSON
Caractérisation des transducteurs ultrasonores.
St.Erienne (France) 30 Novembre et 1er Décembre, 1976
Société Française de Métallurgie. Colloque sur les progrès dans les méthodes d'investigations des métaux.
- (3) R. DENIS
Characterisation of ultrasonic transducers using Cholesteric Liquid Crystals. J.R.C.-Ispra, Materials Division
NDT Section EUR 5710.e, 1976
- (4) S.CRUTZEN, R.DEBEIR, B.JEHENSON, and F. LUCHTMANS
JRC-Ispra, Italy
A. BAROSI, L. ROSAI, T.A. GIORGI, Saes-Getters, Milan, Italy
Use of neutron Radiography for quantitative measurement of sorbed Hydrogen in Getters and quality control of nuclear fuel pins - IAEA/SR-7-21, Oslo, May 1976.
- (5) J.R.C. -ISPRA, Biannual Report 1974-1975 EUR 5550.e

6. BUDGET

- 6.1. Total investments on the past programme actions to be continued (Technical Support to Nuclear Power Stations)
nearly 100.000 u.a.
- 6.2. Total investment for 1977-1980 / \approx 200.000 u.a.

7. PERSONNEL

1977-1980 : 6 men/year

	<p>CLASSIFICATION: 11.1</p>
<p>TITLE: Experimental studies of UO₂ fuel densification.</p>	<p>COUNTRY: Norway</p> <p>SPONSOR: OECD Halden Reactor Project</p> <p>ORGANIZATION: " " " "</p>
	<p>PROJECT LEADER: K.D. Knudsen</p>
<p>INITIATED: 1973</p> <p>STATUS: Near completion</p>	<p>COMPLETED:</p> <p>LAST UPDATING: April 1977</p>

1. *General Aim* To generate experimental data on UO₂ fuel densification.
2. *Particular Objectives* Determine the effect of UO₂ pellet physical characteristics on fission induced fuel densification.
3. *Experimental Facilities* A special rig containing 12 fuel rods, each rod having a fuel stack elongation detector which is monitored continuously to measure fuel stack length changes as a function of power and exposure.
4. *Project Status* The experiment has been initiated.
5. *Progress to Date* The twelve fuel rods have reached a maximum burn-up of 17,000 MWd/t UO₂. Seven rods continue to give reliable information, showing that stack length has either been stabilized or reached a minimum. Swelling appears to be occurring in two high density rods.
6. *Next Step* Continue irradiation to measure fission product swelling of the fuel.
7. *Reference Documents* Sixteenth Annual Report, OECD Halden Reactor Project, 1975, page 31.
8. *Degree of Availability* Upon request from the Nuclear Energy Agency.

NUCLEAR
SAFETY RESEARCH INDEX

1977

VOLUME II
WATER REACTORS
(continued)

11.2. Steel pressure vessel, pressure vessel internals
and primary circuit

PROJECT TITLE : The crack closure effect in crack propagation under narrow band random loading	CSNI INDEX NUMBER : 11.2
SPONSORING COUNTRY : Australia.	ORGANISATION : University of N.S.W.
DATE INITIATED : 1976 DATE COMPLETED : Continuing	PROJECT LEADER : Dr.K.P. Byrne, Dept., of Mechanical & Industrial Engineering.

Description :

The aim is to investigate the crack closure effect when fatigue cracks are propagated under narrow band random loading. Stresses associated with acoustically excited vibrations in reactors can cause fatigue cracks to grow, and such stresses are often of a predominately narrow band random nature.

A small testing machine has been built, capable of applying in-plane sinusoidal and narrow band random loads on sheet specimens. Crack growth rates will be measured and possible mechanisms evaluated for propagation under these loading conditions.

The work receives some support from the Australian Institute of Nuclear Science and Engineering.

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31.719

PROJECT TITLE : NUCLEAR REACTOR MATERIALS RESEARCH	CSNI INDEX NUMBER : 11.2
SPONSORING COUNTRY : FINLAND	ORGANISATION TECHNICAL RESEARCH CENTRE OF FINLAND REACTOR MATERIALS RESEARCH
DATE INITIATED : 1970 DATE COMPLETED : CONTINUES	PROJECT LEADER : KARI TÖRRÖNEN

Description :

1. GENERAL AIM

To generate basic experimental data and knowledge on materials and components as well as their behaviour in light-water cooled nuclear steam supply systems.

2. PARTICULAR OBJECTIVES

- Materials selection
- Fabrication methods (welding, heat treating, etc.)
- Behaviour of stressed materials and components (fracture toughness, fatigue)
- Corrosion

The main emphasis is on pressure vessel steels, stainless steels and Ni-base alloys.

3. PROJECT STATUS

Detailed fracture analysis have been performed on Cr-Mo-V pressure vessel steel base material, the analysis of the main butt weld is under way. The critical heat treatment parameters of Cr-Mo-V, A 533 B and A 508 pressure vessel steels have been evaluated. The microstructure, the mechanical properties and the heat treatment parameters have been unambiguously correlated. Mechanical properties: the structure and corrosion properties of inconel-type alloys and Ti-stabilized 18/10 and 18/12 stainless steel base materials and overlay welded claddings have been investigated.

(cont.)

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PROJECT TITLE : NUCLEAR REACTOR MATERIALS RESEARCH	CSNI INDEX NUMBER : 11.2
SPONSORING COUNTRY : FINLAND	ORGANISATION : TECHNICAL RESEARCH CENTRE OF FINLAND REACTOR MATERIALS RESEARCH
DATE INITIATED : 1970 DATE COMPLETED : CONTINUES	PROJECT LEADER : KARI TÖRRÖNEN

Description : (cont.)

4. REFERENCE DOCUMENTS

Reactor Materials Research reports (partly confidential),
conference papers etc.

5. DEGREE OF AVAILABILITY

Upon request from the Technical Research Centre of Finland.

Titre Développement des méthodes de calcul pour l'analyse mécanique des circuits primaires des réacteurs à eau.	Pays : FRANCE Organisme directeur : CEA/DSN
Titre (anglais) Advances in computation methods for the mechanic analysis of PWR primary systems.	Organisme exécuteur : CEA/DEMT Responsable : B. BARRACHIN (DSN-SETSSR) R. ROCHE (DEMT)
Date de démarrage : 1/01/76 Date prévue d'achèvement : 31/12/81 Etat actuel : en cours Dernière mise à jour : 1/1/77	Scientifiques :

Objectif général :

Cette action vise la tenue mécanique du circuit primaire principal des réacteurs nucléaires à eau sous pression.

Objectifs particuliers :

- Amélioration de la définition et de la validation des critères utilisés dans les codes de construction.
- Calculs d'éléments de structure
- Analyse des règles du code ASME section III
- Développement des méthodes de calcul des structures mécaniques
- Application et développement de la mécanique de la rupture.

Etat de l'étude :

1) Avancement à ce jour :

- Mise au point de codes de calcul 2D et 3D (option plasticité)
- Application des structures de PWR et vérification des analyses effectuées par le constructeur
- Etude critique du domaine de validité de la LEFM

2) Résultats essentiels :

- Analyse détaillée du comportement des tubulures de sortie d'une cuve PWR 900.
- Validité de la LEFM. Comparaison avec les méthodes élasto-plastiques (Intégrale de RICE)

Prochaines étapes :

- Analyse détaillée de la fermeture d'une cuve PWR 900
- Instabilité plastique des composants : vérification des coefficients de sécurité.
- Validité de la LEFM pour des structures et changements complexes.

Titre Evaluation de la probabilité de rupture d'une cuve de réacteur à eau	Pays : FRANCE Organisme directeur : CEA/DSN
Titre (anglais) Failure probability calculation of a water reactor pressure vessel.	Organisme exécuteur : CEA+EURATOM+FRAMATOME Responsable : J. DUFRESNE (DSN-SETSSR)
Date de démarrage : 01/01/76 Date prévue d'achèvement : 31/12/77 Etat actuel : En cours Dernière mise à jour : 18/11/76	Scientifiques : Répartis dans différents laboratoires.

Objectif général :

Evaluation de la probabilité de rupture d'une cuve de réacteur à eau.
 Comparer cette probabilité avec la probabilité de rupture des canalisations du circuit primaire.
 Identifier le poids relatif des différents paramètres intervenant dans la rupture d'une cuve.
 Rechercher l'emplacement des ruptures les plus probables.

Objectifs particuliers :

Relever des dimensions de défauts dans les cuves de réacteur.
 Rassembler les données sur le comportement des matériaux. Définir les conditions de fonctionnement programmées et accidentelles.
 Valider les lois de la mécanique des ruptures. Valider les lois de propagation des fissures. Valider les critères de rupture. Etablir un code général à partir d'histogrammes. Etudier la propagation des fissures dans des conditions particulières (défauts réels, mode II et III, propagation sous eau, à basse fréquence).

Installations expérimentales et programme :

Travaux effectués avec le concours d'EdF et de Framatome, ainsi que, par contrats, avec d'autres industries et universités. Les calculs et mise en oeuvre de codes sont effectués au CCR Ispra.

.../...

Etat de l'étude :

1) Avancement à ce jour :

Recensement des formules de calcul des coefficients d'intensité de contrainte. Recherche sur les résultats publiés de ruptures fragiles sur éprouvettes. Recherche bibliographique sur les lois de propagation des fissures. Analyse des données statistiques sur les ruptures de récipients conventionnels. Analyse des différentes méthodes d'évaluation probabiliste de ruptures de cuves. Recueil de données sur la propagation de défauts dans des récipients conventionnels en cours d'exploitation. Définition des principales situations auxquelles est soumise la cuve. Modification du code COVAL (EURATOM Ispra) pour tenir compte du degré de corrélation de plusieurs variables.

Résultats essentiels :

La formule de calcul du coefficient d'intensité de contrainte établie par Shah et Kobayaski est pessimiste de 15 %. La dispersion des résultats est de ± 10 %.

Prochaines étapes :

Validation des critères de rupture. Propagation des défauts réels. Recueil de données sur les défauts de cuve. Mise en place du code de calcul.

Documents de référence :

"Programme d'étude concernant l'évaluation de la probabilité de rupture d'une cuve de réacteur à eau", J.DUFRESNE - Rapport DSN 94.
"Etude probabiliste de la rupture de cuve de chaudière nucléaire à eau ordinaire, premier rapport d'avancement, 15-3-76/15-9-76", J.DUFRESNE - Rapport DSN 116.

Titre Recherche des procédures simplifiées pour l'évaluation des facteurs d'intensité de contrainte dans les enceintes sous pression.	Pays : FRANCE
	Organisme directeur : CEA
Titre (anglais) Research of simplified procedures for the evaluation of stress intensity factor in the pressure vessel.	Organisme exécuteur : GAAA
	Responsable : B. BARRACHIN
Date de démarrage : 01/10/76 Date prévue d'achèvement : 01/09/77 Etat actuel : en cours Dernière mise à jour : 19/11/76	Scientifiques : S. BANDHARI

Objectif général :

Cette étude a pour but d'élaborer des procédures simplifiées pour l'évaluation des facteurs d'intensité de contraintes pour des fissures dans des ouvertures renforcées sous divers chargements.

Objectifs particuliers :

Application à la cuve d'un réacteur PWR.

Etat de l'étude :

Avancement à ce jour :

Démarrage. Etude bibliographique des cas bidimensionnels et tridimensionnels.

Titre Etudes de sûreté des composants du circuit primaire des réacteurs PWR.	Pays : FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Safety analysis of PWR primary circuit components.	Organisme exécuteur : CEA/DSN - SETSSR
	Responsable : B. BARRACHIN (DSN - SETSSR)
Date de démarrage : 01/01/77 Date prévue d'achèvement : 31/12/79 Etat actuel : A lancer Dernière mise à jour : 15/12/76	Scientifiques :

Objectif général :

Ces études ont pour but de préciser le comportement des composants du circuit primaire des réacteurs PWR en conditions accidentelles.

Objectifs particuliers :

Comportement de l'acier de cuve en milieu pressurisé influence du beurrage.
 Interprétation des règles ASME en conditions accidentelles.
 Comportement de composants particuliers en conditions accidentelles.

<u>Classification: 11.2</u>	
<u>Title 1 (Original Language):</u> Statusbericht Reaktordruckbehälter (RA 63 - II.2., Jahresbericht A 76)	<u>COUNTRY:</u> BRD
	<u>SPONSOR:</u> BMFT
	<u>ORGANIZATION:</u> IRS, Köln
<u>Title 2 (English):</u> Status Report Reactor Pressure Vessel	<u>Project Leader:</u> Röhrs Rittig
<u>Initiated (Date):</u> Dec. 28, 1971	<u>Completed (Date):</u> December 1976
<u>Status:</u> continuing	<u>Last Updating (Date):</u> December 1976

1. General Aim

Within the Reactor Safety Research Program the studies relating to the reactor pressure vessel technique are one of the key points. This Status Report was made to determine really substantive priorities for the use of the available funds.

2. Particular Objectives

In detail the following targets are aimed:

1. To show the latest status of the technical development to the Federal Republic of Germany with respect to design, fabrication, test and examination of reactor pressure vessels (RPV) for light water reactors fabricated by means of rolling construction or forged steel construction by pointing out the principal aspects for the safety assessment.
2. To set up a research program for the extension of the basis of the RPV safety assessment and for an improvement of the techniques of design, manufacturing, and test.
3. Concentrated representation of targets, research programs, and essential results of the 12 topics of the Heavy-Section-Steel-Technology- (HSST) Program executed in the USA.

3. Program

To provide a broad spectrum of experiences and guaranteed objectivity to a large degree it was decided to proceed as follows.

By order of the Federal Minister for Education and Science (BMBW) the Institute of Reactor Safety (IRS) took the lead. Experienced industrial firms, associations, and safety consultant institutions elaborated special contributions according to specifications and by order of the IRS. A group of independent experts analysed these reports with respect to principle essentials as to the safety assessment of RPV. An editing board consisting of leading experts with special experience in RPV-technique took the whole leading.

The Status Report will consist of three volumes.

Volume I will contain all results in concentric form. Starting from the state of RPV-technique at the end of 1973 positive as well as negative knowledge aspects will be shown. As the status report will essentially serve for setting up substantiated priorities for research activities the concise form of volume I will concentrate partly strongly on the gaps of knowledge.

Volume II will consist of text and figures of the data- and information collection. Here the status of RPV-technique will be layed down as found out by the authors of the special reports and as coordinated subsequently in detail.

Volume III will give a summary of the HSST Program presently running in the USA. For each of the 12 topics of this program information will be given as regards objective, research program, project status and essential results.

4. Experimental Facilities, Computer Codes

Not relevant.

5. Progress to Date

Drafts of Volume II (Collection of data and information) and Volume III (HSST-Program) were finished.

6. Results

Volume III has been printed. The editing board has reviewed Volume II.

7. Next Steps

Preparing printing of Volume II.

8. Relation with other Projects

RS 46 Study on Current and Planned Research
Projects in the Pressure Vessel Field
STAATLICHE MATERIAL PRÜFUNGSANSTALT, Stuttgart, University
1969 - 1971

9. References

Reports in the series IRS-Forschungsberichte respectively

GRS-Fortschrittsberichte:

1975 IRS-F-29 (in English)
Jan.-March 1976 IRS-F-30 (in German)
April - June 76 IRS-F-31 (")
July - Sept. 76 IRS-F-33 (")
Oct. - Dec. 76 GRS-F-35 (")

10. Degree of Availability of the Reports

Unrestricted distribution.

<u>Classification: 11.2</u>	
<u>Title 1 (Original Language):</u> Grob- und Detailspezifikation des Forschungsprogramms Komponentensicherheit (1. Phase) (RS 192 - II.2, Jahresbericht A 76)	<u>COUNTRY:</u> BRD
	<u>SPONSOR:</u> BMFT
	<u>ORGANIZATION:</u> MPA Stuttgart
<u>Title 2 (English):</u> Rough and detailed Specification Research Program "Structural Integrity of Components"	<u>Project Leader:</u> Dr.-Ing. D. Sturm Dr.-Ing. L. Issler
<u>Initiated (Date):</u> November 13, 1975 <u>Status:</u> completed	<u>Completed (Date):</u> April 30, 1976 <u>Last Updating (Date):</u> December 31, 1976

1. GENERAL AIM

The research program "Structural Integrity of Components" integrates all activities concerning reactor pressure vessel problems running or planned in the Federal Republic of Germany. Objectives of this program are to give basic information for the quantification of safety margin in order to reduce further the remaining risk through a specified further development of the reactor pressure vessel technique.

2. PARTICULAR OBJECTIVES

In order to be able to work on this research program "Structural Integrity of Components" with largest possible efficiency it has been planned to divide the program into subsequent phases. For the first phase, which will last approx. 4 years, a rough specification had to be worked out and in addition to the first year a detailed specification.

3. RESEARCH PROGRAM

The first phase of the research program "Structural Integrity of Components" covers the following single projects:

- 1) Possibility of Defect Detection (Defect Classification)

- 2) Homogeneity of the Mechanical Properties (Small Scale Specimens - not irradiated, irradiated)
- 3) Integral Test on Large Scale Specimens
- 4) Integral Test on Intermediate Size Vessels
- 5) Theory

They had to be classified in regard to basic proceedings including test setup, test performance, test time, temporal operating sequence and costs.

4. EXPERIMENTAL FACILITIES, COMPUTER CODES

not necessary for the specification.

5. PROGRESS TO DATE

Working-out of the specification which consists of a rough specification for the complete project of the first phase and of a detailed specification for the first test year. Preparation of the structure of targets, structures of projects, network plans and lists of costs.

6. RESULTS

Completion of the specification.

7. NEXT STEPS

not applicable, as project has been completed.

8. RELATION WITH OTHER PROJECTS

See Annual Report A 75.

9. REFERENCES

Quarterly Reports in IRS Research Reports, period reported on: January 76 until June 76 (German).

Annual Report A 75 (English)

10. DEGREE OF AVAILABILITY OF THE REPORTS

Upon request at BMFT.

<p><u>Title 1 (Original language)</u> Ricerca sperimentale sull'accumulo delle deformazioni e sulle rotture per fatica di tubazioni soggette a carichi variabili ripetuti.</p>	<p><u>Classification</u> 11.2</p>
<p><u>Title 2 (English)</u> Experimental research on incremental collapse and plastic fatigue of piping components.</p>	<p><u>Country</u> ITALY <u>Sponsor</u> CNEN <u>Organisation</u> University of Pisa</p>
<p><u>Date initiated</u> 1976 <u>Date completed</u> 1980 <u>Last updating</u> May 1977</p>	<p><u>Project Leader</u> NERLI G.</p>

This research is concerning with extending understanding of (i) incremental collapse phenomena, with progressively increasing deflections, and (ii) plastic fatigue phenomena in stainless steel piping components subjected to variable repeated loads.

Loads will be determined that produce (i) permanent strains and deformations in excess of the design limits and (ii) plastic fatigue ruptures in T branched pipes fixed at their flanged ends and loaded in two sections by variable repeated loads. The reference is to "emergency" and "faulted" load conditions as they are defined in ASME III Code.

The models are fabricated by stainless steel pipe and simulate some primary circuit piping components. The models will be tested using an existing experimental rig.

Experimental incremental collapse results will be compared with calculated ones by an existing computer program based on finite element technique. The program takes into account the effects of material strain hardening, where Section III of the ASME Code is based primarily on a limit analysis approach, that is on an elastic-perfectly plastic behaviour of the material.

One important objective of the research is to determine material and weld (of flanges) qualification parameters to be used for obtaining the best results by the calculus.

Some researches on plastic behaviour of carbon and stainless steel pipes were just performed at "Istituto di Impianti Nucleari" and at "Istituto di Ingegneria Meccanica" of Florence University/1,2,3,4,5/.

The results of the present research will be entirely published.

The research will be performed under a contract with CNEN.

References

1. - A. DEL PUGLIA, G. FILIPPELLI, G. NERLI
"Esperienze sul collasso di tubazioni iperstatiche non piane"
Disegno di Macchine N. 3 (1973).

<u>Title 1 (Original language)</u> Ricerca sperimentale sull'accumulo delle deformazioni e sulle rotture per fatica di tubazioni soggette a carichi variabili ripetuti.	<u>Classification</u> 11.2
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2. - A. DEL PUGLIA, G. NERLI
"Experimental Research on Elastoplastic Behaviour and Collapse Load of Statically Indetermined Space Tubular Beams"
2nd Int. Conf. on Structural Mechanics in Reactor Technology, Berlin, Sept. 1973.
3. - P. CITTI, A. DEL PUGLIA, G. NERLI
"Experimental Study on the Effect of Variable Repeated Loads on Steel Piping Components"
3rd Int. Conf. on Structural Mechanics in Reactor Technology, London, Sept. 1975.
4. - P. CITTI, A. DEL PUGLIA, G. NERLI
"Esperienze su tubazioni soggette a carichi variabili ripetuti"
3rd Convegno Nazionale AIAS, Bologna, Oct. 1975.
5. - P. CITTI, P. RISSONE
"Calcolo e verifica sperimentale della deformata di una travatura in campo elastico-incrudente"
IV Convegno Nazionale AIAS, Roma, Set. 1976.

<p><u>Title 1 (Original language)</u> Fatica ad alta temperatura di elementi strutturali</p>	<p><u>Classification</u> 11.2</p>
<p><u>Title 2 (English)</u> High Temperature Fatigue on Structural Elements</p>	<p><u>Country</u> ITALY <u>Sponsor</u> CNEN <u>Organisation</u> University of PISA</p>
<p><u>Date initiated</u> 1976 <u>Date completed</u> 1980 <u>Last updating</u> May 1977</p>	<p><u>Project Leader</u> A. DEL PUGLIA</p>

Description:

Research program: to gather experimental information on the High Temperature Low Cycle Fatigue behaviour of simple tubular welded structures in AISI 304, 316 and other materials. The load conditions are characterized by:

- temperature (mainly comprised between 430° and 650°C)
- imposed strain (ranging from 0,5 to 2%)
- hold time (from 0 to 30 min and over).

A particular effort has been performed at properly characterizing the specimens and related manufacturing technology.

Up to now have been tested butt welded pipe segments with flanged ends, 500 mm long and with 60,3 mm outer diameter.

Facilities: test apparatus, which is able to house specimens up to 650mm length and to apply a 20 ton. maximum axial load.

Temperature and load cycle control instrumentation.

Load, axial and diametral deformation transducers and related instrumentation.

Acoustical emission instrumentation.

Reference documents

1. - DEL PUGLIA A.

Sul comportamento a fatica ad alta temperatura di elementi strutturali in acciaio inossidabile di tipo austenitico.

Atti Ist. Meccanica Applicata e Costruzione di Macchine - Università di Pisa n° 50 - Pisa (1974).

<u>Title 1 (Original language)</u>	<u>Classification</u>
Fatica ad alta temperatura di elementi strutturali	11.2

2. - DEL PUGLIA A., BARTOLI G.
Prove preliminari ad alta temperatura di elementi tubolari saldati in acciaio inossidabile.
Atti III° Convegno AIAS - Bologna 1975.
3. - DEL PUGLIA A., MANFREDI E., CABRELE G., ZOLA S.
Fatica ad alta temperatura di elementi strutturali tubolari saldati in acciaio inossidabile: primi risultati.
Atti IV Convegno AIAS - Roma (1976).
4. - DEL PUGLIA A., MANFREDI E., TOMMASSETTI G.
High Temperature Fatigue Experiments on Welded Stainless Steel Structures.
4th International SMIRT Conference - San Francisco (1977)

<p><u>Title 1 (Original language)</u> Sull'inizio di propagazione di una rottura in un recipiente in pressione costruito con materiale duttile.</p>	<p><u>Classification</u> 11.2</p>
<p><u>Title 2 (English)</u> On the starting of a crack propagation in a cylindrical pressure vessel of ductile material.</p>	<p><u>Country</u> ITALY <u>Sponsor</u> CNEN - CNR <u>Organisation</u> University of Pisa</p>
<p><u>Date initiated</u> 1968 <u>Date completed</u> 1980 <u>Last updating</u> May 1977</p>	<p><u>Project Leader</u> CARMIGNANI Costantino</p>

Description

The aim of this research program is to evaluate the possibility of using linear elastic fracture mechanics concepts to forecast the starting of crack propagation in a ductile material.

In the first phase it was developed a theoretical work to evaluate the elastic stress field around the tip of the crack (Ref. 1.2). After, several cylindrical pressure vessels with axial through thickness cracks, were bursted by internal pressure (Ref. 3.4.5). The results of this experimental work demonstrated the possibility of using cylindrical pressure vessels, like those above mentioned, to determine the critical parameters (K_c or COD) of the material.

Now an experimental work is completed to evaluate size effect on the determination of the critical parameter with the method above indicated (Ref. 6).

At present we are working on an experimental research with the aim to evaluate the behaviour, by using linear and elasto-plastic fracture mechanics concepts, of cylindrical vessels with axial or circumferential throughwall, or partial-through, cracks.

Vessels will be loaded with internal pressure and/or bending moment.

Facilities used are a pressurization system with an electric/manual pump for bursting tests at room temperature and a frame for bending tests.

Relation to other projects

Research outlined is strictly selected to joint experimental programs developed by CSM (Centro Sperimentale Metallurgico), Palermo University Laboratory and CNEN.

Reference documents

1. CARMIGNANI C., DEL PUGLIA A.

Applicazioni di metodi di calcolo elettronico all'analisi del comportamento elastico di piastre rettangolari contenenti una fessura passante centrale, caricate normalmente al loro piano medio.

Atti Ist. Mecc. Appl. - Univ. Pisa, n° 11 - Anno Acc. 1968-'69.

<u>Title 1 (Original language)</u>	<u>Classification</u>
Sull'inizio di propagazione di una rottura in un recipiente in pressione costruito con materiale duttile.	11.2

2. CARMIGNANI C., CELLA A.
Elastic Analysis of Cracked Thin Shells by the Finite Element Method.
Proc. 1st SMIRT, Berlin, 1971, vol. 5 pp. 57+ 76.
3. CARMIGNANI C.
Ricerca teorica e sperimentale sulla propagazione delle rotture nei recipienti in pressione.
Atti del Convegno Nucleare di Pisa, vol. 1, CNEN, 1971, pp.279+ 319.
4. CARMIGNANI C., REALE S.
Elaborazione dei risultati della ricerca sperimentale sulla propagazione delle rotture nei recipienti in pressione.
Atti Istituto di Impianti Nucleari - Univ. Pisa. RP 153(73), 1973.
5. CARMIGNANI C., CIBECCHINI P., REALE S.
Caratterizzazione di un acciaio al Nichel per impieghi a bassa temperatura con i criteri della meccanica della frattura.
Proc. 5th Int. Conf. on Exp. Stress Analysis, Udine, 1974, paper 18 pp. 2.5 + 2.12.
6. CARMIGNANI C., REALE S.
Valutazione dell'effetto scala nella determinazione del parametro critico del materiale, mediante prove di scoppio di modelli cilindrici fessurati.
Atti del IV Convegno Nazionale AIAS, Roma, Sept. 1976.

PROJECT TITLE : Pipe Rupture Tests of Primary Coolant System of LWR	CSNI INDEX NUMBER : 11.2
SPONSORING COUNTRY : Japan	ORGANISATION : JAERI, Tokai
DATE INITIATED : April 1975 DATE COMPLETED : Progressing (March 1977)	PROJECT LEADER S. Miyazono

Description :

1. Objectives

The objectives are to investigate the behavior of crack initiation and propagation of single or plural cracks by cyclic loading at room temperature, to study whether cracks grow to the size of critical length or not, and to study the behavior of hot water or steam jet and the pipe whipping by pipe reaction following blowdown using the piping models of primary coolant system of Light Water Reactors.

2. Project Status

Test facilities are now being constructed for these tests.

3. Essential Results

None

4. Projection

The Fatigue Test Facility (PRPC-1) will be completed in August 1977 and the Pipe Rupture Test Facility (PRPC-2) will be completed in January 1978. The tests will be immediately started after the completion of these test facilities.

5. Report

None

TH-Delft		CLASSIFICATION: 11.2	
TITLE: Elastisch-plastisch breukmechanica onderzoek aan stompen van nucleaire hoge-drukvaten (EPOSS)		COUNTRY: NETHERLANDS. SPONSOR: Ministry of Social Affairs ORGANIZATION: TH-Delft	
TITLE: (ENGLISH LANGUAGE): Elastic-plastic fracture analysis research on cracks at nozzle intersections of nuclear high-pressure vessels (EPOSS)		PROJECTLEADER: Latzko	
INITIATED: August 1974	LAST UPDATING: May 1977	SCIENTISTS: V.d. Ruijtenbeek Broekhoven Peeters	
STATUS: in progress	COMPLETED: 1977 (scheduled)		

General aim

Crack extension behaviour in heavy section nuclear steel pressure vessels in areas of complicated geometry.

Particular objectives

Evaluation of the applicability of the J-integral concept (an elastic-plastic fracture mechanics concept) for predicting elastic-plastic crack extension for complex crack configurations in nuclear pressure vessels, notably cracks in nozzle corner regions.

Experimental facilities and programme

Main activities are :

a. theoretical investigations

- * Computation of J-integral values by the finite element method for 2-dimensional configurations, ranging from simple test specimens to uniaxially loaded plates with cracks emanating from a central hole;
- * Computation of J-integral values by the finite element method for some 3-dimensional configurations, viz.
 - bars with a quarter-circular edge crack
 - uniaxially loaded plates with quarter-circular cracks emanating from a hole
 - flat plates with a central nozzle and a crack at the nozzle corner
- * Evaluation of the applicability of simplified approximation procedures to determine J for said configurations.

b. experimental investigations

- * J_{Ic} -tests on standard specimens for the model material, i.e. Al 2024-T3
- * Model-tests on 2-dimensional configurations, i.e. uniaxially loaded plates with cracks emanating from a central hole
- * Model-tests on the 3-dimensional configurations mentioned under a.

Project status

- * Procedures for efficient computation of J-values by the finite element method have been established
- * Computations of J-values for simple 2-dimensional configurations have been completed
- * Computations of J-values for more complicated 2-dimensional configurations are in progress
- * J_{Ic} -tests on standard specimens are in progress
- * Experimental investigations on uniaxially loaded plates with cracks emanating from a central hole are in progress

Next steps

Continuation of investigations of 2-dimensional configurations; initiation of theoretical and experimental investigations on 3-dimensional configuration

Relation with other projects

The study is an extension of the BROS-project into the elastic plastic regime

Reference documents

Report MMPP-110, Delft Un. of Techn., Lab. for Thermal Power and Nucl.Engineering

Degree of availability

Through Ministry of Social Affairs, C.R.V., Postbox 69, Voorburg, Netherlands

Budget

Approx. Dfl. 750.000

Personnel

100 Manmonths.

Rotterdam Dockyard and others		CLASSIFICATION: 11.2
TITLE: <u>Breukanalyse Onderzoek aan Stompen (BROS)</u>		COUNTRY: NETHERLANDS. SPONSOR: Ministry of Economic Affairs ORGANIZATION: Rotterdam Dockyard and others
TITLE: (ENGLISH LANGUAGE): Fracture analysis research on nozzle intersections (BROS)		PROJECT LEADER: C.J. Drijver
INITIATED: March 1972	LAST UPDATING: Febr. 1976	SCIENTISTS: M.J.G. Broekhoven H.J.M. van Rongen A. de Sterke M.J.A. Koning
STATUS: in progress	COMPLETED: End 1977	

General aim

Crack extension behaviour in heavy section nuclear steel pressure vessels in areas of complicated geometry.

Particular objectives

In particular attention will be paid to cracks occurring in areas of complicated geometry and stress distribution, notably nozzle corner regions. The program covers the following items:

- early detections of defects
- detailed surveillance of the growth of defects
- establishment of prediction for further growth of defects by fatigue and fracture.

Experimental facilities and program

The main activities are:

- theoretical research directed towards the development of efficient computation methods to calculate the crack extension behaviour for complicated configurations, such as nozzle corner cracks, using advanced analytical and finite element techniques,
- experimental investigations of crack extension behaviour in uni-axial and bi-axial loaded cracked nozzle-on-flat-plate models, manufactured mainly from nuclear grade pressure vessel steel (ASTM A508 C1.2),
- determination of fatigue and fracture (elastic and elastic-plastic) related material parameters for A508 material,
- research on the applicability of Acoustic Emission techniques to detect, locate and characterize crack extension in the material of concern,
- the main test facility to be used is a bi-axial loading bench, suitable for plate dimensions 700x700x15 mm, capacity 200 tonnes static force, 200 tonnes dynamic loading, frequency 1Hz.

Project status

- A series of nozzle-on-flat-plate models has been uni-axially fatigue loaded and, after sufficient crack growth, fractured by overload.
- Computer programs have been completed.
- Acoustic Emission testing has been completed
- Fatigue and fracture (J_{IC}) tests on standard specimens are in progress.
- The bi-axial loading bench is under construction.

Next steps

The next steps will be the testing of flat plate test pieces using the bi-axial loading bench and the publishing of the final technical reports.

Relation with other projects

Results of this project will be used within the "EPOSS" project.

Reference documents

A series of about 50 technical and progress reports have been prepared. Nearly all these reports are written in the Dutch language.

Degree of availability

The reports have been submitted to the Ministry of Economical Affairs, Laan van Nieuw-Oost Indië 123, The Hague. Requests for obtaining copies should be sent to this address.

Budget

f 6.10⁶

Personnel

46 manyears

PROJECT TITLE : Elastic-plastic fracture mechanics	CSNI INDEX NUMBER : 11.2
SPONSORING COUNTRY : Sweden	ORGANISATION : AB Atomenergi
DATE INITIATED : 1974-07-01 DATE COMPLETED : 1977-06-30	PROJECT LEADER : Börje Östensson

Description :

Fracture mechanics evaluation of cracks in reactor pressure vessels has been restricted by the lack of valid fracture toughness data at temperatures above room temperature.

This project aims to derive fracture toughness data between room temperature and 350°C for pressure vessel steel A 533-B. The work is concentrated around the J-integral approach. Methods have been developed to detect the onset of stable crackgrowth e.g. the point on the test record where the fracture toughness should be calculated.

Different methods to detect initial cracks have been developed and tested. The best results have been obtained by measuring the potential drop over the crack, where a direct current is flowing through the rod.

Three point bend specimens ranging in thickness from 25 to 100 mm have been tested at RT, 100°C and 300°C. No influence of thickness has been observed.

In the temperature region around 250°C a decrease in fracture toughness has been observed. The temperature at minimum fracture toughness is influenced of strain rate.

The testing is continued at very low strain rate.

PROJECT TITLE: Critical Experiments, Measurements, and Analyses to Establish a Crack Arrest Methodology for Nuclear Pressure Vessel Steels	INDEX NUMBER: 11.2
SPONSORING COUNTRY: USA	ORGANIZATION: Battelle Columbus Lab.
DATE INITIATED: 9/74 DATE COMPLETED: Continuing STATUS AS OF:	PROJECT LEADER: G. T. Hahn

Description:

1. General Aim:

To provide a complete characterization of crack arrest as related to the primary system of LWR's and especially to the RPV's. This includes critical experiments to validate arrest theory and to verify dynamic analyses; development of a standard testing procedure, including specimen design; acquisition of an arrest toughness data based on RPV steels; and analyses using these data to assure safe operation of LWR's.

2. Particular Results:

a. Program Administration

Efficiently manage the program, monitor subcontracts, procure special materials and equipment and issue progress and topical reports. Assure communication of technical details among the participants in the NRC/EPRI crack arrest program on a continuing basis and participate in NRC/EPRI crack arrest meetings, participate in ASTM and other technical society meetings as required.

b. Dynamic Fracture Mechanics Analyses

Complete a two-dimensional analysis of crack propagation and arrest in a pressurized cylinder subjected to thermal shock. Compare results from this effort with the HSST experiments at ORNL and experiments in the laboratory and report results. Provide computer analysis for load/thermal shock combinations achievable in the laboratory for reduction of data from SEN specimens, both ordinary and duplex. Analyze the run/arrest behavior of a crack that has branched.

c. Standard Test Practice - Unirradiated Steels

Develop a standardized test practice for the measurement of K_{IDm} (arrest fracture toughness) and submit to the ASTM on March 22, 1977. Complete the SEN experiments and evaluate the variables of crack jump length and side grooves. Determine crack arrest specimen measurement capability by testing specimens at thicknesses at least four inches thick.

d. Standard Test Practice - Irradiated Steels

Develop a standardized test practice for the measurement of K_{IDm} (arrest fracture toughness) with due consideration to size limitations imposed by surveillance capsules in operating LWR's, as well as the required measurement capacity. Complete the effort on compound specimen development, and report the advantages, disadvantages, limitations, etc., of the method. Develop the special instrumentation, equipment, and techniques required for remote-handling, crack propagation and arrest experiments.

e. Development of Crack Arrest Toughness Data Base

Develop an unirradiated crack arrest toughness data base for RPV steels using the method(s) developed in Task 2.c., and at toughness levels of practical interest (well above NDT). Obtain one or more full curves of dynamic fracture toughness, K_{ID} , versus velocity, a , for characterization. Plate, forging, weldment, and HSST thermal shock material must be included in the data base. Consideration will be given, depending upon availability, of substitution of weldment for the plate and forging material.

Initiate the irradiation of weldment specimens as developed in Task 2.d., to develop limited crack arrest toughness data. This task is expected to be completed in FY 1978. The specifics of the irradiation level will be provided by NRC. Approximately five specimens are envisioned.

f. Cooperative Program with The University of Illinois

Monitor and report the accomplishments on the dynamic analysis of DCB, SEN, and cracked ring geometries, on K of a rapidly running crack approaching a free surface, and on measurements of K_{ID} . This task must be completed in FY 1977.

g. Round Robin Testing For Arrest Fracture Toughness

Initiate, monitor, and manage a round robin testing program for unirradiated crack arrest toughness using the three standard test practices as submitted to the ASTM. The material will be supplied by NRC, and current plans are for at least seven participants. Perform preliminary tests to assure that the crack arrest toughness data obtained will be at levels of practical interest (well above NDT).

3. Experimental Facilities and Program:
 - a. Testing laboratories at Battelle-Columbus.
 - b. Program: See item 2 above.
4. Project Status:
 - a. Results to date: Crack arrest methodology and standard specimens proposed. Irradiation effects program started.
 - b. Essential results: See item 1 above.
5. Next Steps: See item 2 above.
6. Relation To Other Projects: Part of NRC/EPRI Program on Crack Arrest and closely coordinated with the HSST program.
7. Reference Documents: Quarterly Progress Reports and Topical Reports.
8. Degree of Availability: All available through NTIS.

PROJECT TITLE: HEAVY SECTION STEEL TECHNOLOGY PROGRAM	INDEX NUMBER: 11.2
SPONSORING COUNTRY: USA	ORGANIZATION: OAK RIDGE NATIONAL LAB.
DATE INITIATED: 1965 DATE COMPLETED: CONTINUING STATUS AS OF:	PROJECT LEADER: G. D. WHITMAN

Description:

1. General Aim:

To provide a thorough, quantitative assessment of heavy section reactor vessel fracture characteristics including a realistic evaluation of fracture potential and development of fracture prevention criteria. Especially included are the effects of irradiation; flaw growth mechanisms; significance of thermal shock, considering warm prestressing, crack propagation and arrest; the significance of cracks residing in weld repair regions, in the transition and upper shelf toughness regimes; and crack arrest characterization from model and intermediate vessel tests.

2. Particular Objectives:

a. Program Management, Fracture Mechanics and Analysis

Efficiently manage HSST program, monitor subcontracts, procure special materials and equipment, issue progress and topical reports and organize and conduct program review and planning committee meetings. Participate in ASME, and PVRC meetings as required. Complete the evaluation of equivalent energy and COD methods of elastic-plastic fracture analysis and prepare topical report. Continue evaluation of J-integral method and report results. Perform fracture toughness testing to verify the analytical methods. Provide analytical support for the thermal shock and the ITV test programs, prepare significance reports on intermediate vessel testing on complex geometries and inside surface flaws to include simplified methods of analysis. Issue report on ITV-7A. Complete the photoelastic evaluation of nozzle corner cracks. Perform concurrent finite element analyses of the photoelastic nozzle corner crack experiments and if results correlate well, use finite element method to evaluate cracked feedwater nozzles subjected to pressure, force, moment, and thermal loadings typical of operation. Report of these photoelastic and finite element efforts on nozzle corner cracks. Conduct crack growth analysis of the nozzle corner region. Use these data to recommend test parameters for ITV-10.

b. Fatigue and Crack Propagation

Fatigue crack growth rate and LWR crack growth analyses - continue to develop fatigue crack growth rate data including the effects of material, LWR water chemistry, temperature, R-ratio, irradiation, cyclic rate, hold time, loading rate, etc., compatible with design transient loadings. Submit a conservative upper bound relationship between da/dN and ΔK to NRC to update the current code data. Issue a topical report on updated crack growth analyses on LWR pressure vessels, contingent on the data that are developed.

c. Task III: Irradiation Effects

Complete the irradiation of the low ductile shelf weld material. Check out the methods to be used for obtaining data on ductile shelf testing of the irradiated specimens. Determine the ductile shelf toughness of each weld, starting with the smallest specimens, building up to the largest and performing the minimum number of tests. Each test on larger specimens must have NRC concurrence on test conditions prior to the test. NRC concurrence is also required for the initial tests on small specimens. Evaluate the correlation between the ductile shelf toughness and the C_{y} shelf energy. Publish a final report on the methods used to evaluate high levels of fracture toughness from small specimens. Report the results of this second 4T-CTS irradiation program. Perform another similar irradiation program on material that will be supplied.

d. Task IV: Thermal Shock

Complete verification of the method of analysis used to predict crack propagation and arrest in a reactor vessel subjected to ECCS operation following LOCA. Analyze and report on the significance of the thermal shock analyses and experiments as related to reactor vessel safety. Carefully analyze experiment TSE-3, using arrest toughness as characterized by BCL, on the basis of a run-arrest experiment. The shape of the crack front as it progressed radially and the effect of the plug holes on initiation and propagation must be evaluated. Report these results. Perform a similar experiment, TSE-4, using TSV-2. This must be planned, analyzed and conducted as an initiation and arrest experiment, and must be done in close cooperation with BCL. Report the results. Evaluate the use of LN2 to perform a meaningful thermal shock experiment to verify the beneficial effects of warm prestressing.

e. Task V: ITV Testing

Plan, analyze, and test up to ten small model experiments related to the FY 78 run-arrest experiment on ITV-7C, and report the results. Qualify a through-the-thickness weld repair with vertical vessel and weld repair region orientation, and perform a weld repair per ASME XI procedures. Plan, analyze, and execute a ductile shelf test, ITV-7B, with a flaw in the HAZ of the repair weld, sized similar to those in ITV-7 and 7A, and at the same temperature. Current thoughts are that this should be a hydraulic test, but with a boot similar to that used in ITV-7A. Report the results. Qualify a partial thickness gouge weld repair with vertical part and gouge orientation, the gouge residing in a stress relieved, submerged ARC weldment. Repair a similar gouge put into the longitudinal weld on ITV-8 with the weld repair done in the same manner. Current thoughts are that the depth of the gouge will be approximately 2/3 of the vessel thickness. The residual stresses must be very carefully evaluated on the qualification piece of prolongation. After analysis before test, plan and execute a transition temperature test, ITV-8, with flaw in the HAZ of the repair weld. Consideration should be given to a similar flaw located in the plate material. Report the results.

f. Task VI: PCR V Tendon Corrosion

Simultaneously test multiple wires under stress conditions (70%UTS) identical to those employed in prestressed concrete structures. Wires will be individually loaded ASTM A421 grade BA wire, and the center wire from a 7 wire strand tendon (ASTM A416 grade 270). Each wire is to be exposed to an environment, at about 70°F, containing a controlled quantity of contaminant amounting to 50, 200, and 500 PPM of nitrate contaminates and 200, 2000, and 20,000 PPM of sulfate contaminates for a period of about one year. Electrochemical potential to be monitored.

The tensioned wires are to periodically be examined by microscopic inspection while under tension. The carrier solution containing the contaminants will be circulated and periodically analyzed so that adjustments can be made to maintain the desired contaminant composition. Report the results.

Based on the results of microscopic examinations, and other tests deemed necessary, recommend allowable maximum limits on nitrates and sulfates in grease, grout or concrete used as corrosion inhibitors on prestressing elements forming part of nuclear containment structure that might be located on land or in an ocean environment.

g. Task VII: Foreign Research

Systematically collect, maintain and review foreign research (papers, results, reports, etc.) applicable to safety of LWR primary systems in the areas of fracture mechanics, metallurgy, welding and structures fabrication. Perform travel to foreign countries as necessary, especially western Europe and Japan and within NRC funding to assist in obtaining and evaluating research data. Travel plans to be approved by NRC in advance. Summarize research and provide up-date in quarterly progress reports. Based on reviews, identify the validity and usefulness of the foreign research to NRC for application to safety and licensing of LWR's. Recommend by letter to WRSR on a timely basis, the application of pertinent, well-founded research results.

3. Experimental Facilities and Program:

a. Facilities: Testing laboratories and facilities, both at Oak Ridge, Tennessee and at subcontractors locations, such as Westinghouse Research and Development Laboratories, and Hanford Engineering Development Laboratory.

b. Program: See Item 2 above.

4. Project Status:

a. Results to date: see Progress Reports for details.

b. Essential results: to provide materials property data and analysis methods to assure the safe operation of nuclear power reactors.

5. Next Steps: Essentially as noted in Item 2, above, with special emphasis on irradiation experiments, thermal shock ductile shelf testing practices, and repair weld evaluation.

6. Relation to other projects: This is part of a coordinated effort of research activities to assure the safety of nuclear power reactors.

7. Reference documents: Quarterly Progress Reports and Topical Reports.

8. Degree of availability: All available through NTIS.

11.2.1. Material properties

PROJECT TITLE : Corrosion Problems in Reactor Materials	CSNI INDEX NUMBER : 11.2.1
SPONSORING COUNTRY : Australia	ORGANISATION : Australian Atomic Energy Commission, Private Mail Bag, Sutherland, NSW, 2232
DATE INITIATED : 1. July 1974 2. July 1972 DATE COMPLETED : Continuing	PROJECT LEADER : Dr. D.H. Bradhurst Materials Division

Description :

Objective: To study aspects of materials corrosion relevant to the long-term safety of reactors, namely stress-corrosion cracking of steels, and the corrosion behaviour of zirconium alloys before and after irradiation.

Proposed Investigations:

- 1) The mechanism of stress-corrosion cracking of nuclear grade steels, including ferritic and stainless grades of interest for pressure vessels, and high strength steels.
- 2) The effect of reactor radiation on the corrosion of zirconium alloys in "vault gas" environment.

OECD NUCLEAR ENERGY AGENCY - 38, Boulevard Suchat, 75016 PARIS
Telex : 030608 AEN NEA - Tel : 524 82-00

31.719

PROJECT TITLE : Deformation and Fracture Behaviour of Reactor Materials	CSNI INDEX NUMBER : 11.2.1
SPONSORING COUNTRY : Australia	ORGANISATION : Australian Atomic Energy Commission, Private Mail Bag, Sutherland, NSW, 2232
DATE INITIATED : July 1970 DATE COMPLETED : Continuing	PROJECT LEADER : Dr. K.U. Snowden Materials Division

Description :

Objective:

To study the effects of temperature, neutron irradiation and stress on zirconium alloys and steels used for structural purposes in reactors.

Proposed Investigations:

The programme consists of systematic studies of the effects of creep and fatigue stressing on various alloys. Particular attention will be given to fracture behaviour and the conditions required to induce unstable crack growth under static and cyclic stress conditions.

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Telex : 630668 AEN NEA - Tel : 524 82-00

14.560

PROJECT TITLE : Fracture Behaviour and Irradiation Embrittlement of Steels	CSNI INDEX NUMBER : 11.2.1
SPONSORING COUNTRY : Australia	ORGANISATION : Australian Atomic Energy Commission, Private Mail Bag, Sutherland, NSW, 2232
DATE INITIATED : 1. January 1969 2. July 1974 DATE COMPLETED : Continuing	PROJECT LEADER : Dr. K. Brown Materials Division

Description :

Objectives:

- 1) To provide information necessary for the eventual application of Australian steels in nuclear reactors.
- 2) To understand the fracture behaviour of steels in relation to performance and safety of primary circuit and core components.

Proposed Investigations:

- 1) The effects of neutron irradiation on the transition temperature of Australian ferritic steels and on the yield parameters of low carbon and structural steels are being studied.
- 2) The relationships between microstructure and fracture behaviour of pressure vessel steel sections will be studied by fracture mechanics and fractographic techniques.

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Telex : 630668 AEN NEA - Tel : 524 82-00

14.560

PROJECT TITLE : Radiation Damage in Reactor Materials	CSNI INDEX NUMBER : 11.2.1
SPONSORING COUNTRY : Australia	ORGANISATION : Australian Atomic Energy Commission, Private Mail Bag Sutherland, NSW, 2232
DATE INITIATED : July 1970 DATE COMPLETED : Continuing	PROJECT LEADER : Dr. P.M. Kelly, Materials Division

Description :

Objective:

To investigate the relationship between structure and properties in reactor materials and the effect of neutron irradiation on this relationship.

Proposed Investigation:

The damage produced by neutron irradiation of zirconium and its alloys has been investigated using transmission electron microscopy. This will be continued to provide a complete picture of the damage process in zirconium. Future work will be concentrated on radiation damage in pressure vessel steels and the effect of irradiation on embrittlement.

Classification: 11.2.1.

Title: Dynamisk brudmekanik	Country: DENMARK
	Sponsor: Res. Establ. Risø
Title: Dynamic Fracture Mechanics Crack propagation in inhomogenous steel	Organization: Res. Establ. Risø
	Scientists: A. Nielsen C. Debel
Initiated date: 1973	Completed date:
Status: Some basic investigations finished	

1. General aim: To develop fracture mechanics approaches to evaluate crack propagation in inhomogenous steel in which the fracture toughness varies along the path of the crack.
2. Particular objectives: Welded connections in steel structures constitute areas within which the fracture toughness varies considerably. Welded connections are furthermore the most likely sites for defects able to cause failure of the steel structure. A particular aim of this project is to evaluate the critical size of a brittle volume as related to the general fracture toughness of the base metal.
3. Experimental facilities and programme: Tensile testing, fast bending and instrumented impact testing machines are available. A mathematical analysis has been made and an experimental investigation has been finished using duplex DCB test pieces of a type comparable to those used by G. Hahn, et.al., Battelle, Columbus, USA.
4. Project status: Danish reports on the above mentioned activities have been worked out and English versions are under way. More experiments are needed before significant conclusions can be drawn.
5. Next steps: Experiments to be continued.
6. Relation with other projects: Experiments will be designed to be carried out in the facility resulting from the project Pressure Testing Facility.
7. Reference documents: RISØ-M-1842. Carsten Engel: "Energetiske og kinetiske overvejelser vedrørende revners bevægelse over fasegrænser". RISØ-M-1897. C.P. Debel: "Dynamisk Brudmekanik. Licentiat-rapport".
8. Degree of availability: No limitations.

Titre Comportement des organes d'isolement du circuit primaire des réacteurs à eau.	Pays : FRANCE
Titre (anglais) Behaviour of isolation valves of PWR primary system	Organisme directeur : CEA/DSN
Date de démarrage : 1/01/77 Date prévue d'achèvement : 31/12/81 Etat actuel : Démarrage Dernière mise à jour : 01/01/77	Organisme exécuteur : CEA.DEMT Responsable : B.BARRACHIN (DSN-SETSSR) R. ROCHE (DEMT) Scientifiques :

Objectif général :

Cette étude a pour but de préciser le comportement des organes d'isolement du circuit primaire des réacteurs à eau.

Objectifs particuliers :

L'étude devra essentiellement permettre d'apprécier la validité des arguments présentés quant au bon fonctionnement de ces organes en condition normale et en condition accidentelle.

Prochaines étapes :

Essais préliminaires d'un couple siège-corps (fin 77)
Calcul dynamique-plastique du comportement (fin 78)
Essais complémentaires (1980)
Etablissement de règles générales (1981)

Titre Comportement des lignes de tuyauteries lors d'un accident .	Pays : FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Behaviour of pipings in the event of an accident	Organisme exécuteur : CEA/DEMT
	Responsable : B. BARRACHIN (DSN- SETSSR) R. ROCHE (DEMT)
Date de démarrage : 1/03/77 Date prévue d'achèvement : 31/12/81 Etat actuel : Démarrage Dernière mise à jour : 1/4/77	Scientifiques : B. VRILLON

Objectif général :

Le programme a pour but d'étudier le comportement des tuyauteries en cas de rupture (eau pressurisée). Il concerne essentiellement les tuyauteries de petit diamètre appartenant aux circuits auxiliaires : RCV, RIS, ANG, ASG. Les essais sont réalisés dans des conditions proches de l'échelle réelle.

Objectifs particuliers :

Le programme comporte essentiellement, outre la mise au point du dispositif expérimental :

- La détermination des critères de fouettement.
- La mesure des efforts de réaction.
- La prise en compte des interactions possibles.
- Le dimensionnement des supportages et autres limiteurs de débattement.
- La tenue des organes d'isolement.

Installations expérimentales et programme :

Boucle HUREPOIX (CEN-SACLAY).

Prochaines étapes :

Mise au point de la boucle et premiers essais (fin 77)
Détermination des critères (fin 78)
Dimensionnement des supportages et comportement des organes d'isolement (fin 79).

Titre Dommages consécutifs aux défauts des structures et/ou évolutions de chargement.	Pays : FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Damages subsequent to structure defects and/or load changes	Organisme exécuteur : CEA/DEMT
	Responsable : B. BARRACHIN (DSN - SETSSR) R. ROCHE (DEMT)
Date de démarrage : 1/01/76 Etat actuel : en cours	Date prévue d'achèvement : 31/12/80 Dernière mise à jour : 01/01/77
Scientifiques :	

Objectif général :

Cette étude a pour but de vérifier les lois de comportement en fatigue et fissuration des matériaux utilisés dans les circuits primaires des réacteurs à eau PWR.

Objectifs particuliers :

- Influence du milieu
- Influence de la biaxialité
- Comportement sous chocs thermiques

Etat de l'étude :

1) Avancement à ce jour :

Les essais effectués ont permis de préciser le comportement en fatigue des aciers pour cuve (acier Creusot-Loire type A 508 classe 3) et des alliages austénitiques en milieu PWR.

2) Résultats essentiels :

- Pour l'acier type A 508 classe 3, la courbe de fatigue du Code ASME section III est acceptable.
- Pour les aciers austénitiques, la courbe de fatigue du Code ASME section III est trop optimiste et devrait être réajustée.

.../...

Prochaines étapes :

- Fatigue sous sollicitation biaxiale (fin 77)
- Essais sur récipients : corrélation avec les essais sur éprouvettes (1980)

Documents de référence :

"Low Cycle Fatigue of Austenitic Alloys in Hot Water"
C.GARNIER, R. ROCHE, B.BARRACHIN.
Smirt L 6/7 Londres 1975

"Low Cycle Fatigue of Steels for Nuclear Pressure Vessels in Hot Water"
C.GARNIER, G.KOWALCZUK, R. ROCHE, B.BARRACHIN.
Smirt L 8/h San Francisco 15 - 19 août 1977.

131-11-31/4111-66

<p>Titre</p> <p>Propagation de fissures de fatigue sur les matériaux de structure irradiés.</p>	<p>Pays : FRANCE</p> <p>Organisme directeur : CEA/DSN</p>
<p>Titre (anglais)</p> <p>Fatigue crack propagation on irradiated materials.</p>	<p>Organisme exécuteur : CEA/DTECH</p> <p>Responsables : B. BARRACHIN (DSN-SETSSR) P. PETREQUIN (DTECH-SERMA)</p>
<p>Date de démarrage : 1/01/76 Date prévue d'achèvement : 31/12/79 Etat actuel : en cours Dernière mise à jour : 1/1/77</p>	<p>Scientifiques :</p>

Objectif général :

Le programme a pour but de déterminer les lois de propagation des fissures de fatigue sur les matériaux de structure irradiés.

Objectifs particuliers :

Les essais sont plus spécialement orientés sur les aciers pour cuves de réacteurs à eau sous pression PWR.

Etat de l'étude :

1) Avancement à ce jour :

La machine de fatigue a été montée en cellule chaude et est actuellement opérationnelle. Par ailleurs les essais d'irradiation sont en cours.

Prochaines étapes :

- Essais sur matériaux de cuve :
- à l'ambiante (fin 77)
 - en température élevée (fin 78)

Titre Comportement des aciers pour cuves - tenue à l'irradiation	Pays : FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Research programme on irradiation embrittlement of pressure vessel steels.	Organisme exécuteur : CEA/DTECH
	Responsable : B. BARRACHIN (DSN- SETSSR)
Date de démarrage : 01/01/76 Etat actuel : en cours	Date prévue d'achèvement : 31/12/79 Dernière mise à jour : 30/11/76
Scientifiques : P. PETREQUIN (DTECH - SRMA)	

Objectif général :

Le programme a pour but de déterminer le comportement sous irradiation neutronique de l'acier Creusot-Loire (type A 508 classe 3) utilisé pour la fabrication des cuves PWR du programme nucléaire français.

Objectifs particuliers :

Le programme comporte essentiellement :

- 1) Les mesures de KCV et NDT,
- 2) Les mesures de KIC sur grosses éprouvettes,
- 3) L'étude de la restauration,
- 4) Les mesures de KID et JIC sur petites éprouvettes,
- 5) La participation au programme AIEA.

Installations expérimentales et programme :

Réacteur expérimental TRITON (CEN-Fontenay-aux-Roses).

Etat de l'étude :

- 1) Avancement à ce jour :

Le point 1 est terminé. Les points 2,3,4 sont en cours. Le point 5 est prévu pour 1977.

- 2) Résultats essentiels :

- 1) Evolution des caractéristiques de fragilité sous irradiation. Les essais ont porté sur des échantillons prélevés dans les viroles de coeur des 5 premières cuves du programme français. Les irradiations ont été menées à 290 degrés C et flux intégré de 4×10^{19} NVT. Le déplacement de la température de transition est faible pour le métal de base et plus important pour les soudures.

.../...

- 2) Les mesures de KIC ont été effectuées sur matériaux non irradiés. Un dispositif pour l'irradiation de grosses éprouvettes est en cours de réalisation.
- 3) Les premiers essais ont permis de préciser la cinétique de restauration et d'examiner les équivalences temps-température.

Prochaines étapes :

- 1) Mesure de KIC sur grosses éprouvettes irradiées.
- 2) Vérification des équivalences temps - température.
- 3) Détermination des courbes de transition de KID. Définition de la méthode et mesures de JIC.
- 4) Définition du programme AIEA.

<u>Classification:</u> 11.2.1				
<u>Title 1 (Original Language):</u> Bestimmung bruchmechanischer Sicherheitskriterien für elastisch-plastisches Werkstoffverhalten (RS 90 A-II.2, Jahresbericht A 76)	COUNTRY: BRD			
	SPONSOR: BMET			
	ORGANIZATION: KWU, Erlangen AEG HE/IMU Ffm.			
<u>Title 2 (English):</u> Determination of Fracture Mechanical Safety Criteria for Elastic-Plastic Behaviour of Materials	<u>Project Leader:</u> Dr. Klausnitzer Prof. Detert			
	<table border="0"> <tr> <td><u>Initiated (Date):</u> 1. 1. 76</td> <td><u>Completed (Date):</u> 31. 12. 76</td> </tr> <tr> <td><u>Status:</u> Completed</td> <td><u>Last Updating (Date):</u> 31. 12. 76</td> </tr> </table>	<u>Initiated (Date):</u> 1. 1. 76	<u>Completed (Date):</u> 31. 12. 76	<u>Status:</u> Completed
<u>Initiated (Date):</u> 1. 1. 76	<u>Completed (Date):</u> 31. 12. 76			
<u>Status:</u> Completed	<u>Last Updating (Date):</u> 31. 12. 76			

1. General Aim

For the collection of safety criteria with respect to the fracture due to elastic-plastic material behaviour the steel 22 NiMoCr 37 was investigated by the methods of fracture mechanics.

2. Particular Objectives

With the COD-concept, the J-Integral concept and the concept of resistance curves the methods of the linear elastic fracture mechanics were tried to be extended on the elastic-plastic material behaviour.

3. Research Program and 4. Test Facilities

In continuation of the research program RS 90 tests were carried out with big plates which were loaded until fracture. The experimental data were compared with analytical calculations in order to refine the calculation methods.

5. Progress to Date

The plates with a hole in the center got a mechanical crack, which was prolonged by oscillation technique. When the plates were instrumented the tests began at room temperature. Strain, COD, acoustic emission and forces were measured.

The results were evaluated.

6. Results

The evaluation of the results is still under work. The determination of the fracture force by the concept of resistance curves has shown good agreement with the results of the tests on small specimens.

7. Next Steps

Work has been completed.

8. Relation with Other Projects

RS 90 Determination of Fracture Mechanical Safety Criteria
 for Elastic-Plastic Behaviour of Materials

9. References

10. Degree of Availability

<u>Classification:</u> 11.2.1	
<u>Title 1 (Original Language):</u> Schweißversuche zum Plattieren von Reaktor-Druckgefäßen (RS 91 - II.2, Jahresbericht A 76)	<u>COUNTRY:</u> BRD
	<u>SPONSOR:</u> BMFT
	<u>ORGANIZATION:</u> KWU, Erlangen
<u>Title 2 (English):</u> Welding Tests on Cladding of Reactor Pressure Vessels	<u>Project Leader:</u> Dr. Klausnitzer Prof. Dr. Detert
<u>Initiated (Date):</u> 1. 12. 72 <u>Status:</u> Continuing	<u>Completed (Date):</u> 31. 7. 77 <u>Last Updating (Date):</u> 31. 12. 76

1. General Aim

It is the aim of the present work to examine the influence exerted by different parameters on residual stresses using a suitable measurement method.

2. Particular Objectives

In order to protect the inner surface of reactor pressure vessels against corrosion, band or strip welding procedure is used for cladding the ferritic vessel with austenitic stainless steel. During cladding high residual stresses occur in thick-walled components. In addition, the different thermal expansion coefficients of stainless steel and ferritic steel induce further residual stresses. These stresses cause under-clad-cracks during postweld heat treatment. Therefore, the knowledge of the magnitude and of the distribution of residual stresses after welding and after postweld heat treatment are of special interest.

3. Research Program and 4. Experimental Facilities

Cladding was made on A 508. The dimension of the plates was about 125 mm thick, 240 mm wide and 300 mm long. The thickness of the austenitic cladding was about 5 mm. The following measurement method was used: strain gauges were placed on the surface of the

ferritic side of the plate and the cladded, opposite side was machined in steps. The strain response to the removal of the material was measured using strain gauges. These strains are used for compensating the stress distribution in cladding direction of the plate in dependance of the plate thickness.

5. Progress to Date

On a specimen with the bore hole method residual stresses were measured. The weld material, heat influence zone and basic material were studied. The instrumentation was specially arranged in order to measure the stress parallel and vertical to the connection weld.

The bore hole method was extended from 5 mm depth to 25 mm. Several methods were tested to get a smooth calibration function.

6. Results

First attempts were made to approximate the calibration curves of the extended bore hole method with analytic functions of second or third order. There were inconstancies however from one kernel to another. Therefore Fourier-approximations will be used.

7. Next Steps

The Fourier-method will be tested on austenitic cladding materials.

8. Relation with Other Projects

RS 101 Reactor Pressure Vessel Urgent Program 22 NiMoCr 37

9. References

10. Degree of Availability

<u>Classification:</u> 11.2.1	
<u>Title 1 (Original Language):</u> "Forschungsprogramm Reaktordruckbehälter" - Dringlichkeitsprogramm 22 NiMoCr 37 - (RS 101, RS 101A - II.2, Jahresbericht A 76)	<u>COUNTRY:</u> BRD
	<u>SPONSOR:</u> BMFT
	<u>ORGANIZATION:</u> MPA Stuttgart
<u>Title 2 (English):</u> Reactor Pressure Vessel - High Priority Program 22 NiMoCr 3 7 - (RS 101, RS 101A, Annual Report A 76)	<u>Project Leader:</u> Prof. Dr.-Ing. E. Krägeloh
<u>Initiated (Date):</u> December 1, 1972	<u>Completed (Date):</u> December 31, 1977
<u>Status:</u> partly completed	<u>Last Updating (Date):</u> December 31, 1976
	Prof. Dr.-Ing. H. Uetz

1. GENERAL AIM

Determination of possible critical conditions in welding seams (strength seams) and their effect on strength or safety against fracture with the objective of optimizing the material choice and the parameters during welding and heat treatment (see also Annual Report IRS-F-18).

2. PARTICULAR OBJECTIVES

Thorough metallographic as well as electron microscopic investigations have confirmed that intercrystalline micro cracks can develop in the coarse grain zone of weldings (strength seams) in the material 22 NiMoCr 37 (A 508 Cl 2) during heat treatment after welding - especially during stress-relief heat treatment. While the reason for crack formation can be explained by the relaxation embrittlement in the range of the stress-relief heat treatment temperature, the question is, how much such coarse grain zones can influence the strength and toughness of the structural component. The Urgent Program 22NiMoCr 37 is supposed to compile the necessary documents for a safety analysis for reactor pressure vessels in accordance with the specific material crack conditions.

As a way towards solutions parameter studies have been considered, which can be divided in tests with small and large specimens as well as "component tests". In

addition to the conventional simulated welding treatments, which can only be carried out on small specimens, simulated tests on large specimens have been made. Electron beam welding has been used for the creation of geometrically defined HAZs.

Parallel to the experimental tests it is tried to determine temperature fields and heat stresses theoretically.

3. RESEARCH PROGRAM

The research program has been classified in the following nine partial projects:

- a) intermediate vessels (ZB 1)
- b) welding of discs into reactor pressure vessel ring (S-UP-Ro)
- c) simulated specimens - tests on the welding simulator (Si-SSi)
- d) simulated specimens - annealing heat treatment and relaxation tests (Si-G/R)
- e) simulation with electron beam (Si-E-beam)
- f) submerged-arc-welding, circumferential seam (S-UP-RN)
- g) submerged-arc-welding, longitudinal seam (S-UP-LN)
- h) deformation and fracture mechanism (VBM)
- i) device for the testing of large specimens (10 000 Mp Project)

4. EXPERIMENTAL FACILITIES, COMPUTER CODES

See Annual Report IRS-F-24.

5. PROGRESS TO DATE

The projects ZB 1, S-UP-Ro, Si-SSi, Si-G/R, Si-E beam, S-UP-RN, S-UP-LN, and VBM have been completed by composing the third Technical Report.

10 000 Mp Project

Completion of further small parts, construction and production of auxiliary devices and facilities for the machine, planning of the installation. Construction works for the testing hall (RS 101A) have been started.

6. RESULTS

See Annual Report A 75.

7. NEXT STEPS

10 000 Mp Project:

Completion of the foundation. Mounting of the clamping device.

8. RELATION WITH OTHER PROJECTS

See Annual Reports IRS-F-18

9. REFERENCES

Quarterly Reports in IRS Research Reports, period reported on:

January 76 - December 76 (German)

Annual Report A 76 (English).

10. DEGREE OF AVAILABILITY OF THE REPORTS

Upon request at BMFT.

<u>Classification: 11.2.1</u>	
<u>Title 1 (Original Language):</u> "Bestrahlung von Druckgefäßstählen im Rahmen des Koordinierten Programmes der IAEA" (RS 85 - II.2, Jahresbericht A76)	COUNTRY: BRD SPONSOR: IAEA ORGANIZATION: GKSS, Geesthacht
<u>Title 2 (English):</u> Coordinated Programme on Irradiation Embrittlement of Pressure Vessel Steel	<u>Project Leader:</u> Dr.-Ing. F. J. Schmitt
<u>Initiated (Date):</u> April 1973 <u>Status:</u> finished	<u>Completed (Date):</u> 1976 <u>Last Updating (Date):</u> December 1976

1. General Aim

The purpose of the programme is to promote coordinated research in the field of irradiation embrittlement of reactor pressure vessel steels, at an international level. The results obtained will give a possibility for a more precise interpretation of individual surveillance results.

2. Particular Objectives

Eight institutes in seven nations are participating in this programme whose basic goal is to establish that the bases for describing embrittlement, measurement of the neutron spectrum and fluence on mechanical properties (on a typical-standard reference-plate of steel) were sufficiently standardized to permit direct intercomparison among international programmes and to compare embrittlement sensitivity of national steels with the standard steel.

3. Research programme

Samples of the pressure vessel steel ASTM A533 B Class 1 (Plate 03 of the HSST-Programme) were irradiated and different materials tests were carried out, to study the influence of neutron dosis and irradiation temperature on the mechanical properties.

4. Experimental Facilities

Further Charpy-V-notch and tensile specimens were neutron irradiated in several cap-

sules in the Material Test Reactor FRG-2 in Geesthacht:

- a) Irradiation of Cv- and tensile specimens at a temperature of 290 °C to a dose of $4,0 \cdot 10^{18}$ nvt, $E > 1\text{MeV}$.
- b) Irradiation of Cv- and tensile specimens at temperatures between 40 °C and 75 °C to doses of $2,9 \cdot 10^{18}$; $1,6 \cdot 10^{19}$ and $1,0 \cdot 10^{20}$ nvt.
- c) Irradiation of fracture mechanics specimens WOL-1X at 290 °C and to doses of $2,2 \cdot 10^{19}$ and $8,5 \cdot 10^{19}$ nvt.

The mechanical tests were conducted in the Hot Cells of the GKSS.

5. Progress to Date

All specimens have been irradiated, some were postirradiation annealed and the following mechanical tests were carried out in the Hot Laboratory:

- Impact test at the whole transition region and determination of the transition-temperature, shear fracture, lateral expansion.
- Tensile test (tensile strength, yield strength, elongation, reduction of area.)
- Fracture toughness test.

6. Results

6.1 Irradiations at 290 °C and $4,0 \cdot 10^{18}$ nvt

Impact tests

The shift of the NDT-Temperature for transverse specimens is

$\Delta \text{NDTT} = 22 \text{ } ^\circ\text{C}$ (41 J - criterium)

" = $20 \text{ } ^\circ\text{C}$ (50 % fibrous fracture)

The lateral expansion of 0,9 mm was reached at 50 °C.

The shift of the NDTT for longitudinal specimens is $\Delta \text{NDTT} = 20 \text{ } ^\circ\text{C}$ (41 J)

" = $8 \text{ } ^\circ\text{C}$ (50 % fibrous

fracture).

The lateral expansion of 0,9 mm was reached at 34 °C.

Tensile tests

Increase of the tensile strength at room temperature from 617 N/mm^2 to 632 N/mm^2 .

Increase of the yield strength from 458 N/mm^2 to 489 N/mm^2 .

6.2 Irradiations between 40 °C and 75 °C

- a) Dose of $2,9 \cdot 10^{18}$ nvt

Impact tests

Δ NDTT = 60 °C (transverse specimen)
" = 68 °C (longitudinal ") 41 J.
 Δ NDTT = 40 °C (transverse specimen)
" = 31 °C (longitudinal ") 50 % fibrous fracture

Tensile tests

Increase of σ_{UT} from 617 N/mm² to 636 N/mm²
Increase of σ_y from 458 " to 598 "
b) Dose of 1,6 . 10¹⁹ nvt

Impact tests

Δ NDTT = 113 °C (transverse specimen)
" = 140 °C (longitudinal ") 41 J.
 Δ NDTT = 82 °C (transverse specimen)
" = 106 °C (longitudinal ") 50 % fibrous fracture

Tensile tests

Increase of σ_{UT} from 617 N/mm² to 766 N/mm²
Increase of σ_y from 458 N/mm² to 766 N/mm²
c) Dose of 1,0 . 10²⁰ nvt

Impact tests

Δ NDTT = 182 °C (transverse specimen)
" = 172 °C (longitudinal ") 41 J.
 Δ NDTT = 155 °C (transverse specimen)
" = 132 °C (longitudinal ") 50 % fibrous fracture

Tensile tests

Increase of σ_{UT} from 617 N/mm² to 912 N/mm²
Increase of σ_y from 458 " to 912 "

6.3 Irradiation of fracture mechanics specimens

Type: WOL-1X

Irradiation temperature: 290 °C

" doses : 2,2 . 10¹⁹ and 8,5 . 10¹⁹ nvt

The temperature shift $\Delta FMTT$ reaches values of 54 °C and 70 °C for the two neutron doses

The experimental work has been finished and the results will be evaluated.

7. Next_steps

A final report will be prepared.

8. Relation_with_other_projects

See annual reports 1974 and 1975

9. References

- a) Quarterly reports in the series IRS-Forschungsberichte RS85, (in German),
- b) F.J. Schmitt, Zeitschrift für Werkstofftechnik, 7, (1976), 241-247 (in German)

10. Availability_of_the_Reports

- a) is available from the GRS, Cologne, Germany
- b) is available from the Verlag Chemie, GmbH.
D-6940 Weinheim, Germany.

<u>Classification:</u> 11.2.1	
<u>Title 1 (Original Language):</u> Untersuchungen zum Verschweißen sowie an Schweißverbindungen dickwandiger Reaktorbauteile (RS 102-01 - II.2, Jahresbericht A 76)	<u>COUNTRY:</u> BRD Germany
	<u>SPONSOR:</u> BMFT
	<u>ORGANIZATION:</u> FhG - IFAM
<u>Title 2 (English):</u> Narrow-Gap welding	<u>Project Leader:</u> I.D.Henderson
<u>Initiated (Date):</u> 1.5.73	<u>Completed (Date):</u> 31.12.76
<u>Status:</u> Finished	<u>Last Updating (Date):</u> December 1976

1. General Aim

The research program is aimed at evaluating the gas-shielded arc-welding process of Narrow-Gap welding for joining of heavy sections of the fine-grained low-alloyed pressure vessel steel 22 NiMoCr 3 7. The investigations are primarily designed to obtain information regarding the applicability of this newly developed welding process for one-sided welding of thick sections up to 300 mm, and specifically to determine the mechanical and metallurgical properties of welded joints produced in forged sections of the nominated reactor construction steel.

2. Particular Objectives

Until the conception of this research project, a welding machine of this type and a welding process of this nature had not been investigated in Europe. To begin the research program it was therefore necessary to set up a laboratory facility and to install the equipment necessary to conduct such an investigation. Following this preparatory work, the welding tests could begin.

Using a range of welding consumables and various welding conditions, preliminary investigations were carried out on 100 mm rolled sections

to establish the optimum parameters and electrode compositions for the main tests. Welds were then made in 200 and 300 mm forgings, whereby emphasis was not only placed on the weldability and functionability of the machine but also on the comparison of the mechanical and metallurgical properties of the welded joints in various heat-treated conditions with those properties established for the base material. The welding tests were conducted mainly in the flat position. As a result of increasing interest in the site-welding of heavy sections in the horizontal-vertical position, a preliminary investigation was also done in the three-o'clock position on 50 mm rolled plate. Other special investigations include the measurement of the temperature distribution and thermal cycles in the heat affected zone and welding with non-metallic backing strips to produce a suitable root geometry for supplementary cladding processes.

3. Research Program

The research program was initially drawn up to include welding tests on the steel 20 MnMoNi 5 5 as well as the steel 22 NiMoCr 3 7. As a result of difficulties in procuring the steel 20 MnMoNi 5 5 in small quantities and in sizes required for these welding tests, the investigations on this base material could not be carried out as part of this research program. It was also planned to produce welded joints using other welding processes such as electro-slag, submerged-arc and MMA-processes for the purpose of comparing the results.

However, after considering the state-of-the-art and with the wealth of knowledge which already existed with respect to the conventionally applied welding processes, it was decided to restrict this research program to welding tests using only the Narrow-Gap process and to concentrate on the determination of the influence of the welding conditions on the mechanical and metallurgical properties of the joints. Of particular interest was the effect of preheating and post-weld heat treatment on the weld metal and HAZ properties.

4. Experimental Facilities

The Narrow-Gap welding machine was originally conceived to permit one-

sided welding of section thicknesses up to 230 mm. Following the successful completion of the welding tests on 200 mm thick forged sections, modifications were made to the welding torch assemblies and to the work holding system to permit the tests on heavier sections to be done. Fig. 1 shows the welding head arranged for welding of 300 mm thick forgings.

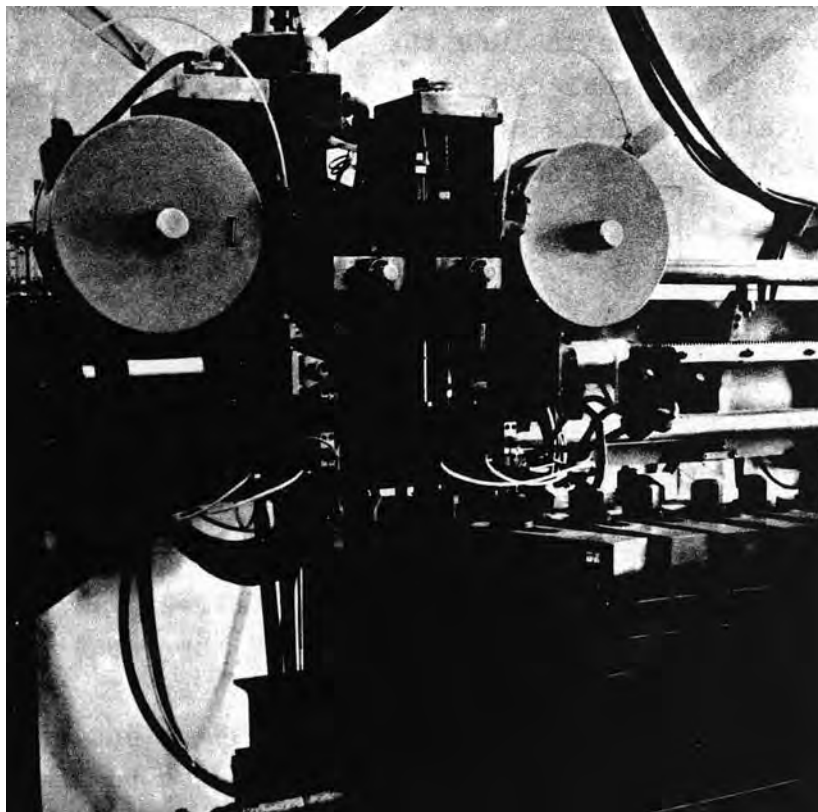


Fig. 1: General view of the Narrow-Gap welding head and 300 mm thick test pieces prepared for welding.

These tests were done using a preheating temperature of 200 to 250 °C. The welded joints were then non-destructively and mechanically tested in both the as-welded and in the stress-relief heat-treated conditions.

Further modification of the machine and the holding system enabled orientation welding tests on 50 mm thick plate in the horizontal-

vertical position to be carried out. Fig. 2 shows the arrangement of the test facility for these welding investigations.

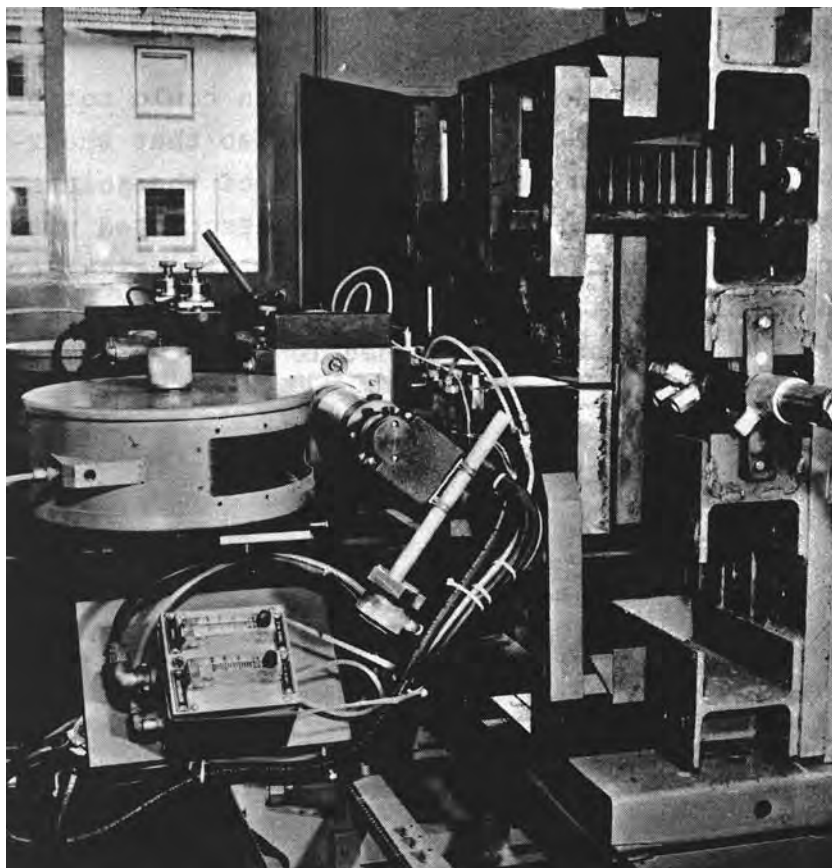


Fig. 2: Narrow-Gap welding of 50 mm test pieces of the steel 22 NiMoCr 3 7 in the three-o'clock position

5. Progress to Date

The extension of the research program period by six months and an increase in the funds by approximately 200 TDM allowed the planned research work to be successfully completed on 31.12.76. The extensive testing of welded joints produced in 200 mm forged sections of the steel showed that in addition to the preheating of the weldments to 200 to 250 °C, a stress-relief heat-treatment at 600 °C is necessary to achieve optimum fracture toughness in the weld metal. The fact

that in the as-welded condition the Narrow-Gap weld metal exhibits comparable mechanical properties to the base material in the forged condition meant that this test condition could not be excluded for the welding tests on 300 mm sections.

6. Results

The non-destructive testing of these joints could not be conducted successfully on full-thickness specimens, so that the x-ray examinations were carried out on smaller segments of the joints. These tests revealed the usual low level of porosity associated with gas shielded arc welding. The mechanical testing and metallurgical examination of the joints simply duplicated the test results established on 200 mm test pieces. Fig. 3 shows a cross-section of a weld seam produced in

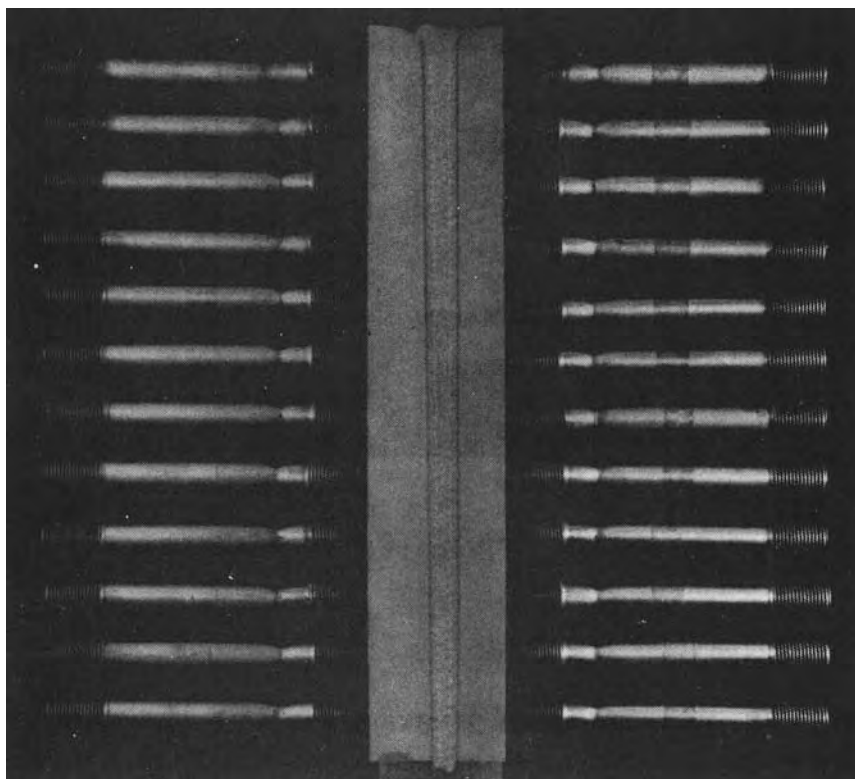


Fig. 3: Section of a Narrow-Gap weld seam in 300 mm thick forgings of the steel 22 NiMoCr 3 7 with tested round tensile specimens in the as-welded (LHS) and stress-relief heat-treated conditions (RHS)

300 mm thick sections. As a result of an excessive allowance for shrinkage, the weld seam is not uniformly parallel but despite the gap width of 12 mm near the top of the weld, a satisfactory seam geometry with adequate side-wall and centre-line fusion resulted. The tested tensile specimens from the section thickness reveal that in both the as-welded and the stress-relief heat-treated condition, the welded zone is not the critical region with respect to tensile strength. Tensile testing of mini-tensile specimens extracted from the small heat affected zone revealed tensile strengths (0,2-proof stresses) in the HAZ of approximately 660 N/mm^2 in the as-welded condition and between 474 and 620 N/mm^2 in the heat-treated condition. The base material 22 NiMoCr 3 7 has in comparison a yield strength of 478 to 550 N/mm^2 at room temperature.

In Fig. 4 is a cross-section of a Narrow-Gap weld seam produced in the horizontal-vertical position.

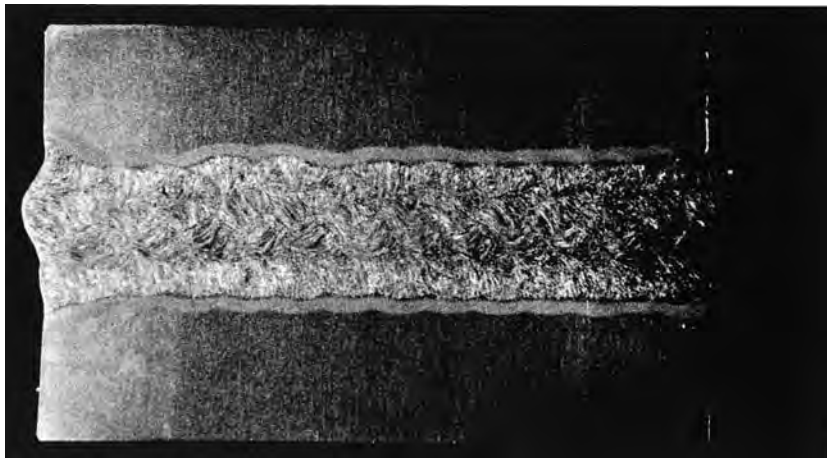


Fig. 4: Section of a Narrow-Gap weld in 50 mm thick plate produced in the three-o'clock position

7. Next Steps

Considering the good results obtained during the Narrow-Gap welding tests on section thicknesses from 50 to 300 mm using a one-sided welding technique it is to be expected that this work will be continued. How-

ever, the financing of a continuation project must be met jointly by the industry as well as governmental authorities such as the BMFT.

Following discussions with several representatives of companies involved in pressure vessel construction, the main interest seems to be directed to the evaluation of the Narrow-Gap welding process for welding of girth seams in heavy sections of the pressure vessel steel 20 MnMoNi 5 5 in the three-o'clock position. These investigations would simulate the present welding problem arising as a result of on-site construction and assembly of large pressure vessels for both pressurized water reactors and boiling water reactors.

8. Relation with Other Projects

Associated with the Reactor Safety Program there have been no additional projects approved which relate to Narrow-Gap welding. However, several developments related to this subject are worth reporting:-

- a. During a visit to Japan in September 1976 it was shown that considerable emphasis is being placed on the development and application of narrow gap welding techniques for thick section welding.
- b. In Italy, a major construction firm is developing a narrow gap welding process for joining of heavy section austenitic and ferritic steels for nuclear applications.
- c. Following a FhG-Seminar in IFAM on 14.6.76 several firms in Germany have shown interest in supporting a continuation project to further evaluate the Narrow-Gap welding technique.
- d. The committee of the DVS-Working Group UG 30-1 is presently considering ways and means of promoting this research work.

9. References

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10. Degree of Availability of the Reports

All reports about this research project are unclassified.

<u>Classification: 11.2.1</u>	
<u>Title 1 (Original Language):</u> Bruchverhalten von teilweise durchgehenden Rissen (RS 102-11 -II.2, Jahresbericht A 76)	<u>COUNTRY:</u> BRD
	<u>SPONSOR:</u> BMFT
	<u>ORGANIZATION:</u> IFKM Freiburg
<u>Title 2 (English):</u> Fracture Properties of Part-Through Cracks	<u>Project Leader:</u> Dr. E. Sommer Dipl-Ing.H.Kordisch
<u>Initiated (Date):</u> May 1973	<u>Completed (Date):</u> Oct. 30, 1976
<u>Status:</u> Finished	<u>Last Updating (Date):</u> December 1976

1.- General Aim

The general aim is to study the fracture properties of part-through cracks in nuclear reactor components with experimental methods, based on the concepts of fracture mechanics.

2.- Particular Objectives

The extension of part-through cracks in thick-walled plates and tubes under fatigue loading has been investigated. Growth characteristics have been determined by analysing the fracture surfaces. The crack extension in the direction of the semiaxes as well as the crack growth along the whole front has been evaluated and interpreted in terms of the parameters: initial crack geometry, boundary conditions and type of loading.

3.- Research Program

To show the propagation characteristics, artificial markings were induced on the fracture surfaces during a fatigue test by an overload technique. In order to establish the test conditions, a calibration curve da/dN versus ΔK measured on compact specimens is used.

4.- Experimental Facilities, Computer Codes

The fatigue tests were performed on a 1.6 MN servohydraulic machine with an

internal pressure device giving up to 150 N/mm^2 and the calibration tests on a 0.25 MN servohydraulic machine.

5.- Progress to Date

Growth characteristics have been determined for the extension of part-through cracks under fatigue loading. The part-through cracks were located in plates and tubes of the reactor steel 22 Ni Mo Cr 3 7 (Fig.1). Markings on the fracture surfaces introduced by an overload technique were used to locate the crack front at specified time intervals and hence to evaluate the crack growth. The loading was confined to mode-I and was applied by either simple tension or internal pressure.

6.- Results

A careful evaluation of these fracture surfaces shows the following results (Fig. 2):

1. For small crack depths (a), i.e. ($a/d < 0.5$) crack growth in thickness (d) direction predominates over crack growth in the width direction - as expected from analytical calculations.
2. For large crack depths ($0.6 < a/d < 1$) the converse behaviour occurs - in contradiction to the analysis of the investigated case.

When the relative growth $\Delta L / \Delta a - \Delta L$ being the crack growth at the polar angle φ of a crack front; Δa the crack growth at $\varphi = 90^\circ$ - is plotted versus the polar angle φ , it can be demonstrated that the maximum crack growth is shifted from the centre of the crack front at $\varphi = 90^\circ$ to smaller angles as the crack depth increases (Fig. 3,4).

In order to explain these results a simple model will be discussed. This takes into account the variable crack resistance along the crack front which arises due to changes in the stress state. The interaction between stress state and material response is again affected by geometry as well as by the degree and nature of loading, the loading rate and temperature.

7.- Next Steps

The project is completed . Final report is forthcoming.

8.- Relation with other projects

This project is closely related to RS 102-12, RS 102-13 and RS 90.

9.- References

- Annual Report A 75 (Series IRS Forschungsberichte)
- E.Sommer, L. Hodulak, H. Kordisch: Growth Characteristics of Part-Through Cracks in Thick-Walled Plates and Tubes.
ASME Paper No. 76-PVP-39

10.- Degree of Availability of the Reports

The reports can be ordered from

Gesellschaft für Reaktorsicherheit GRS
5000 Köln - 1, Glockengasse 2

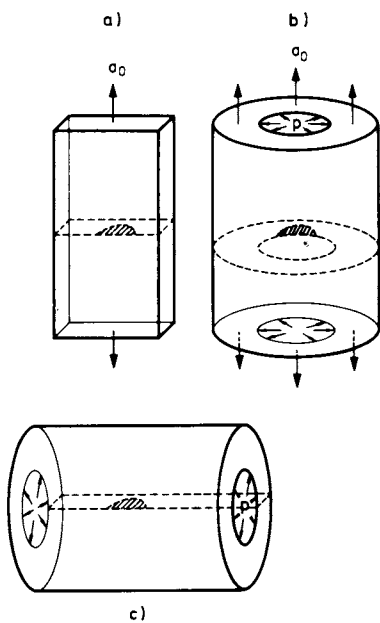


Fig. 1 :

Specimen configurations
a) Part-through crack in a plate;
b) Circumferential crack in a tube
c) Axial crack in a tube

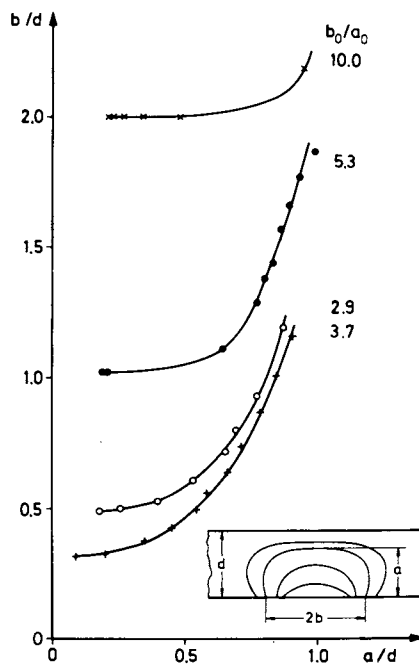


Fig. 2:

Normalized crack extensions in the direction of width b and depth a for different starter notches

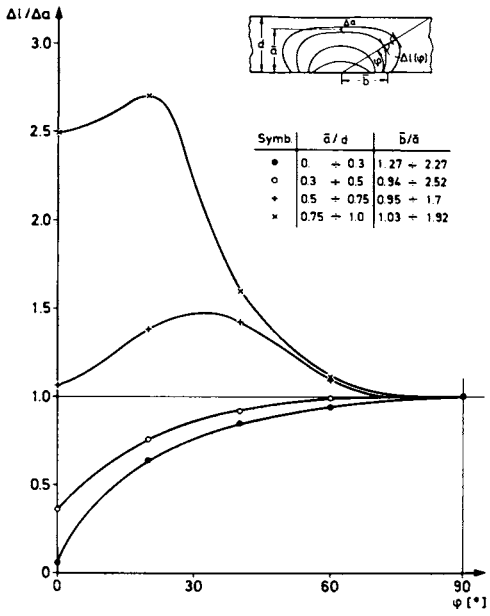


Fig. 3:
Relative crack growth $\Delta l/\Delta a$ versus
polar angle φ

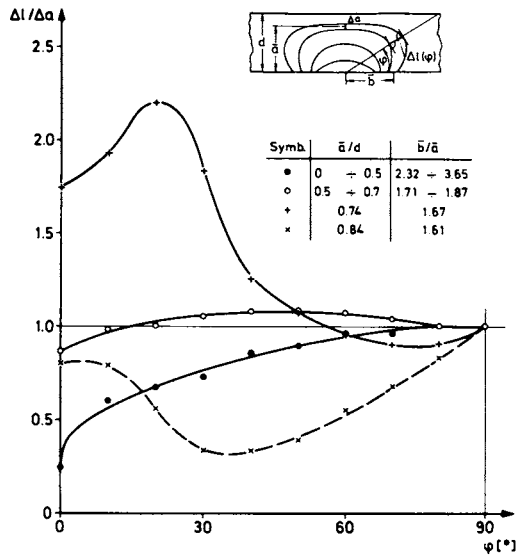


Fig. 4:
Relative crack growth $\Delta l/\Delta a$ versus
the polar angle φ for an axial crack
in a tube

<u>Classification: 11.2.1</u>	
<u>Title 1 (Original Language):</u> Untersuchungen zum Rißersatz und Arrest (RS 102-12 - II.2, Jahresbericht A 76)	<u>COUNTRY:</u> BRD
	<u>SPONSOR:</u> BMFT
	<u>ORGANIZATION:</u> IFKM Freiburg
<u>Title 2 (English):</u> Investigation of Crack Initiation and Arrest	<u>Project Leader:</u> Dr. J.F. Kalthoff
<u>Initiated (Date):</u> 1.5.1973 <u>Status:</u> Completed	<u>Completed (Date):</u> 30.9.1976 <u>Last Updating (Date):</u> December 1976

1. General Aim

The general aim in connection with the reactor safety program is to improve the knowledge of the fracture behaviour of reactor components in which cracks or cracklike defects have to be assumed.

2. Particular Objective.

The particular objective is the investigation of crack initiation and crack arrest phenomena. Stress analytical and materialspecific aspects are considered. The tools of analysis are from linear elastic fracture mechanics.

3. Research Program

Cracks are initiated and subsequently arrested in wedge loaded double-cantileverbeam (DCB) specimens. The research program is carried out in two stages:

3.1 In model experiments the principal influence of dynamic effects on the crack arrest process is investigated.

3.2 Experiments with steel (22 NiMoCr 3 7) specimens are performed to measure crack initiation and especially crack arrest toughness values.

In both cases the crack velocities prior to arrest - varying as a consequence of length and sharpness of the starter notches - are measured.

4. Experimental Facilities, Computer Codes

A crack arrest testing device (load capacity 700 KN) has been built and is used together with a Cranz-Schardin high speed camera (24 photographs at a minimum interval time of 1 μ s) to measure velocities and stress intensity factors for propagating cracks. Computer codes to evaluate the experimental data, especially to determine static as well as dynamic stress intensity factors and crack arrest toughness values have been developed.

5. Progress to Date

- 5.1 Applying the shadow spot method dynamic stress intensity factors for propagating and subsequently arresting cracks in DCB-specimens (made from an epoxy resin - Araldit B) have been measured and compared to corresponding static values. From these data fracture toughness values were determined. Several test series for two specimen sizes (321x127x10 mm³ and 134x53x10 mm³) in which the crack velocity prior to arrest was varied from 15 up to 300 m/s were carried out.
- 5.2 Following the guide line of the model experiments several series of steel (22 NiMoCr 3 7) experiments were carried out. DCB specimens of the ordinary or Duplex type - with or without side grooves - (each 25 mm thick) were tested. The material, differently heat treated, was investigated under different crack initiation conditions. The crack surfaces have been investigated metallographically.

6. Results

- 6.1 The model experiments showed that dynamic effects influence the crack arrest process and have to be taken into account in determining crack arrest toughness values and in applying a crack arrest safety analysis. In particular it was found:

- At the beginning of crack propagation the dynamic stress intensity factor K_I^{dyn} is smaller than the corresponding static value K_I^{stat} , at the end, especially at arrest, K_I^{dyn} is larger than K_I^{stat}
- Dynamically determined crack arrest toughness values K_{Ia}^{dyn} are larger than the corresponding statically determined values K_{Ia}^{stat} . These K_{Ia}^{dyn} -values do not depend on the crack velocity prior to arrest and were found to be identical for both specimen sizes.

According to these results corrections have to be made to crack arrest safety analyses (ASME Pressure Vessel and Boiler Code Section XI - A 5300) as shown in Fig. 1. Consequences of such dynamic corrections are illustrated in Figs. 2 a and b for two special cases.

- 6.2 The wedge loading system has been optimized by a low friction insert and allows for an undisturbed symmetric (controlled by means of stress coating techniques) loading of the steel specimens and a

faultless measurement of the crack opening to calculate stress intensity factors. Cracks could be initiated and arrested in plane specimens only by hardening of the whole specimen or by using so called Duplex specimens which have an electron beam welded starter region made from an ultra high strength steel. Better results were obtained in side grooved specimens. In all cases the crack velocity could be influenced (although with poor reproducibility) by variation of length and sharpness of the starter notch. Crack arrest was observed after crack jump distances of a few mm up to some cm. Difficulties were encountered in measuring the crack velocities by high speed photography or ladder gauges, velocities now are obtained according to the Hahn-Shmueli crack jump - crack velocity relation. This relation could be verified experimentally. Crack arrest toughness values K_{Ia}^{stat} then were determined for different crack velocities prior to arrest. Metallographic investigations showed, that the different stages of crack propagation in an arrest experiment (i.e. initiation at an overload, acceleration and subsequent deceleration) could be related to characteristic changes in the crack surface appearance (ductile dimple fracture, followed predominantly by cleavage fracture and then distinct ductile fracture behaviour again).

7. Next Steps

The model experiments are completed; some supplementary steel experiments are carried out. The final report is in progress.

8. Relation with Other Projects

This project is closely related to RS 102-11, -13 and -14.

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10. Degree of Availability

The reports can be ordered from Gesellschaft für Reaktorsicherheit, Glockengasse 2, 5 Köln 1.

Fig. 1: Initiation and arrest considerations for a crack in the wall of a pressure vessel

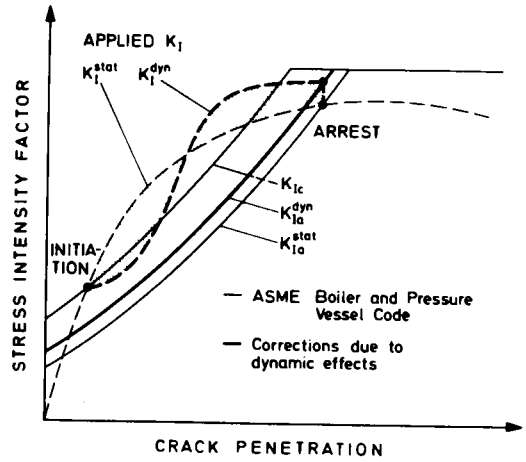
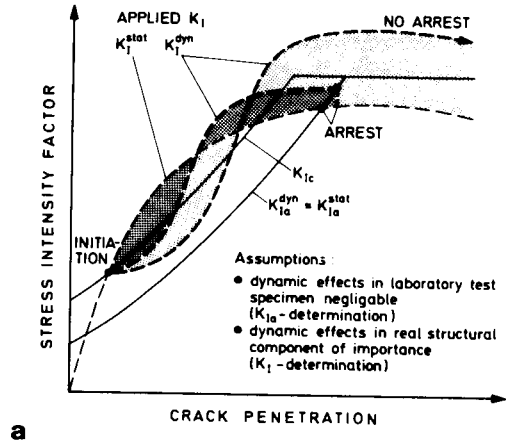
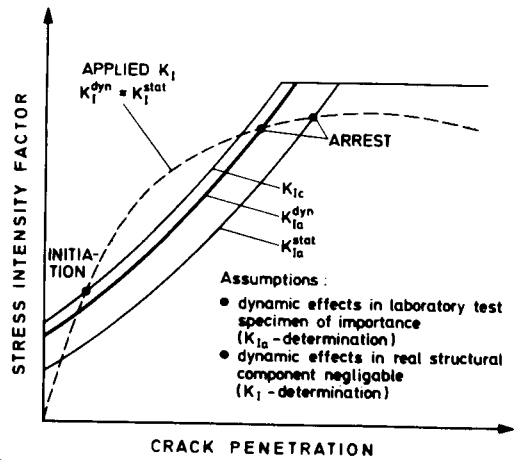


Fig. 2: Consequences of dynamic corrections for crack arrest safety analyses



a



b

<u>Classification: 11.2.1</u>	
<u>Title 1 (Original Language):</u> RiBaufweitung (COD) zur Beurteilung der Bruchgefährdung von Bauteilen (RS 102-13 -II.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: IFKM Freiburg
<u>Title 2 (English):</u> The Use of COD/COS Fracture Criteria in the Assessment of Component Strength	<u>Project Leader:</u> Dr. J.G. Blauel T. Hollstein
<u>Initiated (Date):</u> May 1973 <u>Status:</u> completed	<u>Completed (Date)</u> June 30, 1976 <u>Last Updating (Date):</u> December 1976

1. General aim

The general aim is to improve the knowledge on the assessment of safety for components of a reactor plant, in which cracks or cracklike defects have to be assumed.

2. Particular objectives

The particular objective is the assessment of the fracture behavior of components by measuring the crack opening displacement as a function of geometry and load parameters and comparing it to the material specific values COD_c/COS_c .

3. Research program

Measurement of the crack opening displacement COD along the full crack length for Compact (C) and Single -Edge -Notched (SEN) specimens with different thicknesses and crack lengths up to fracture. The load displacement curves are also evaluated in terms of the energy integral J. Two modifications of the steel 22 Ni Mo Cr 3 7 are investigated.

4. Experimental Facilities

Servo-hydraulic testing system, mechano-electrical, photoelectrical and photographic equipment for the measurement of COD, scanning electron microscope for the investigation of fracture mechanisms.

5. Progress to Date

The crack tip opening displacement COS and the J-integral have been determined experimentally for increasing loads for series of C-specimens and SEN-specimens with different crack lengths and thicknesses at room temperature. Critical values of COS, G and J of the material at fracture have been deduced and their correlations have been examined.

6. Results

- 6.1 For the material, geometry and loading conditions investigated values of COS were determined in an exact and reproducible way by direct optical measurement at the specimen surface. At the same time these values could be inferred from a single clip gage displacement measurement at the open end of the specimen on the basis of a plastic hinge mechanism with a load dependent center of rotation. As compared to other existing models a description is achieved which is the same and conservative over the whole load range for all thicknesses and crack lengths of one type of specimen. Critical values $COS_{IC} = 0.26 \pm 0.05$ mm and 0.39 ± 0.05 mm, respectively, were determined for the two materials at the onset of stable crack extension (see next paragraph).
- 6.2 J-values were determined from C-specimen tests using the compliance method of Begley and Landes (1972) and the single specimen procedure of Rice, Paris and Merkle (1973). For these two methods a close correspondence was found only if a variable factor of proportionality $2.3 \geq N \geq 1.9$ for relative crack lengths $0.47 \leq a/W \leq 0.70$ could be assumed (see Sumpter and Turner (1975) and Kuna (1976)). The method of interrupted loading with heat tinting and the electrical potential method were used to generate crack resistance (J - Δa) curves allowing the determination of the onset of stable crack extension as a safe limit prior to attainment of maximum load and complete fracture. From extrapolations $\Delta a \rightarrow 0$ critical values of the J-integral $J_{IC} = 0.20 \pm 0.04$ MN/m and 0.28 ± 0.06 MN/m, respectively, were determined. A possible dependence on the crack length is indicated only for the 75 mm specimens but together with a change in fracture mechanism.
- 6.3 A detailed scanning electron microscope study of the fracture surfaces showed that first stable crack growth was preceded - at still lower J-values - by a blunting of the crack tip region through formation of a stretch zone. This localized tearing process is strongly influenced by microstructural inhomogeneities

(e.g. the flat rolled Mn-sulphides in the less tough of the two materials). Taking that into account by a blunting line intersecting the resistance curve changed the critical values only within the existing scatter band.

- 6.4 A linear relationship is shown to exist between consistently determined values of COS and J for situations even beyond onset of stable crack growth. The term of proportionality is material dependent and includes the yield stress and a plastic constraint factor which amounts to 1.5 ± 0.2 and 1.1 ± 0.2 for the two steel modifications.

7. Next steps

The project is completed.

8. Relation with other projects

This project is closely related to RS 102-11, RS 102-12 and RS 90.

9. References

Final report (in German) IFKM 4/76

Hollstein, T.; Blauel, J.G.; Urich, B.: Ribaufweitung COD/COS zur Beurteilung der Bruchgefährdung von Bauteilen aus 22 Ni Mo Cr 3 7
Reaktortagung 1976, 813-816

10. Degree of availability

The reports can be ordered from
Gesellschaft für Reaktorsicherheit GRS
5000 Köln - 1, Glockengasse 2

<u>Classification: 11.2.1</u>	
<u>Title 1 (Original Language):</u> Stoßwellen als mögliche Schadensursachen in Kernkraftwerkskomponenten - Experimentelle Spannungsanalyse - (RS 102-14 - II.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: IFKM Freiburg
<u>Title 2 (English):</u> Damage of Nuclear Reactor Components by Shock Wave Loading	<u>Project Leader:</u> S. Winkler
<u>Initiated (Date):</u> 1.5.1973 <u>Status:</u> Completed	<u>Completed (Date):</u> 30.4.1976 <u>Last Updating (Date):</u> December 1976

1.- General Aim

In addition to static or quasistatic and thermal loading of reactor pressure vessels and components - which are very well investigated - extremely dynamic loading conditions (shock waves) may occur, for example by impinging fragments or projectiles, by internal explosions or - after certain cases of partial failure - by impact of the accelerated reactor pressure vessel against the containment. The investigation of damage in these dynamic cases is general aim of this work.

2.- Particular Objectives

The particular objective is the investigation of (1) the distribution of the produced dynamic stress fields (shock waves), (2) the interaction of these shock waves with the material, and (3) the initiation of damage in the material.

3.- Research Program

In this program plane shock waves are produced by impinging a target with a flat impactor in a gas gun. Reflected shock waves cause tension fields in the target by which microcracks are nucleated and subsequently activated to grow further, thus leading to internal damage in the material. Crack initiation thresholds and crack growing should be a function of material structure, history and temperature. The research program therefore consists of:

- 3.1 Investigation of materials with different crack nuclei and of the influence of material structure (texture), heat treatment and temperature during loading on crack initiation.
- 3.2 Measurement of the change in conventional material strength properties for such materials, containing damaged regions which are produced by impact.
- 3.3 Measurement and/or calculation of stress pulse profiles in order to understand their interaction with the material.

4.- Experimental Facilities, Computer Codes

Stress pulses are produced by the IFKM gas gun and target temperature (in situ - heating) by a RF heat generator. The loaded specimens are investigated by common metallographic methods including scanning electron microscopy. Shock wave shapes are measured by a new method using the pressure dependent voltage generation in bimetallic junctions. A one dimensional wave propagation code (PUFF⁺) is available for the calculation of stress distributions in an impacted specimen.

5.- Progress to Date

Several series of impact experiments and subsequent metallographic studies have been carried out to investigate

5.1 the nature and the behaviour of crack nuclei, the influences of texture, annealing, and temperature on the initiation threshold which indicates the sensitivity against shock loading, especially for the commonly used steel 22 Ni Mo Cr 3 7 .

5.2 In tensile experiments the strength of shock loaded (i.e. damaged) material has been measured in various directions with respect to the damaged zones.

5.3 A new method has been applied for measuring the shape of the shock pulses at the rear surface in order to get a better knowledge of the stress distribution at the instant of damage. The wave propagation code has been adjusted to the local UNIVAC computer. Difficulties were encountered because of the lower accuracy of this machine.

6.- Results

6.1 For steel, in particular the reactor pressure vessel steel 22 Ni Mo Cr 3 7 ,

⁺ developed by Stanford Research Institute, Menlo Park, California , U.S.A.

the following results on the nucleation and growth of microcracks were found: Damage occurred by exceeding a shock wave pressure threshold of approximately 25-26 kbar (\approx 120 m/s impact velocity at 25°C and about 20-22 kbar at 300°). The exact value depends slightly on texture and annealing.

Two main types of crack nuclei were found: impurities (sulphides and oxides) and ferrite precipitations, latter are less important for the reactor vessel steel. Microcracks were nucleated either by brittle cleavage of grains or brittle cracking of grain boundaries. Ductile extension and coalescence of microcracks lead to macroscopic fracture, by which the integrity of the vessel or component may be endangered severely.

6.2 Tensile specimens were machined from the damaged zone of the targets impacted with different velocities. Especially because of cracks nucleated at flat inclusions parallel to the rolling plane the tensile strength perpendicular to the rolling plane decreases linearly by about 50 % by increasing impact load from 30 to 60 kbar. Controversely the tensile strength parallel to the rolling or the transverse direction decreases only by less than 10 % of the strength of the unloaded material.

6.3 Shock profiles are measured successfully by using the method of voltage generation at a bimetallic junction during pressure loading. From these measurements the stress distribution at the damaged plane inside the specimen can be extrapolated with a higher accuracy than before.

The conclusion which can be made at this status of the investigations is that the effective tensile stress at the damage plane in our experiments does not reach the value of the impact pressure. The reason is the finite risetime of the shockwave (e.g. 0.5 microseconds for an impact pressure of 45 kbar in a 22 Ni Mo Cr 3 7 sample of 6 mm thickness. That means that the material especially at lower impact velocities, where the risetime increases, will be damaged at even lower stresses than reported above.

7.- Next Steps

The work has been finished. The final report is available.

8.- Relation with Other Projects

This work is related to the investigations 11, 12 and 13 within the project RS 102.

9.- References

Results are published in the

IFKM Report 8/74 (German)

and

IFKM Report 3/76 (German), Final Report,

both entitled "Shock Waves as Possible Causes of Damage to Nuclear
Reactor Components"

10.- Degree of Availability

This reports can be ordered from

Gesellschaft für Reaktorsicherheit

Glockengasse 2, 5 Köln - 1

<u>Classification:</u> 11.2.1				
<u>Title 1 (Original Language):</u> Bestrahlungseinfluß auf Festigkeit und Relaxation von hochfesten austenitischen Stählen und Nickellegierungen für Verbindungselemente der Kernstruktur (RS 133 - II.2, Jahresbericht A 76)	<u>COUNTRY:</u> BRD			
	<u>SPONSOR:</u> BMFT			
	<u>ORGANIZATION:</u> KWU, Erlangen			
<u>Title 2 (English):</u> Irradiation Influence on Strength and Relaxation of Austenitic Steels and Ni-alloys for Core Structure Connections	<u>Project Leader:</u> H. Debray			
	<table border="0" style="width: 100%;"> <tr> <td style="width: 50%;"><u>Initiated (Date):</u> 1. 2. 75</td> <td style="width: 50%;"><u>Completed (Date):</u> 30. 9. 76</td> </tr> <tr> <td><u>Status:</u> Completed</td> <td><u>Last Updating (Date):</u> 31. 12. 76</td> </tr> </table>	<u>Initiated (Date):</u> 1. 2. 75	<u>Completed (Date):</u> 30. 9. 76	<u>Status:</u> Completed
<u>Initiated (Date):</u> 1. 2. 75	<u>Completed (Date):</u> 30. 9. 76			
<u>Status:</u> Completed	<u>Last Updating (Date):</u> 31. 12. 76			

1. General Aim and 2. Particular Objectives

The influence of neutron irradiation on the strength and relaxation of materials for core structure connections of light water reactors had to be investigated.

3. Research Program

For three characteristic materials and two heat-treatment conditions of each quantitative experimental results were expected:

1. Steel Nr. 1.4550, representative for austenitic steel as used in LWR's.
2. Inconel-X-750, representative for connection material (frequently used in USA).
3. Steel Nr. 1.4980 (USA: A 286) with lower tendency towards selective corrosion and lower thermal expansion coefficient compared with Inconel-X-750.

4. Experimental Facilities

Irradiation in special irradiation channels in the Obrigheim Power Reactor (KWU). Post irradiation testing in the hot cell facility of KWU, Erlangen. Testing of unirradiated samples in the materials testing laboratory of KWU, Erlangen. Machines used for this program: Tensile testing machine and high-frequency pulsator (Fatigue tester).

5. Progress to Date

The evaluation of the irradiated and unirradiated samples was continued.

The possibility of determining the modulus of elasticity from irradiated tension samples was investigated. Theoretically the influence of the irradiation on the modulus of elasticity was studied in order to find out, whether for the conversion of tension-data to fatigue curves (Larger-Manson-Coffin-model) the E-module after irradiation was necessary.

6. Results

The measured values of the modulus of elasticity (unirradiated) were within $\pm 6\%$ in the range of the producers data.

The monitor evaluations showed that a mean neutron fluence of $3,9 \cdot 10^{20} \text{ cm}^{-2}$ ($E > 1 \text{ MeV}$) had been in the middle of the channel and $5,1$ or $3,0 \cdot 10^{20} \text{ cm}^{-2}$ for the samples, directed to the core or in the opposite direction.

7. Next Steps

The work has been completed.

8. Relation with Other Projects

9. References

10. Degree of Availability

<u>Classification:</u> 11.2.1								
<u>Title 1 (Original Language):</u> Literaturstudie: Selektives Korrosionsverhalten von in Leichtwasser-Reaktoren eingesetzten Werkstoffen (RS 159 - II.2, Jahresbericht A 76)	<u>COUNTRY:</u> BRD							
	<u>SPONSOR:</u> BMFT							
	<u>ORGANIZATION:</u> KWU, Erlangen							
<u>Title 2 (English):</u> Literature Study on Selective Corrosion of LWR- Materials	<u>Project Leader:</u> Stieding							
	<table style="width: 100%; border: none;"> <tr> <td style="width: 50%;"><u>Initiated (Date):</u></td> <td style="width: 50%;"><u>Completed (Date):</u></td> </tr> <tr> <td>-1. 4. 75</td> <td>30. 6. 76</td> </tr> <tr> <td><u>Status:</u></td> <td><u>Last Updating (Date):</u></td> </tr> <tr> <td>Completed</td> <td>31. 12. 76</td> </tr> </table>	<u>Initiated (Date):</u>	<u>Completed (Date):</u>	-1. 4. 75	30. 6. 76	<u>Status:</u>	<u>Last Updating (Date):</u>	Completed
<u>Initiated (Date):</u>	<u>Completed (Date):</u>							
-1. 4. 75	30. 6. 76							
<u>Status:</u>	<u>Last Updating (Date):</u>							
Completed	31. 12. 76							

1. General Aim

Some materials (austenitic, ferritic and martensitic steels and Ni-alloys) which are used in the nuclear steam generating system were to be investigated, whether they could be attacked by selective corrosion under certain conditions. As a first step a literature study has been conducted on the behaviour of some selected materials.

2. Particular Objectives

In order to raise the safety of operating reactors and to get more information about the risks of longtime effects the knowledge of the behaviour of primary materials had to be extended over the limits of the usual operation lifetime data.

3. Research Program

In the first step the following details will be studied from the literature:

- corrosion under RPV-cladding, caused by material failures
- influence of the corrosion on the failure behaviour and the crack growth
- hydrogen induced corrosion
- determination of the corrosion limits of materials containing Ni
- influence of gaps and different material combinations
- electrochemical investigations on the corrosion mechanism.

4. Experimental Facilities

No experimental facilities necessary.

5. Progress to Date

The literature study was continued, regarding Ni-Cr-Fe alloys in the PWR primary circuit and in the BWR.

The intercrystalline corrosion was studied

- under imperfection of the RPV cladding
- for different material combinations in pure hot water
- on the influence of special mechanisms, investigated by electrochemical methods.

6. Results

The published data on the behaviour of austenitic Ni-Cr-Fe alloys with respect to intercrystalline corrosion, caused by Cr-pauperisation of the grain boundaries, allow a theoretical calculation of the attack on Ni-base-alloys after different heat conditions.

The contact of Inconel 600 with C- and low alloyed steel or stainless steel does not increase the intercrystalline corrosion in pure hot water. Austenitic Cr-Ni-steel however increases the perforation-corrosion of RPV-steel in BWR-water.

RPV steel will be perforated in pure hot water. The velocity however was not defined quantitatively. Some publications show that the deepest holes have an asymptotic statistic and with the aid of the Gumbel-statistic the perforation probabilities and perforation times can be determined.

7. Next Steps

The work has been completed.

8. Relation with Other Projects

9. References

10. Degree of Availability

<u>Classification:</u> 11.2.1	
<u>Title 1 (Original Language):</u> Reduktion der Aktivierung und Kontamination von Reaktorkreisläufen (PNS 4123 - II.2, Jahresbericht A 76)	<u>COUNTRY:</u> BRD
	<u>SPONSOR:</u>
	<u>ORGANIZATION:</u> PNS/GfK
<u>Title 2 (English):</u> Reduction of Activation and Contamination of Reactor Loops	<u>Project Leader:</u> I. Michael, IRB
<u>Initiated (Date):</u> Jan. 1974 <u>Status:</u> Continuing	<u>Completed (Date):</u> Dec. 1977 <u>Last Updating (Date):</u> Dec. 1976

1. General Aim

The aim of this investigation is reduction of radiation dose to maintenance and repair personnel in nuclear power reactors.

2. Particular Objectives

Laboratory scale investigation of the metal loss rate to the water and deposition of corrosion products under conditions of the modern pressurized water reactors, i.e.: INCOLOY 800 and other stainless steel materials, temperature 342 °C, pressure 150 bar, water chemistry (boron acid, LiOH).

3. Research Program

The influence of the water chemistry was examined in several steps started with pure water. The program will be finalized in 1977 with all additional components in the water.

4. Experimental Facilities

The protective gold plating applied on the inner side of the autoclave system I had to be renewed after investigation of the metal loss rate

of INCOLOY 800 specimens by several test series.

The autoclave system II (same dimensions as the autoclave system I) also provided with an inner gold plating was nearly completed.

A small sized autoclave system III was completed which allows to study the metal loss rate to pressurized water of standard chrome-nickel steel, the material making up 15 % of the inner surface of primary circuits in PWRs.

5. Progress of Date

Investigations were performed of the metal loss rate of the INCOLOY 800 steam generator material at 342 °C and 150 bar with oxygen contents in the pressurized waters of 0,05 mg/kg, extending over twice the running period of 1000 hours and, also, with additional alkalization using lithium hydroxide (pH-value 9,7).

The older version of the atomic absorption spectrometer used for the determination of the metal content in water was automated for laboratory purposes.

6. Results

Reduction of the oxygen content in deionate through noble gas sweeping which is much more effective than chemical reduction by hydrazine was improved in combination with the autoclave system I. It appeared that the minimum oxygen content of 0,05 mg/kg required for primary loop water of pressurized water reactors can still be underrun by the factor of 20.

In the two metal loss rate runs an abrupt, discontinuously enhanced metal concentration was found in the pressurized water after about 400 hours. It suggests the growth of the surface layer on the INCOLOY specimens until an instable thickness is reached at which the metal oxide begins to detach.

Automation on the laboratory scale of the atomic absorption spectroscopy essentially resulted in making shorter the handling and evaluation period and in a higher accuracy of analytics.

7. Next Steps

Three autoclave systems will be available to continue the investigations. The discontinuous metal loss rate observed calls for further investigations. They will relate to the chemico-physical partial processes of transport of the metal and metal oxide content released in the pressurized water and of the determination of the solid content, classified by size.

8. Relation with Other Projects

The physico-chemical analysis of the corrosion layers is performed by Dornier System, Friedrichshafen.

9. References

I. Michael, C. Plog,
Dünnschichtanalyse von Druckwasserkorrosions-Schichten auf INCOLOY 800-Proben (to be published in "METALL")

<u>Classification:</u> 11.2.1	
<u>Title 1 (Original Language):</u> Erweiterung des Einsatzbereiches kobaltfreier Werkstoffe für Reib- und Verschleißbeanspruchung im NDES von Leichtwasserreaktoren (RS 207 - II.2, Jahresbericht A76)	COUNTRY: BRD
	SPONSOR: BMFT
	ORGANIZATION: KWU, Erlangen
<u>Title 2 (English):</u> Extension of the Use of Cobalt-Free Materials for Resistance of Friction, Wear and Corrosion in the Nuclear Steam Supply System (NSSS) of LWRs	<u>Project Leader:</u> H. Stieding
<u>Initiated (Date):</u> 1. 5. 76	<u>Completed (Date):</u> 30. 9. 79
<u>Status:</u> Continuing	<u>Last Updating (Date):</u> 31. 12. 76

1. General Aim

Reduction of contamination of the NSSS by products of corrosion and wear, which together with the coolant pass through the reactor core and thereby become activated.

2. Particular Objectives

- a) Selection and (eventually in cooperation with the manufacturers) development of Co-free plating materials, which can also be exposed to impact stress conditions and which can be worked more easily than Everit 55. Investigations of the behaviour of these materials und conditions of sliding, impact and corrosion.
- b) Selection and evaluation of plasma-sprayed materials with satisfactory corrosion and wear properties under PWR and BWR conditions.
- c) Selection and evaluation of stainless steels which are expected to show better friction and wear behaviour than material No. 1.4550.

3. Research Program

- 3.1 Selection of test materials:
pre-selection, acquisition, fabrication of samples, adhesion tests, friction and wear tests at 20°C.
- 3.2 Corrosion tests:
fabrication of samples, corrosion tests at 320°C, tests for resistance to intergranular corrosion.
- 3.3 Friction and wear tests:
fabrication of samples, tests at 20°C and 300°C.
- 3.4 Mechanical tests:
fabrication of samples, adhesion tests, brittleness test
- 3.5 Evaluation of test results

4. Test Facilities

The required test facilities have either been used in previous tests or have recently been developed (hydraulic apparatus for RT tests; stand for high-temperature wear tests at 300°C with simultaneous influence of corrosion; static and refreshing autoclaves for long term corrosion tests) and are available for the above described tests.

5. Progress to Date

A literature search has been made to select suitable, commercial FeCrC alloys. Well-known manufacturers have also been contacted in order to discuss the possibilities of fabricating suitable plating materials and their willingness to do so.

6. Results

Literature searches gave the following results:

- a) There are very few suitable FeCrC alloys available on the market.
- b) "Pantanax 25/0 Mo", a TEW-made FeCrC alloy, presently not available for commercial use, is expected to be a suitable material for the planned tests.
- c) In addition to the FeCrC alloys (high-chrome cast-iron), overlay welding materials ("Nitronic"; Ni-based alloys) should also be tested.

Discussion with the manufacturers led to the following results:

a) FeCrC alloys

Co-free plating materials showing the required combination of characteristics have not yet been studied. The major problem here is the temperature requirement (approx. 320 - 350°C) during long term operation (40 years), which requires extensive work in order to investigate the suitability of just one FeCrC alloy. Therefore it was agreed to proceed step by step. A TTT diagram will be made of Pantanax 25/0 Mo-G to provide a basis for specific development.

b) Nitronic steels

Comparative tests, performed by the manufacturers, showed that the friction and wear behaviour of these steels at RT is fairly good. Additional tests are required on: corrosion behaviour, friction and wear behaviour under defined conditions at RT and 300°C and the temperature stability of platings of these materials.

c) Tribaloy

According to the manufacturers, the discussed alloy T 700 can be applied by shielded arc welding. Unfortunately, the material shows low ductility. While temperature stability seems to be guaranteed, friction wear and corrosion behaviour have still to be evaluated. As a first step, samples of cast material will be carefully pre-tested.

7. Next Steps

According to the current task matrix, the next steps will be as follows:

- establishment of a material list
- acquisition of test material

At the same time a literature survey will be made on "Plasma sprayed layers".

8. Relation to Other Projects

Preparatory work has been done under the scope of R & D task RB 72 I/S 11.

9. References

P. Hofmann

Wear resistance of materials and protective layers with respect to their behaviour in water-cooled nuclear power plants
final report R & D task RB 72 I/S 11, KWU, Aug. 1973.

10. Degree of Availability

The report is classified Company Confidential

PROJECT TITLE : Assessment of Irradiation Embrittlement on Reactor Pressure Vessel	CSNI INDEX NUMBER : 11.2.1
SPONSORING COUNTRY : Japan	ORGANISATION : JAERI, Tokai
DATE INITIATED : April 1974 DATE COMPLETED : Progressing (March 1977)	PROJECT LEADER : T. Kodaira

Description :

1. Objective

The investigations have been done in order to clarify the behavior of neutron irradiation embrittlement on the currently used heavy section steels and other higher strength steels such as ASTM A542 or A543, and to ensure the structural integrity of reactor pressure vessels.

2. Project Status

Two programs are now in progress. The first is surveillance test program on the pressure vessel of nuclear power plants under operation, and the second is neutron irradiation research program based on transition temperature and fracture mechanics approaches for the assessment of integrity of the pressure vessel and for the development of advanced steels.

The following programs were conducted.

- a) The 1st surveillance test of Shimane reactor (BWR) and 5th surveillance test of Tsuruga reactor.
- b) The neutron irradiation and post-irradiation tests of several pressure vessel steels with different strength level.
- c) J_{IC} fracture toughness tests of currently used heavy section steels at pre-irradiation state.

3. Essential Results

- a) The structural integrity was established in the pressure vessels of Shimane and Tsuruga reactors.
- b) Interim results have been obtained.

(Continued from the preceding page.)

- c) J_{IC} fracture toughness of non-irradiated steels was successfully obtained using small specimens.

4. Projection

- a) Two surveillance tests of currently operated BWR reactors will be carried out in this year.
- b) The post-irradiation tests of several pressure vessel steels with the different strength level will be performed in forthcoming period.
- c) The neutron irradiation test will be done in this year in order to obtain the J_{IC} fracture toughness of domestic heavy section pressure vessel steels.

5. Reports

T. Kodaira and K. Ishimoto, "Surveillance Test of Light-Water Power Reactors, Part I, Practice and Items of the Tests", JAERI-M-5868 (Sept. 1974).

PROJECT TITLE : Interaction between Fatigue and Creep of Nuclear Reactor Materials	CSNI INDEX NUMBER : 11.2.1.
SPONSORING COUNTRY : Japan	ORGANIZATION : The Japan Welding Engineering Society
DATE INITIATED : May, 1973 DATE COMPLETED : July, 1976	PROJECT LEADER : K. Iida

Description :

1. Objectives

The objectives are to investigate experimentally the interaction between fatigue and creep of AISI type 304 and 316 stainless steels and ASTM type 2%Cr-1Mo steel and to study experimentally and theoretically the thermal and mechanical ratcheting.

2. Project Status

The tests of the interaction between fatigue and creep of three materials have been performed at 350, 450, 550 and 650°C to study the effects of strain rate, holding time and temperature on the fatigue life of their base metals and weld metals.

Three bar models and cylinders have been used to investigate experimentally and theoretically the thermal and mechanical ratcheting.

3. Essential Results

It is clarified that the fatigue life of three materials is shortened when higher the testing temperature is and longer the holding time is.

The loading conditions in which thermal of mechanical ratcheting occurs are obtained by three bar models and cylindrical models.

4. Projection

This project had been completed in July, 1976.

5. Reports

- 1) K. Iida and etc., "Research on Structural Design Code considering Interaction between Fatigue and Creep of Nuclear Reactor Materials", Japan Welding Engineering Society, 48 FCI (Feb.1975)
- 2) Ibidem; 49 FCI (Feb. 1976)
- 3) Ibidem; 50 FCI (Oct. 1976)

OECD NUCLEAR ENERGY AGENCY - 38, Boulevard Suchet, 75016 PARIS
Telex : 630668 AEN NEA - Tel : 524 82-00

PROJECT TITLE : On Safty Margin of the Simplified. Plastic Fatigue Design Method for Nuclear Components	CSNI INDEX NUMBER : 11.2.1
SPONSORING COUNTRY : Japan	ORGANISATION : Ishikawajima-Harima Heavy Industries Co., Ltd.
DATE INITIATED : April 1975 DATE COMPLETED : Progressing	PROJECT LEADER : Masaki Kitagawa

Description :

1. Objectives : A method recommended in ASME B & PV Code Sect. III is believed to be partially non-conservative. The present experimental investigation will supply basic data for determining the safty margin of the ASME Code Method.
2. Project Status : An experimental work is under progress.
3. Essential Results : In the above code method, two design constants are used. One of them is related to the cyclic stress-strain exponent. A series of cyclic stress-strain experiments had been performed, and the conservatism of the design constant had been proved. Experiments are in the progress to discuss the second design constant.
4. Projection : Investigation is scheduled at elevated temperatures.
5. Report : Informal report to MITI and to Prof. Iida of Tokyo Univ.

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31.719

PROJECT TITLE : Study on Assesment of Safety against Fatigue Crack Propagation in Nuclear Pressure Vessels and Pipe Circuits	CSNI INDEX NUMBER : 11.2.1.
SPONSORING COUNTRY : Japan	ORGANIZATION : The Japan Welding Engineering Society
DATE INITIATED : 1/4/1976 DATE COMPLETED : 31/3/1979 (planned)	PROJECT LEADER : Prof. Kunihiro IIDA

Description :

The present research project is destined to the study of the following items, which are of the most importance in safety evaluation of nuclear pressure vessels and pipe circuits:

- 1) Fatigue crack propagation from single or multiple flaws under axial tension, simple bending and general combined tensile and bending loads
- 2) Application of numerical methods and photoelasticity on the analysis of stress intensity factor of surface and internal cracks
- 3) Micro-fractography on crack propagation and crack coalescence under cyclic fatigue loadings
- 4) Examination and improvement of safety evaluation analysis procedures presented in ASME Boiler and Pressure Vessel Code, Section XI

The program of research contains: basic mechanical tests of specimen materials, A533B steel and SUS 304 steel, fatigue crack propagation tests under various flaw shape condetions and loading conditions in plates and cylinders, analysis of stress intensity factor both in numerical methods and 3-dimensional photoelasticity, micro-fractography by scanning electron microscope and a large-scale model test to confirm the validity of analysis in the final year of research.

OECD NUCLEAR ENERGY AGENCY - 38, Boulevard Suchet, 75016 PARIS
Telex : 630668 AEN NEA - Tel : 524 82-00

PROJECT TITLE : Study on the behavior of the local structure of nuclear reactor piping for its safety evaluation	CSNI INDEX NUMBER : 11.2.1. 11.2.2.
SPONSORING COUNTRY : Japan	ORGANIZATION : The Japan Welding Engineering Society
DATE INITIATED : July, 1975 DATE COMPLETED : March, 1977	PROJECT LEADER : Y. Ando

Description :

1. Objectives

The objectives are to investigate the phenomena of Intergranular Stress Corrosion Cracking (IGSCC) in Nuclear piping system from the viewpoints of : (1) localized stresses, (2) material conditions, and (3) environmental conditions.

2. Project Status

A computer code calculating localized elastic-plastic stresses has been made by using a finite element method. The accuracy of this code has been checked with some experimental results. The code is being improved to analyze cracked pipes. A basic experimental study of cracked pipe is also in progress.

No.2 working group of 5SS (1975 year) and 6SS (1976 year) subcommittees consists of more than twenty members from universities and industries which have been conducting joint researches. They have conducted high temperature water SCC tests on Types 304, 304L, 321 and 347 sensitized under conditions of various combination of heating periods and temperatures. In addition, intergranular tests were carried out for the above-mentioned materials.

Temperature history and residual stress were measured for automatically and manually welded joints of 4" x Sch 80 Type 304 tubes to evaluate the relationship between the above-mentioned factors as Cl⁻ and dissolved oxygen concentration on SCC of Type 304 were also investigated.

3. Essential Results

- (1) Local stresses can be calculated by good accuracy using the the computer code developed in this project.
- (2) When sensitized between 550~800°C for 6~600 minutes, Types 304 and 304L suffer from high temperature high purity water SCC at higher temperatures for shorter periods and at lower temperatures for longer periods, though 304L is much less susceptible to SCC than 304.
- (3) Stabilized 321 and 347 is further superior to 304L in SCC resistance in the same environment as above.
- (4) Above-mentioned results were obtained consistently both by the double U bend method and by the constant strain rate method.

4. Projection

The project of second year is to be completed in March, 1977. The project of third year is in the planing stage.

Correlation between furnace sensitization and weld sensitization will be investigated. Measured residual stress values will be compared with calculated ones for both 4" and 14" Type 304 tubes.

5. Reports

Y. Ando, "Research on Local Structural Behavior and Safty Analysis of Nuclear Piping System", JWES, 5SS (Oct. 1976) (Japanese)

PROJEC TITLE : Study on Dynamic Fracture Toughness of Nucleare Pressure Vessel Steel	CSNI INDEX NUMBER : 11.2.1. & 11.2.4.
SPONSORING COUNTRY : Japan	ORGANISATION : Ship Research Institute Ministry of Transport
DATE INITIATED : April, 1976 DATE COMPLETED : Progressing (March, 1977)	PROJECT LEADER : E. Fujii

Description :

1. Objective

The objective of this project is to experimentally investigate and realize the dynamic fracture toughness of structural steels and to find the relations between some other notch-toughness tests.

2. Project Status

2.1. Program Description

In this program, dynamic brittle fracture tests are scheduled under some variable test conditions of applying load velocity, type and size of specimens and load condition (tension or bending).

2.2 Progress

Test facility using in this program is under construction.

3. Essential Results

None

4. Projection

In the practical study, it is scheduled to investigate the dynamic fracture toughness of nuclear pressure vessel steels and to evaluate the effect of dynamic loading conditions.

5. Reports

None

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31.719

PROJECT TITLE : Study on Structural Safety of Nuclear Pressure Vessel against to Inner Pressure	CSNI INDEX NUMBER : 11. 2.1. & 11.2.4.
SPONSORRING COUNTRY : Japan	ORGANISATION : Ship Research Institute Ministry of Transport
DATE INITIATED : April, 1974 DATE COMPLETED : Progressing (March, 1977)	PROJECT LEADER : E. Fujii

Description :

1. Objective

The objective of this project is to investigate the brittle fracture strength of the pressure vessel models including some cracks under internal pressure, and to find the correlation between the flat plate and the vessels, and scale effect on the brittle fracture characteristics.

2. Project Status

2.1 Program Description

In this program, two series of models are tested to obtain brittle crack initiation and propagation behaviors from the crack in vessel wall. The first one is the cylindrical vessel model series and the other is cylindrical vessel with nozzle as a stress raiser.

The brittle fracture test of model vessels are scheduled under some variable test conditions of temperature, crack size and model dimension.

2.2 Progress

The brittle fracture characteristics of steel used for vessel models were investigated on flat plate specimens. Cylindrical vessels of 25mm wall thickness without and with nozzles are carried out. (To be continued on the next page)

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Telex : 630668 AEN NEA - Tel : 524 82-00

(Continued from preceding page)

3. Essential Results

The brittle fracture characteristics of model vessel steel and brittle fracture behaviors of model vessels were obtained.

4. Projection

Test data obtained in these tests will be analyzed to evaluate brittle fracture strength and failure analysis of pressure vessel.

5. Reports

None

TNO - Metaalinstituut		CLASSIFICATION: 11.2.1.
TITLE: Literatuurstudie: (1) Invloed van omgevingscondities op vermoeiings- scheurgroei. (2) Invloed van veroudering op de mechanische materi- aaleigenschappen.		COUNTRY: NETHERLANDS. SPONSOR: Ministry of Social Affairs ORGANIZATION: TNO-Metaalstituut
TITLE: (ENGLISH LANGUAGE): Literature study: (1) Influence of Environment on Fatigue crackgrowth. (2) Influence of aging on fracture related material properties.		PROJECT LEADER: Van Rongen
INITIATED: June 1974	LAST UPDATING: April 1977	SCIENTISTS: —
STATUS: near completion	COMPLETED: mid 1977	

General aim

To improve the understanding of material behaviour during lifetime of nuclear vessels and other pressurized components as influenced by aqueous environment and aging effects. A "state of the art" report will be submitted.

Particular objectives

To review the knowledge in the areas

- . influence of reactor coolant on fatigue crackgrowth in LWR pressure vessels, with particular attention to chemical composition of the coolant, type of pressure vessel steel, frequency, amplitude and 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

1. Progress to date: literature survey completed
2. Essential results: to be reported mid 1977

Next steps: ready for the press

Relation with other projects: -

Reference documents: -

Degree of availability:

Through the Ministry of Social Affairs, C.R.V., Postbus 69, Voorburg, The Netherlands

Budget: Hfl. 50.000,--

Personnel: 2 scientists

NIL & MI - TNO		CLASSIFICATION: 11.2.1/2.3
TITLE: Literatuurstudie van materiaalparameters die nodig zijn bij de interpretatie van defect-localisatie met AE-apparatuur		COUNTRY: NETHERLANDS. SPONSOR: Ministry of Social Affairs ORGANIZATION: and others NIL, MI - TNO
TITLE: (ENGLISH LANGUAGE): Literature survey of material parameters which are necessary for the diagnosis of defects localized with AE techniques.		PROJECT LEADER: Kloots
INITIATED: September 1974	LAST UPDATING: April 1977	SCIENTISTS: Boerstael V.d. Brink
STATUS: in progress	COMPLETED: May 1977	

General aim

This particular survey is a preliminary study of effects to be included in an extensive study which will be initiated in the future and which can be characterized as follows:

"Evaluation of the practical application of Acoustical Emission (AE) techniques during construction, testing and operation of welded constructions, in particular pressurized components to be used in the energy and process industry in order to improve safety, reliability and economic construction of components".

Optimum utilization of AE-apparatus for defect localization and diagnosis is only possible if sufficient data on the AE behaviour of constructional material are available (AE material parameters). This study comprises the use of AE material parameters for nuclear vessels and their application to the diagnosis of defects localized in experiments.

Particular objectives

Study and inventarization of AE research in literature.
 Visits to industry and institutes in the Netherlands.

Experimental facilities and program: -

Project status: to be completed soon

Next steps: final report

Relation with other projects: see under 1

Reference documents: NIL-lastechiek, 40e jaargang, no. 5, mei 1974

Degree of availability: Through Nederlands Instituut voor Lastechiek

Budget: Approx. Hfl. 30.000,--

Additional information:

Survey will be executed by Metaal Instituut (TNO) and will be completed May 1977, approximately.

PROJECT TITLE : Evaluation and study of acoustic emission by materials and structures	CSNI INDEX NUMBER : Water reactors 11.2.3.(11.2.1)
SPONSORING COUNTRY : Switzerland	ORGANISATION : Swiss Federal Institute for Reactor Research, Würenlingen
DATE INITIATED : 1977 DATE COMPLETED :	PROJECT LEADER : L. Hanacek

Description :

1. General aim

The objective of this research-project is to study and evaluate acoustic emission phenomena initiated by deformation of structural and reactor steels. The connection between fracture mechanics and acoustic emission (AE) will be investigated to evaluate the structural safety of pressurised components.

2. Particular objectives

- 2.1 To correlate and optimize AE-measurements by employing the following techniques: ringdown counting, energy, amplitude distribution, frequency analysis. The study will also consider the influence of specimen-form on the character of AE.
- 2.2 Identification of the types of acoustic signals emitted from various sources and relationship between the kinetics of deformation in metals and detected AE from:
 - plastic deformation and local yielding
 - crack initiation and propagation
 - brittle cracking and ductile tearing
- 2.3 AE measurements on fracture toughness specimens during cyclic and tensile loading. This analysis will show the intensity of AE and crack growth rate and connection to the stress intensity factor.

3. Experimental programme

This programme consists of two parts.

First - the basic study (points 2.1, 2.2, 2.3)

Second- the practical applications of AE-method on source location and classification of defects on pressure vessel models.

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PROJECT TITLE : Two-dimensional growth of cracks/fatigue (in original language: Zweidimensionales Risswachstum/Ermüdung).	CSNI INDEX NUMBER : 11.2.1 Water reactors
SPONSORING COUNTRY : Switzerland	ORGANISATION : Swiss Federal Institute for Reactor Research (EIR) Würenlingen
DATE INITIATED : 11.8.1975 DATE COMPLETED : Status: continuing	PROJECT LEADER : M. Prodan Engineering Division

Description :

1. General aim

It is the aim of the present work to examine the growth characteristics and behaviour of part-through cracks in nuclear reactor components (e.g. tubes, vessels). It is intended to use experimental methods, based on the concepts of fracture mechanics. Using the results of this work an assessment will be made of the "leak before break" question (see attached illustration).

2. Detailed objective

"Two-dimensional", semi-elliptical starting flaws are machined into tension plate specimens. "One-dimensional" flaws are machined into compact tension (CT) specimens. The fatigue crack growth rates in both types of specimen will be compared. The growth rates will also be compared with hypothetical crack growth in tubes. During propagation crack depths are periodically measured using an alternating current (AC) electrical potential technique. Crack shape is simultaneously marked by changing the load range. Conventional instrumentation with strain gauges is coupled to a medium-size computer. This arrangement is used to measure strains and stresses in the flawed region of the tension plate specimens. All the experimental results will be compared with published data. An empirical

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relationship will be derived for predicting the shape of crack fronts propagating through a structure. In order to utilise the results for a real component, a simple model will be discussed.

3. Experimental facilities

The fatigue tests with tension plates are performed on a 1000 kN hydraulic machine. The tests with CT specimens are made on a 100 kN mechanical high frequency pulsator and on a 60 kN hydraulic pulsator. The nominal (gross) stresses in the plates are by mode-1 crack stress field 250 ... 300 N/mm².

4. Project status

4.1 Progress to date

- Computational investigation of the usability of the existing "leak before break" criteria;
- Statement of the programme of experimental work (project plan);
- Pretests with BH 43 W steel plates, thickness 20 mm;
- Procurement of 20 MnMoNi 55 steel plates, thickness 30, 50 and 70 mm;
- Electronbeam welding of the plate specimens.

4.2 Essential results to date

The pretests with CT and plate specimens are almost completed. A preliminary report will be prepared, in which the tests are described in detail. The plate specimens were gripped in the test machine using clamping jaws. No distinct change in the polar angle of the maximum crack growth ($\varphi = 90^{\circ}$) was observed as the crack depth increased. The semi-elliptical shape of the crack was retained.

5. Next steps

Further investigations of:

- methods for experimental determination of crack propagation rate (AC potential technique and eventually acoustic emission);
- influence of load interactions;
- deep flaw stress intensity magnification factors;
- correlations between one- and two-dimensional growth;
- comparisons with tests of axially aligned cracks in tubes and other crack configurations;
- low-cycle fatigue and elastic-plastic fracture (failure) parameters;
- alternative steels, temperatures and environments;
- unstable crack growth and arrest phenomena.

6. Relations with other projects

- Reactor safety projects in the IFKM Freiburg and MPA Stuttgart, Germany;
- Investigations of the Central Electricity Generating Board, Berkeley Nuclear Laboratories/England and of the Department of Naval Architecture, University of Tokyo.
The Hitachi Shipbuilding and Engineering Company, Osaka/Japan.

7. Reference documents

EIR-internal reports:

AN-IN-510	11.8. 1975
TM-IN-607	7.10. 1975
TM-IN-624	9.7.1976

8. Budget 1977

Laboratory costs	Sfr.	70'000.-
Materials and instruments	Sfr.	30'000.-
Personnel		1,2 men/year

Würenlingen, 7 April 1977

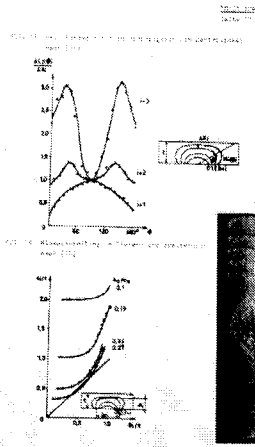
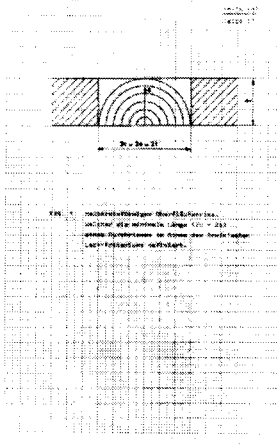
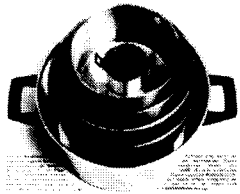
Enclosed: 1 illustration (photo)

Zweidimensionales Risswachstum/Ermüdung

Vorhaben Bruchmechanik

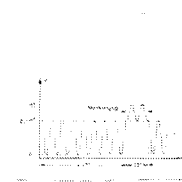
Referenzen (USA, BRD, JAPAN, ENGLAND):

Noch viele Unklarheiten



EIR - Untersuchungen

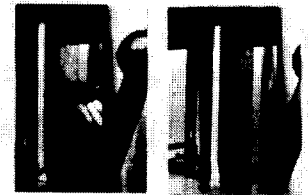
(Vorversuche)



PLATTE MIT ZWEIFACHENRISSEN
 BEI:
 DRUCKBEI 1000 N/mm²
 ANKER-DEHNSTRECKEN 10% (KIP)



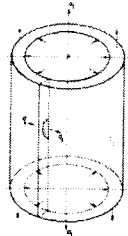
WIE WICHTIG DAS RISSLEITENMESSGERÄT!



934

Anstoss: Das 'Leck-vor-Bruch'-Problem (anwendungsbezogen)

Die Aufgabe ist es, die Funktion der Leckvorbrüche zu verstehen und zu erklären.



- 1. Leckvorbruch
- 2. Leckvorbruch
- 3. Leckvorbruch
- 4. Leckvorbruch
- 5. Leckvorbruch
- 6. Leckvorbruch
- 7. Leckvorbruch
- 8. Leckvorbruch
- 9. Leckvorbruch
- 10. Leckvorbruch

Modell

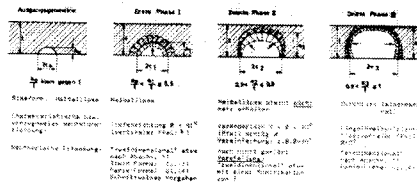
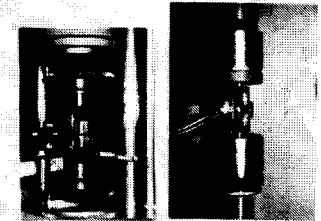
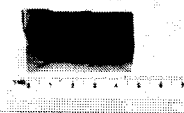
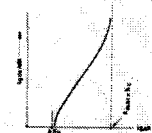


Fig. 10. Darstellung des im Modell des zweidimensionalen Risswachstums.

Korrelation ?

Die Korrelationsfunktion als Funktion der Zeit ist in der Abbildung dargestellt.



PROJECT TITLE : FRACTURE BEHAVIOUR OF STRUCTURAL STEELS	CSNI INDEX NUMBER : 11.2.1
SPONSORING COUNTRY : UNITED KINGDOM	ORGANISATION : REML RISLEY
DATE INITIATED : 1969 DATE COMPLETED :	PROJECT LEADER : DR 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 relationships between small scale fracture toughness tests and full scale experimental pressure vessel tests.

ii. to assess the applicability of COD and J data at temperatures where LEFM methods are invalid.

iii. to determine the effect on fatigue crack growth rate of mixed mode (I/II) loading.

iv. to develop a laboratory test to examine factors influencing crack initiation (and arrest) under thermally induced shock loading.

3. Experimental Facilities

Equipment for failure tests on 1.5 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. tests on 1.5 m dia pressure vessels containing partial thickness and full thickness defects have been completed. Fracture toughness data from small scale fracture mechanics test pieces are being compared with vessel failure data.

ii. the influence of test piece geometry (bend vs tension) on COD and J has been assessed using HY130 steel. The programme is being repeated using A533B steel. Particular attention is focussed on measurement of initiation toughness and its relevance to plant operation in the upper shelf regime.

iii. tests are in progress using angle notched centre cracked plates and modified compact tension test pieces to establish the influence of shear

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PROJECT TITLE :	CSNI INDEX NUMBER :
SPONSORING COUNTRY :	ORGANISATION :
DATE INITIATED :	PROJECT LEADER :
DATE COMPLETED :	

Description :

loading on fatigue crack growth rates. Test specimens are 12.5 and 25 mm thick low alloy pressure vessel steel.

iv. the essential features of thermal shock induced crack initiation and arrest are being studied by applying a known thermal shock to statically loaded three point bend test pieces. Preliminary calculations have been made and a series of test pieces are being prepared.

Reference Documents

Internal documents.

PROJECT TITLE : MECHANICS OF PLANE STRAIN FRACTURE	CSNI INDEX NUMBER : 11.2.1
SPONSORING COUNTRY : UNITED KINGDOM	ORGANISATION : RFL SPRINGFIELDS
DATE INITIATED : 1972 DATE COMPLETED :	PROJECT LEADER : DR B TOMKINS

Description :

1. General Aim

To recognise and assess modes of failure of nuclear reactor pressure vessels.

2. Particular Objectives

To measure the deformation in a thick cracked plate up to the point of fracture.

3. Experimental Facilities and 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 studied under well defined boundary conditions. Direct surface strain measurement around the crack tip will be done and fractography employed to stress fracture development prior to fast fracture propagation.

Reference Documents

Internal documents.

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PROJECT TITLE : METAL FRACTURE (CRACK GROWTH)	CSNI INDEX NUMBER : 11.2.1
SPONSORING COUNTRY : UNITED KINGDOM	ORGANISATION : RFL SPRINGFIELDS
DATE INITIATED : 1971 DATE COMPLETED :	PROJECT LEADER : DR B TOMKINS

Description :

1. General Aim

To recognise and assess modes of failure of nuclear reactor pressure vessels and other circuit components.

2. Particular Objectives

To examine the possibility of component defects growing to a critical size as a result of reactor transients.

3. Experimental Facilities and Programme

A number of special purpose machines have been assembled, which permit specimens of about 1 in section to be stress cycled and thermally cycled under well-controlled conditions.

4. Project Status

a) Crack growth under thermal shock loading. Crack propagation from short initial surface defects has been monitored in specimens which undergo repeated surface cooling shocks of between 100°C and 250°C. The crack growth rates and varying degrees of crack penetration into the material have been analysed using the basic elastic and elastic-plastic crack growth data derived under simple push-pull conditions in an earlier part of the programme. The tests are being carried out on types 304 and 316 stainless steel. The information from these tests is being used to assess the growth of under-clad cracks as a result of service transients and the behaviour of defects in reactor circuits. Further tests will examine the growth of cracks in combined thermal and mechanical loading situations and a new rig will impose heating shocks on to cracked specimens. 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

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PROJECT TITLE :	CSNI INDEX NUMBER :
SPONSORING COUNTRY :	ORGANISATION :
DATE INITIATED : DATE COMPLETED :	PROJECT LEADER :

Description :

cont/d

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.

Reference Documents

Internal documents.

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PROJECT TITLE : METAL FRACTURE (STABILITY)	CSNI INDEX NUMBER : 11.2.1
SPONSORING COUNTRY : UNITED KINGDOM	ORGANISATION : RFL SPRINGFIELDS
DATE INITIATED : 1971 DATE COMPLETED : 1976	PROJECT LEADER : DR B TOMKINS

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 (6000bar) may be employed and cycles of about half that magnitude. This has been applied to locally fatigue crack the base of a notch and thus produce well-defined crack starter conditions.

Reference Documents

Internal documents.

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PROJECT TITLE : VESSEL FAILURE TESTS	CSNI INDEX NUMBER : 11.2.1
SPONSORING COUNTRY : UNITED KINGDOM	ORGANISATION : REML RISLEY
DATE INITIATED : 1969 DATE COMPLETED :	PROJECT LEADER : DR A COWAN

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.

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PROJECT TITLE: A Dynamic Photoelastic Investigation of Crack Arrest	INDEX NUMBER: 11.2.1
SPONSORING COUNTRY: USA	ORGANIZATION: University of Maryland
DATE INITIATED: 9/74 DATE COMPLETED: Continuing STATUS AS OF:	PROJECT LEADER: J. W. Dally

Description:

1. General Aim: To determine appropriate characterization methods for crack arrest in terms of a measurable physical property of the material, development of test methods and specimens for standardized crack arrest toughness measurements, and adoption by the ASME code and regulatory authorities for design and operation of LWR primary system components.
2. Particular Objectives
 - a. Program Management

Efficiently manage the program, monitor subcontracts, procure special materials and equipment, issue progress and topical reports, and participate in ASTM meetings and others as required. Maintain close coordination with the other participants in the NRC/EPRI crack arrest program, in addition to participation in regular meetings.
 - b. Verification of Tentative Standard Test Procedures

Test models of each of the three tentative standard test specimens. Observe specimen-dependent dynamics and make recommendations for change or optimization, where applicable. Investigate the effects of material damping.
 - c. Birefringent Coating Applications

Utilize this testing method to evaluate steel specimens, and to learn of any differences between plastic and steel specimen mechanics. Develop the necessary high speed photographic systems required of this method. Use this method as early as possible on the three tentative standard test specimens, in spite of the recognized problem from side grooves.

d. Thermal Stress Simulation

Develop a mechanical simulation of thermally induced stresses in a ring. Perform crack initiation and arrest studies in this geometry, studying the dynamics, and ascertaining whether a static analysis would correctly or conservatively predict the extent of crack penetration. Report the results.

e. Weld Line Simulation

The influence of the adhesive line in duplex specimens may closely model the weld line in a structure. Complete the investigation of the effects of adhesive toughness and thickness to arrive at a meaningful analysis methodology of a running crack entering and going through a weld. This is imperative to an understanding of duplex specimens, and to the HSST test of ITV-7C, so these results must be closely coordinated with the other participants and the HSST program.

f. Two-Dimensional Dynamic Analysis

- Develop a numerical code for crack propagation along a straight-line path in a two-dimensional body.

g. Consultation

Provide consultation services, particularly of Dr. Irwin, to the NRC as required.

3. Experimental Facilities and Program:

- a. Facilities: Testing laboratories at the University of Maryland.
- b. Program: See item 2 above.

4. Project Status:

- a. Results to date: Arrest characterization in brittle and ductile materials using specimens, including correlation of analysis and experimental methods for K-values.
- b. Essential results: To provide a characterization of crack arrest in the various specimens using both brittle and tough materials.

5. Next Steps: See item 2 above.

6. Relation to Other Projects: Part of NRC/EPRI Program on Crack Arrest.

7. Reference Documents: Quarterly Progress Reports and Topical Reports.
8. Degree of Availability: All Available From NTIS.

PROJECT TITLE: Structural Integrity of Water Reactor Pressure Boundary Components	INDEX NUMBER: 11.2.1
SPONSORING COUNTRY: USA	ORGANIZATION: Naval Research Lab.
DATE INITIATED: 1972 DATE COMPLETED: Continuing STATUS AS OF:	PROJECT LEADER: F. J. Loss

Description:

1. General Aim

To provide analytical and experimental methods and data necessary to assure fracture prevention in LWR primary pressure boundary components. Specifically included are criteria for fracture toughness; fatigue crack growth studies and experiments; irradiation effects, including degradation recovery by post-irradiation heat treatment and sensitivity to reirradiation; and thermal shock studies, including the effects of warm prestressing and a plastic net ligament.

2. Particular Objectives

a. Fracture Toughness Criteria

Develop a dynamic testing practice for the bend beam, both analytically and experimentally. Utilize the finite element method to optimize the location of strain gages for the testing and analysis. Submit to ASTM E-24. Obtain a data base to verify code K_{ID} from dynamic testing. Report these results. Evaluate the structural significance of small specimen tests, characterize low ductile shelf RPV steels in fracture mechanics terms, and correlate C_V to K_{IC} on ductile shelf. Develop the J-integral testing method for both static and dynamic loading for single specimen evaluations. Provide consultation as required, especially on the low ductile shelf, irradiated, material toughness characterization.

b. Cyclic Crack Growth Rate Evaluation

Expedite the on-line performance of the larger capacity autoclave systems. Complete the preliminary matrix and assist in the development of and initiate the main CCGR testing program. More emphasis should be put on the evaluation of welds, both unirradiated and irradiated, in the main program. Automate the data acquisition where possible.

c. Irradiation Sensitivities and Degradation Recovery

Complete the NRC-CE-NRL irradiation effects program and report the results. Determine both the ductile shelf level and the transition temperature shift response to post-irradiation heat treatment, optimizing the heat treatment, and systematically isolate the primary metallurgical and compositional factors governing recovery behavior and reirradiation sensitivity. Initiate an irradiation, anneal, and reirradiation program on materials of primary concern and complete in FY78. Continue support to the 4T-CT irradiation programs via C_v tests to set fluence level. Continue to aid in the updating of regulatory guide 1.99 by providing data on transition temperature shift and drop in ductile shelf energy due to irradiation.

d. Thermal Shock Related Investigations

Complete the effort to fully characterize the beneficial effects of warm prestressing. Perform a statistically designed program to minimize the difference between the warm prestressing temperature and the isothermal-load-to-failure temperature and report the results. Cooperate in NRC-sponsored round robin testing for crack arrest toughness measurement. Conduct further plastic net ligament tests at ductile shelf temperature with low ductile shelf material and within the transition temperature regime. Provide a technical basis for disregarding biaxiality of stresses during warm prestressing.

3. Experimental Facilities and Program:

- a. Facilities - Testing laboratories at NRL and subcontracted irradiation facilities.
- b. Program - See item 2 above.

4. Project Status:

- a. Results to date: Dynamic testing variables have been defined, including specimen dynamic response. Test methods development and some data on cyclic crack growth rate evaluation. The NRC-CE-NRL cooperative program is nearly completed. Warm prestressing and plastic net ligament benefits to thermal shock analysis demonstrated.
- b. Essential Results: See item 1 above.

5. Next Steps: See item 2 above.

6. Relation to Other Projects: This program is part of a coordinated effort to assure the safe operation of nuclear power reactors.
7. Reference Documents: Quarterly Progress Reports and Topical Reports.
8. Degree of Availability: From NRL and NTIS.

11.2.2. Stress-strain analysis

PROJECT TITLE : SHORT RADIUS FLANGED PIPE BENDS	CSNI INDEX NUMBER : 11.2.2
SPONSORING COUNTRY : AUSTRALIA	ORGANISATION : School of Nuclear Engineering University of New South Wales
DATE INITIATED : March 1975 DATE COMPLETED :	PROJECT LEADER : Prof. J.J. Thompson

Description :

Joint Project, School of Nuclear Engineering and Engineering Research Division, Australian Atomic Energy Commission Research Establishment.

Experimental work on the flexibility of short radius flanged pipe bends, to study the effect of the flanges, is complemented by theoretical shell analysis and use of finite element codes. Progress has been made on the formulation and numerical determination of the decay of self-equilibrating end-load systems in the unflanged pipe, which will be utilised for the overall system analysis to provide engineering data on in plane bending of flanged pipe bends.

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14.560

PROJECT TITLE : Non-Symmetric Stresses in Heat Exchangers	CSNI INDEX NUMBER : 11.2.2
SPONSORING COUNTRY : Canada (Atomic Energy Control Board)	ORGANISATION : Ecole Polytechnique de Montreal
DATE INITIATED : March 1973 DATE COMPLETED : February 1977	PROJECT LEADER : Dr. J. Dubuc

Description :

1. General Aim

- 1) To assess the degree of conservatism in heat exchanger designs when non-symmetric thermal loads are more rigorously examined
- 2) To develop a more realistic numerical analysis technique for the treatment of non-symmetric stresses in heat exchanger design

3. Experimental Facilities and Programme

- 1) Carry out an experimental stress analysis on a nuclear heat exchanger tube sheet and channel cover subjected to non-symmetric thermal loading.
- 2) Compare measured values of strains, deflections and rotations with results obtained by numerical analysis of the experimental models.
- 3) Evaluate the simplifying assumptions used in standard calculations to indicate if assumptions provide a good approximation to real stresses.
- 4) Develop more realistic assumptions or correction factors which will enable better designs.

4. Project Status

The results of an experimental stress analysis on the tube sheet of a nuclear heat exchanger were compared with the predictions of models in which axisymmetric conditions were assumed. As a result, an assessment of the effect of the non-axisymmetric stresses was possible enabling the models to be modified to permit a more accurate simulation of the observed behavior. The theoretical and experimental phases of the project were completed in the third quarter of 1976. The final report was received and approved in the first quarter of 1977. This project is now complete.

5. Reference documents

Dubuc, J., Hamerski, B. and Bazergui, A.
Etude Experimentale des Contraintes Thermique d'une Plaque a Tubes"
Rapport Technique EP-76-R-25, Ecole Polytechnique de Montreal, Sept. 1976.

9. Budget

~ \$83,000

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14.560

PROJECT TITLE : A Study of the Flaw Sensitivity of Pipe-to-Pipe Intersections	CSNI INDEX NUMBER : 11.2.2
SPONSORING COUNTRY : Canada (Atomic Energy Control Board)	ORGANISATION : University of Waterloo, Ontario
DATE INITIATED : July 1972 DATE COMPLETED : April 1978	PROJECT LEADER : Prof. D.J. Burns

Description :

1. General Aim

To develop analytical and experimental techniques of assessing the rate of growth of cracks that propagate from pre-existing flaws in pipe-to-pipe intersections subjected to various loading and environmental conditions.

3. Experimental Facilities and Programme

- i) Extend existing stress-strain analyses to additional types of pipe-to-pipe intersections and to validate these analyses by comparison with experimental results.
- ii) Derive crack tip stress intensity factors via these stress-strain analyses and to correlate the results with measured crack growth data.
- iii) Determine the effect of load frequency, size and location; flaw location and size; and environment on the rate of crack growth in piping tees.

4. Project Status

The experimental work has concerned itself with the measurement of crack propagation rates in pipe-tees (designed to ASME Section III, Class I requirements but with artificial internal flaws) subjected to cyclic internal pressure and/or bending at temperatures near 260°C. The pressurizing medium is Dowtherm A, requiring a parallel investigation of the effect of Dowtherm A vs. water on crack growth rates. The flaws are introduced at locations near the weld at the pipe juncture which have been shown to have high stress levels by an experimental strain analysis. Results to date indicate that these pipe tees are quite tolerant to internal flaws (expected cycling under reactor service conditions would not lead to failure). All failures to date have demonstrated the "leak before break" safety criterion.

The theoretical work initially attempted to develop three-dimensional stress analyses using classical analytical techniques. These have met with only limited success. Solutions have been developed for less complex intersections and checked experimentally but the pipe-to-pipe tee has not

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Telex : 63668 AEN NEA - Tel : 524 82-00

PROJECT TITLE : A Study of the Flaw Sensitivity of Pipe-to- Pipe Intersections	CSNI INDEX NUMBER : 11.2.2
SPONSORING COUNTRY : Canada (Atomic Energy Control Board)	ORGANISATION : University of Waterloo, Ontario
DATE INITIATED : July 1972 DATE COMPLETED : April 1978	PROJECT LEADER : Dr. D.J. Burns

Description :

Continuation

yet been successfully analysed. Present work indicates that finite element techniques for the area near the crack superimposed on a general analytical solution of the pipe-tee may be the most promising method.

Both theoretical and experimental programs are now dovetailing into a study of the stress near the crack tip and the determination of stress intensity factors leading to a correlation of these with crack growth rates. Early results of comparative Dowtherm A and water crack growth rates indicate that the Dowtherm rates are higher. Consequently, endurance figures determined during the Dowtherm tests should be conservative.

5. Reference Documents

Redekop, D.

"Three-Dimensional Analysis of an Elastic Plate-Cylinder Intersection by the Boundary-Point-Least-Squares Technique", ASME Journal of Press. Vess. Tech. Paper No. 76-PVP-34.

Redekop, D.

"Three-Dimensional Elastic Analysis of an Intersection of a Spherical Shell with a Cylindrical Shell", Int. Jour. of Pressure Vessels and Pipings, to be published.

Spek, E.J.

"Low Cycle Fatigue of ASTM A106B STEEL, in Silicone Oil at Elevated Temperature and Pressure, Master's Thesis, University of Waterloo, Waterloo, Ontario 1976

9. Budget

\$200,000 - \$250,000

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Telex : 63668 AEN NEA - Tel : 524 82-00

14.560

Classification
11.2.2/11.2.3

<u>Title 1</u>		COUNTRY Denmark
Spændingsanalyse af primære trykbærende stålkompnenter: Sammenligning mellem beregnede og målte tøjninger og spændinger i en BWR pumpestuts.		SPONSOR DAEC, Risø
		ORGANIZATION
		DAEC, Risø
<u>Title 2</u>		<u>Project leader</u>
Stress analysis of primary steelcomponents: A comparison between calculated and measured stress and strain in a BWR main circulation pump nozzle.		S.I. Andersen
		Scientists:
<u>Initiated</u> (date)	<u>Completed:</u> 1977	S.I. Andersen
74.04.01		
<u>Status:</u> 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 calcula-

tions (2 D) has been compared to the 3D- and experimental results.

5. Next steps

None

6. Relations with other projects The work is related to

- 1) Risø investigations of the validation of structural computer codes
- 2) Risø work on the safety of primary pressure system.

7. Reference documents

- [1] S.I. Andersen, J. Reynen, P. Engbæk:
"Stress Analysis of a Main circulation Pump Nozzle".
Risø-TPM-76/1, Jan. 1976.

- [2] S.I. Andersen, T. Henriksson, J. Reynen:
"Stress Analysis of a MCP Pressure Vessel Nozzle".
Paper G 8/d to be published at the 4th Int. Conf.
on Structural Mech. in Reactor Tech., San Francisco 1977.

8. Degree of availability A limited amount of the results may be freely available.

11.2.3. Non-destructive testing, inspection,
surveillance

Classification: 11.2.3.	
Title: Akustisk Emission fra stålkonstruktioner	Country: DENMARK
	Sponsor: ECSC
Title: Acoustic Emission from steel structures	Organization: Res. Establ. Risø
Initiated date: 1977 Status: In progress	Completed date: Scientists: A. Nielsen W. Swindlehurst C. Debel

1. General aim: To provide a basis for the evaluation of AE signals obtained during surveillance of steel structures.
2. Particular objectives: Weld metals and heat affected zones will be given particular attention.
3. Experimental facilities and programme: Testing machines and electronic analysis apparatus are available. A facility for pressure testing of vessels is being designed. Laboratory investigations on particular test pieces are planned to compare amount and particular features of the AE produced by different materials (Base metals, weld metals, HAZ) at different conditions (Heat treatment residual stress) under different testing conditions (Temperature, load configuration). Experiments on intermediate size vessels are then planned to investigate to what extent the results gained from the small test pieces are applicable to steel structures.
4. Project status: The testing facilities are ready for laboratory experiments to start.
5. Next steps: The observations from the laboratory experiments will form basis for a programme for vessel testing.
6. Relation with other projects: This project is related to industrial projects on application of the AE technique.
7. Reference documents:
 RISØ-M-1896. Arved Nielsen "AE surveillance Methods. AE as a supplementary tool for inspection of reactor pressure components".
8. Degree of availability: As decided by the sponsors (European Coal and Steel Community, Danish Teknologirådet).

PROJECT TITLE : QUALITY ASSURANCE	CSNI INDEX NUMBER : 11.2.3 AND 12
SPONSORING COUNTRY : FINLAND	ORGANISATION TECHNICAL RESEARCH CENTRE OF FINLAND REACTOR MATERIALS RESEARCH
DATE INITIATED : 1970 DATE COMPLETED : CONTINUES	PROJECT LEADER : JYRKI JALASTO

Description :

1. GENERAL AIM

To survey quality assurance requirements and methods for the design, construction, installation and operation of structures, components and systems.

2. PARTICULAR OBJECTIVES

- Quality assurance audits of utilities and contractors
 - New non-destructive testing methods
 - Quality assurance plans, manuals, instructions and procedures
- The main emphasis is on the quality assurance for the design, construction and installation of mechanical components.

3. PROJECT STATUS

The audits of the quality assurance systems for power companies have been performed. Application of acoustic emission in research work and in-service inspection has been evaluated. Investigation of the ultrasonic inspection method and its applicability to austenitic welds and castings is under way.

4. REFERENCE DOCUMENTS

Reactor Materials Research reports (partly confidential), conference papers etc.

5. DEGREE OF AVAILABILITY

Upon request from the Technical Research Centre of Finland.

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Telex : 630668 AEN NEA - Tel : 524 82-00

<u>Classification: 11.2.3</u>								
<u>Title 1 (Original Language):</u> Zerstörungsfreie Wiederholungsprüfung an Reaktor- druckbehältern mittels Wirbelstromverfahren (RS 89 - II.3.2; Jahresbericht A 76)	COUNTRY: BRD							
	SPONSOR: BMFT							
	ORGANIZATION: KWU Erlangen							
<u>Title 2 (English):</u> Non-Destructive In-Service Inspection for Reactor Pressure Vessels with Eddy Current Methods	Project Leader: Gräbener							
	<table border="0" style="width: 100%;"> <tr> <td style="width: 50%;"><u>Initiated (Date):</u></td> <td style="width: 50%;"><u>Completed (Date):</u></td> </tr> <tr> <td>1. 12. 72</td> <td>30. 6. 76</td> </tr> <tr> <td><u>Status:</u></td> <td><u>Last Updating (Date):</u></td> </tr> <tr> <td>Completed</td> <td>31. 12. 76</td> </tr> </table>	<u>Initiated (Date):</u>	<u>Completed (Date):</u>	1. 12. 72	30. 6. 76	<u>Status:</u>	<u>Last Updating (Date):</u>	Completed
<u>Initiated (Date):</u>	<u>Completed (Date):</u>							
1. 12. 72	30. 6. 76							
<u>Status:</u>	<u>Last Updating (Date):</u>							
Completed	31. 12. 76							

1. General Aim and 2. Particular Objectives

The eddy current method was to be qualified for the examination of cracks on the surface of clad reactor pressure vessels. Appropriate conditions, detectors and procedures of registration for this method were developed. The limits for detecting flaws on sections under water have been investigated.

3. Research Program and 4. Experimental Facilities

The program was divided into two main parts:
 In a pre-program first tests were carried out with existing detectors and evaluation equipment concerning the detection of flaws depth, considering thereby possible interference effects (detectorsurface distance, material inhomogenities). The pre-program showed which detectors and equipment were the most appropriate. In the main program appropriate detectors and evaluation equipment were developed according to the know-how obtained in the pre-program. With this prototype experiments were carried out under defined conditions. Data registration

and processing were equally developed, as well as a system for carrying the detectors.

5. Progress to Date and 6. Results

The documentation of the results has been completed. The measuring technique was improved, visual inspection device and simple recording were not sufficient for rational inspections. Automation will be necessary.

7. Next Steps

The work has been completed.

8. Relation with Other Projects

RS 27 "Development of Non-Destructive Inspection Techniques for In-Service Inspection Tests of Reactor Pressure Vessels"

9. References

Devrient, et.al.

"Zerstörungsfreie Wiederholungsprüfungen am RDB mittels Wirbelstromverfahren (Teilvorhaben 1 - 4)".

Abschlußbericht Förderungsvorhaben RS 89

Kraftwerk Union Aktiengesellschaft, Erlangen (Okt. 1976)

10. Degree of Availability

The report is company confidential.

<u>Classification:</u> 11.2.3	
<u>Title 1 (Original Language):</u> Akustische Holographie - Labor- und anwendungstechnische Versuche mit dem Holscan 200 (RS 102-20/2.- II.3.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR:BMFT ORGANIZATION: Fraunhofer-Ges. IzfP, Saarbrücken
<u>Title 2 (English):</u> Acoustical holography - laboratory and application tests with the Holscan 200	<u>Project Leader:</u> Dr. V. Schmitz
<u>Initiated (Date):</u> 1.8.1975 <u>Status:</u> Continuing	<u>Completed (Date):</u> 28.2.1977 <u>Last Updating (Date):</u> December 1976

1. General Aim

In the non-destructive testing they have not yet succeeded in reaching an identical, three-dimensional picture of flaws, respectively of greater areas with defects by conventional ultrasonic methods. An exact knowledge of these flaws is necessary to determine the influence of such areas on the stability or safety of reactor pressure vessels. The acoustical holography links the conventional method of pulse-echo with the attitudes of optical holography and allows a non-destructive testing and a geometrically identical projection of flaws concerning either the basical testing or the repetitional testing. In the near future, testing procedures with the aid of acoustical holography should belong to the technical standard.

2. Particular Objectives

The acoustical holography can help to determine the exact size and locus of flaws in thick-walled vessels. We are working at two kinds of application:

- testing of the basic material, the cladding and the interface between austenitic and ferritic junctions of reactor pressure vessels;
- testing of thick-walled components both in laboratory tests and in situ.

3. Research Program

The experiments can be divided into the following points:

- 3.1. Installation of an acoustical holographic equipment,
- 3.2. Experiments on specimens with artificial flaws,
- 3.3. Experiments on specimens with natural flaws.

4. Experimental Facilities, Computer Codes

From the methodological point of view ultrasonic holography is better suited to the examination of thick-walled materials instead of using focal probes as it works with the far-field length of a focal probe. The focal point in this case does not lie at the level of the flaw but can be set at the surface independently of whether the depth of the flaws is known in advance or not. All flaws reflected in the area of the ultrasonic beam will be picked up regardless of their distance from the surface and shown in the reconstruction phase.

Fig. 1 shows the principle. A point source in the surface is emitting a spherical wave and detects the reflected ultrasound, too. To conserve the information of the phase, there is a second wave, the reference wave with an inclination angle. This method has been realized in the holographic equipment holscan 200 of Holosonics, Inc., shown in fig. 2. On the left the tank filled with water containing the material to be examined. Above the mechanical scanner which moves a focal probe in a grid pattern. The ultrasonic signals converted into electronic signals are processed in the electronic processor. To enable the retention not only of the amplitude but also of phase a coherent phase-shifted reference signal is superimposed on the flaw signal. Thus a diffraction, the hologram, is formed which can be made visible on a viewing screen. In the optical processor on the right of the picture the negative produced by the screen is illuminated by a laser beam. The image of the flaws appears in the monitor where it can be measured.

5. Progress to Date

The following examples may serve to give some ideas of the potential of ultrasonic holography.

The test specimen shown on the right in fig. 3 is made of steel and has at four different levels in each case three flat bottom holes with a diameter of 3 mm and at distances from each other of 2, 3 and 4 mm. A 4 MHz transducer was used with a nearfield length in water of 18 inches.

The reconstructed flaws are shown on the left. Even the 3 mm flaws at a depth of 180 mm from the surface can be clearly distinguished and reconstructed. One advantage of holography is that it can employ the single probe technique as well as that the flaw lies within the widely spread beam. So shown in fig. 4, the holes with 30° and 45° slope could be recognized with the correct diameter.

In the next example the dependence of the flaw inclination with transverse wave beams was investigated. The flaws diverge from the acoustic axis (45° to the normal) by up to $\pm 15^\circ$. Nevertheless, all the flaws were unambiguously shown up (fig. 5).

In the following example we used a double clad specimen. The claddings were 2.4 inch wide and 1.2 inch offset against each other. The testing frequency was 1.25 MHz. As anticipated a markedly inferior reproduction of the reflector was the result. There remained however, a noteworthy and promising resolving power and sensitivity (fig. 6).

We have investigated systematically a test piece with a weld. The weld was insonified with longitudinal waves from both sides, that means in 5 inch depth, resp. in 10 inch depth. The length of the weld is about 20 inches, the height about 1 inch and the width more than 10 inches. In addition to the manual testing we have investigated the test piece with a focal probe 4 MHz, longitudinal wave and a diameter of 60 mm. Fig. 7 represents a C-scan image of the whole weld, the ultrasonic hologram and the reconstruction of the flaws. The testing frequency was 2.4 MHz. The holographic reconstruction gives a more detailed image of the flaw and allows to measure immediately the size of the flaws.

6. Results

Ultrasonic holography has shown itself to be the most promising area of testing thick walls. In all cases of experimental results the resolution capability of holography has been demonstrated.

7. Next Steps

A potential development lies in the replacement of the optical processing by a numerical processing scheme (fig. 8). With a numerical reconstruction the inspection data is not presented as a negative but fed into the core store of a computer via an analog-digital converter. The reconstruction then takes place numerically using a fast Fourier transformation and presented on a monitor or a print out. First

experiments on a specimen shown in fig. 9 demonstrate that we can get reasonable good results - fig. 10 - in performing the numerical reconstruction. The advantages of numerical reconstruction are that it permits interferences and distortions to be removed and a free choice in the manner of representations.

8. Relations with Other Projects: none.

9. References

V. Schmitz:

Studie zur Anwendung der akustischen Holographie bei der zerstörungsfreien Prüfung von Reaktorkomponenten und Reaktoranlagen.

Teil I: Literaturrecherche

Izfp-Bericht 740106-TW, Saarbrücken 1974

C. Schmitz:

Eignung des Holscan 200 der Firma Holosonics zur Fertigungs- und Wiederholungsprüfung an Reaktorkomponenten und Reaktoranlagen.

Teil II: Prüfsystemanalyse.

Izfp-Bericht 750207-TW, Saarbrücken 1975

10. Degree of Availability of the Reports

On request by GRS, Köln.

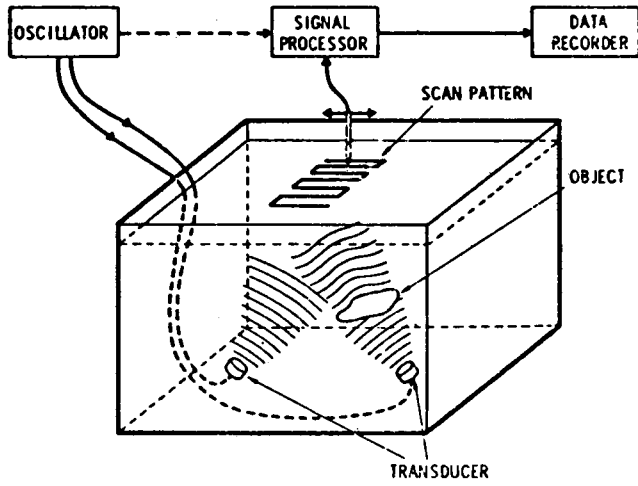


Fig. 1 : Principle of US-holographic

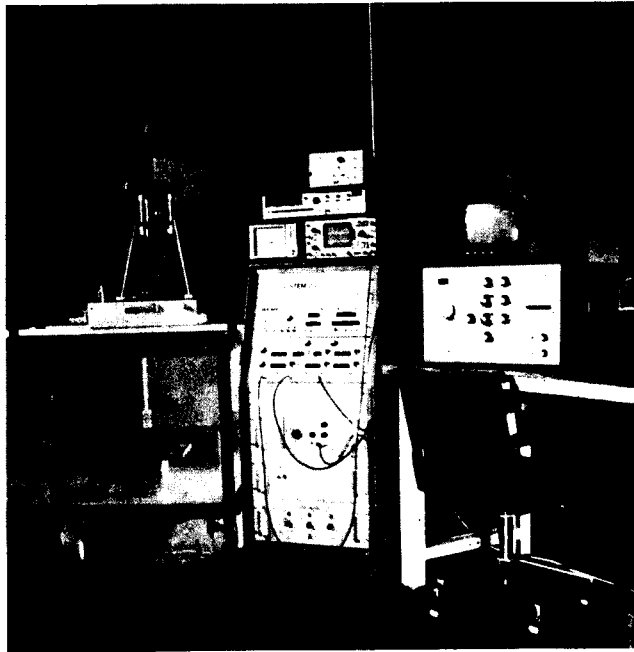


Fig. 2: Holographic equipment Holscan 200 of Holosonics, Inc.

Prüffrequenz : 4 MHz
Schwingerdurchmesser : 50 mm
Nahfeldlänge in
Wasser : 460 mm

Fig. 3:
Testspecimen with
flat bottom holes

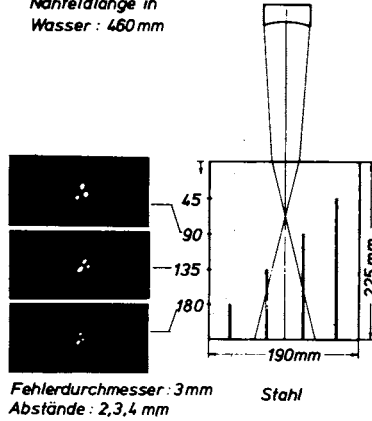
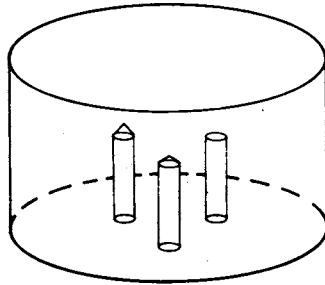


Fig. 4 a/b:
Testspecimen with
holes with 30°
and 45° slope



Mat: Stahl
190 mm
Dicke: 100 mm
Bohrungen: # 10 mm
Tiefe: 50 mm
Dachwinkel: 0°, 30°, 45°



Hologramm



Rekonstruktion

Bohrungen: 10 mm ; Dachwinkel: 0°, 30°, 45° Tiefe: 50 mm

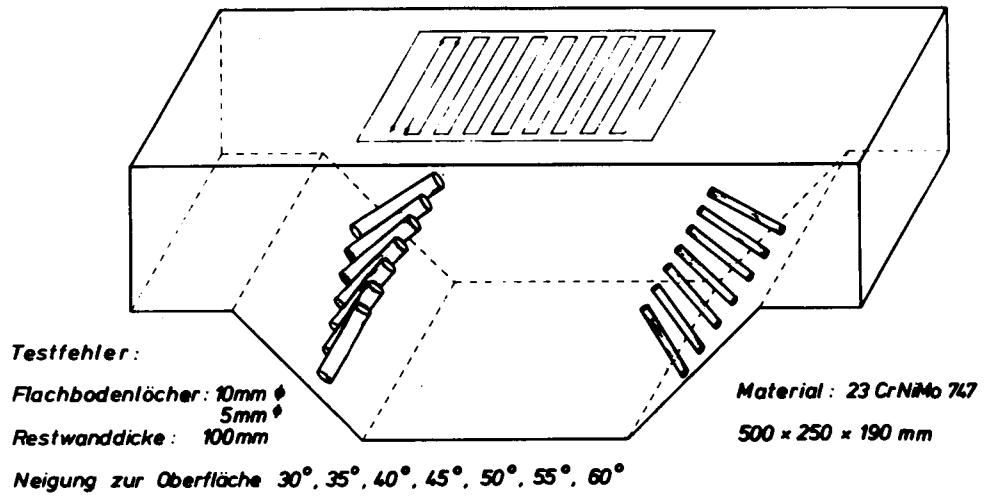


Fig. 5 a/b:
Investigation of the dependence of the flaw inclination with transverse wave beams

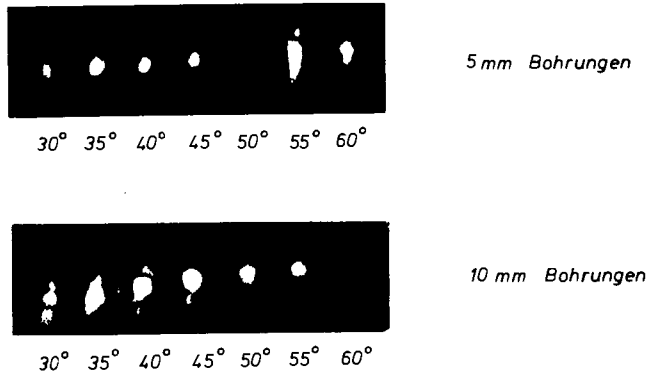
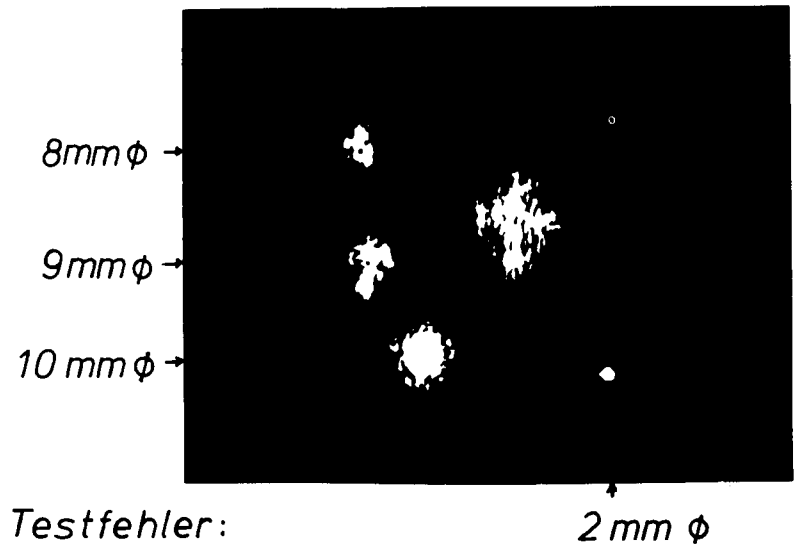


Fig. 6:
Investigation of a double clad specimen
(f = 1,25 MHz)



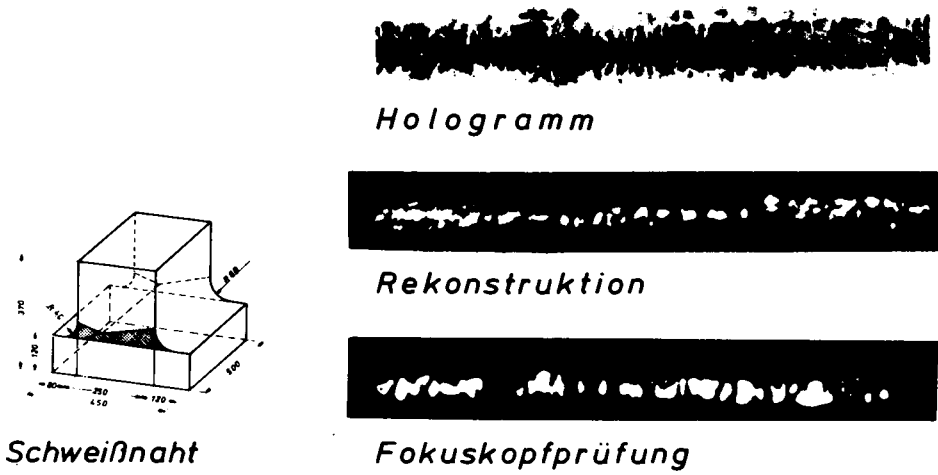


Fig. 7: C-scan image of the whole weld, the ultrasonic hologram and the reconstruction of the flaws.

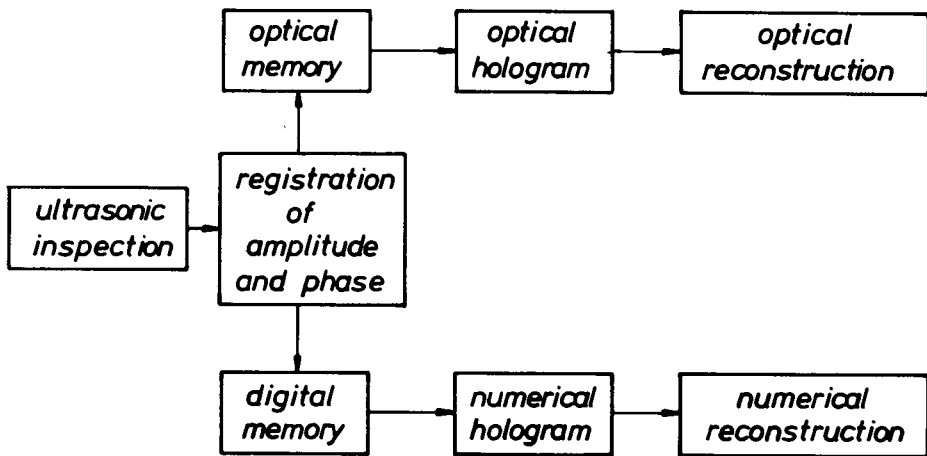


Fig. 8: Scheme of optical and numerical reconstruction.

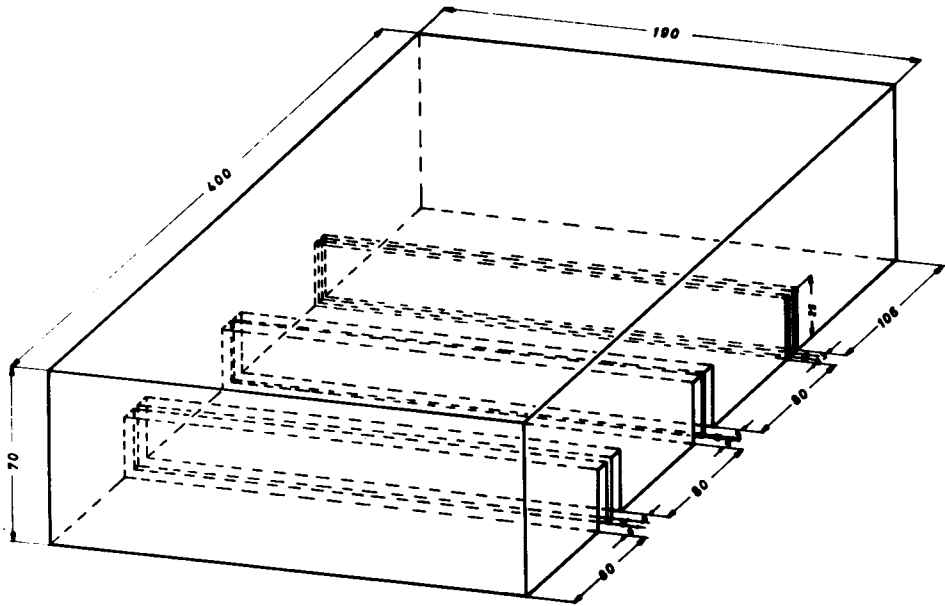


Fig. 9: Specimen for testing numerical reconstruction.

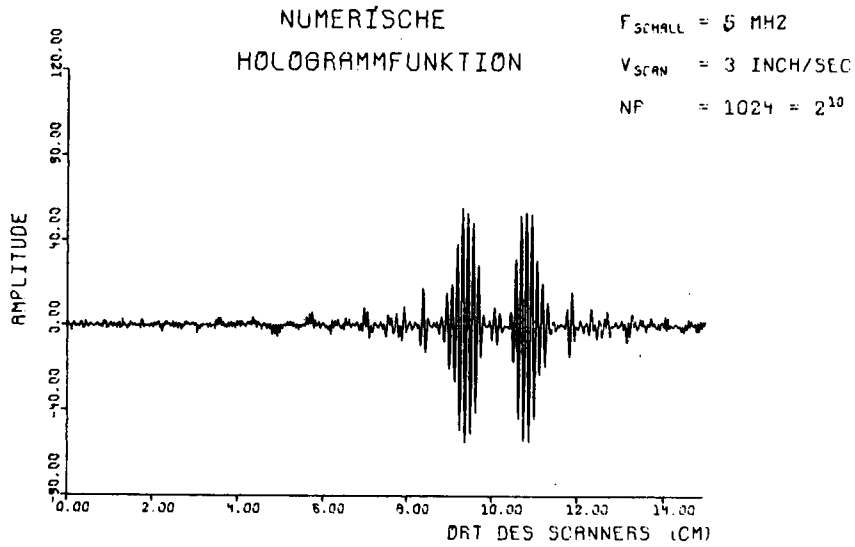


Fig. 10a: Hologram of the specimen of fig. 9

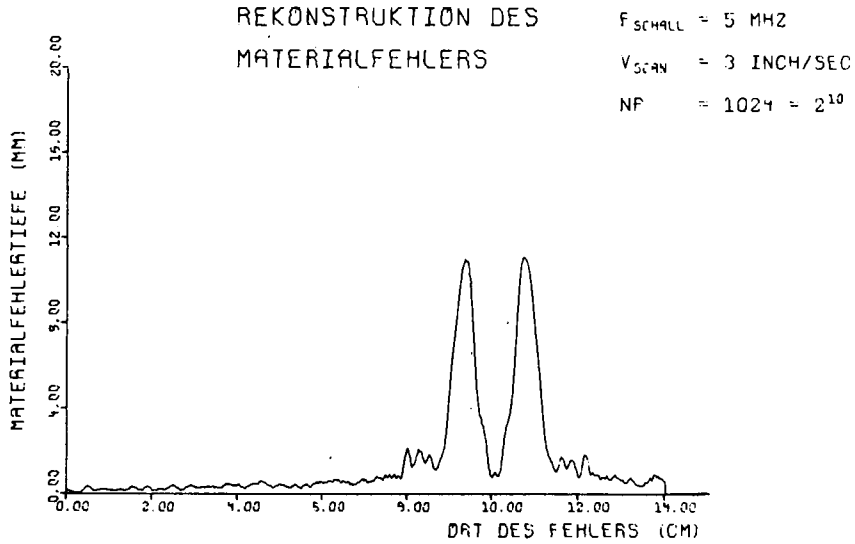


Fig. 10b: Reconstructed image (numerical reconstruction)

<u>Classification: 11.2.3</u>	
<u>Title 1 (Original Language):</u> Durchführung von Untersuchungen zur Rißerkennung an druckführenden Reaktorbauteilen mit Hilfe der optischen Holographie (RS 132-II.3.2, Jahresbericht A76)	<u>COUNTRY:</u> BRD
	<u>SPONSOR:</u> BMFT
	<u>ORGANIZATION:</u>
<u>Title 2 (English):</u> Crack detection in pressurized vessels and reactor components by optical holography	<u>Project Leader:</u> Professor Dr.-Ing. F. Mayinger
<u>Initiated (Date):</u> July 1974 <u>Status:</u> Continuing	<u>Completed (Date):</u> June 1978 <u>Last Updating (Date):</u> December 1976

1. General aim

The aim of the activities is to develop a quick and reliable optical method in order to detect cracks in reactor vessels during fabrication and later in repeating tests.

2. Particular objectives

In the last few years a new nondestructive technique - holographic interferometry - was developed by which one can detect material defects. Up to now it was, however, only used for nonmetallic components.

Our aim is to apply this technique also for the testing of steel. The special objective of this investigation is to detect cracks in or near the surface of reactor vessel steels using surface waves which are recorded by double-pulse holography.

3. Research program

At first, basic experiments with testpieces of simple geometric shape are made. The aim of these preliminary experiments is to find out the smallest crack that can be measured with holographic interferometry in connexion with the impact excitation of the testpieces. The indication of cracks is influenced by various parameters:

1. crack-parameters (shape, size, orientation and position of the crack)

2. testpiece-parameters (geometric shape, thickness and material)
3. test-method-parameters (impact-parameters sensitivity of the method)

Further experiments shall be carried out with testpieces of any geometric shape. After these preliminary investigations experiments will be made with real reactor components like pipelines, elbows, T-pieces, etc. to proof the reliability of this new nondestructive test-method. For these measurements in a nuclear power plant during a repeating test the holographic set-up must be reconstructed to a compact and versatile device.

4. 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. This waves cause surface deformations which can be measured with holographic interferometry.

To produce the waves it is usually necessary to strike the testsection in a suitable manner. This impact excitation may be generated e.g. by a free falling steel ball, by the bullet of an air-gun, or any other stress wave generator. The local impact is only a short-time excitation after which the waves will travel in all directions. In order to record this very fast event holographically a Q-switched ruby laser is used which produces two short light pulses. The first giant pulse illuminates the testpiece shortly before the impact and thus the testpiece in its unstrained condition is recorded. The second laser pulse appears at a certain time after the impact to make visible the propagation of the wave. The whole experimental set-up was already discussed in the annual report A 74.

5. Progress to Date

The experiments with testpieces of simple geometric shape (plates) in connexion with the described technique (particular with the steel ball impact-generator) are finished. It is therefore possible to describe the detectable cracks as a function of different parameters. Furthermore, measurements were made with testpieces of more complex geometric shape like tubes and elbows. In addition, basic investigations were made to use a thin liquid layer above the surface of the testpiece as a signal amplifier.

This liquid layer levitates the amplitude of the generated surface waves. In fact due to the very small disturbance of waves at small cracks it is necessary to amplify this disturbance up to a range that can be measured optically.

Simultaneously, the holographic set-up was rebuilt into a more compact device. With the aid of a new electronic trigger-system it is now possible to switch on the two laserpulses exactly in comparison to the wave generating impact (fig. 1).

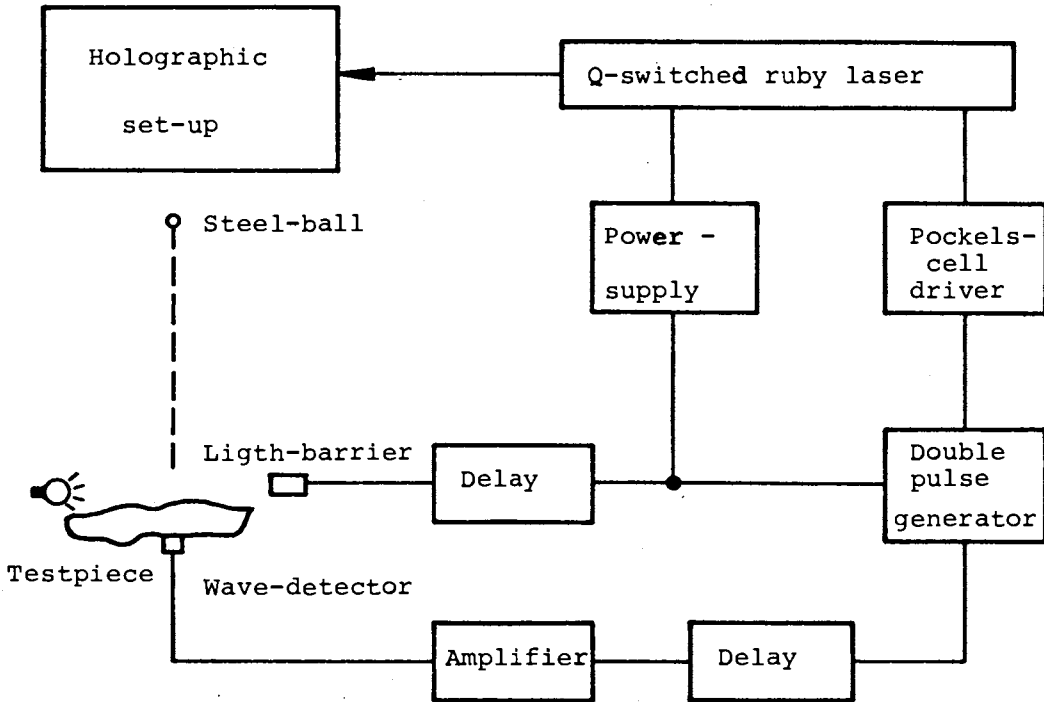


Fig. 1: Experimental set-up

6. Results

Theoretical and experimental investigations have shown, that the detection of very small cracks in connexion with the described holographic method is limited due to the fact that the wave disturbance is extremely small for crack geometries of interest. Without an adequate magnification of the surface motion or an increased sensitivity of the measuring equipment cracks down to a length of few millimeters and a depth greater than the half of the material thickness are always detectable (fig. 2).

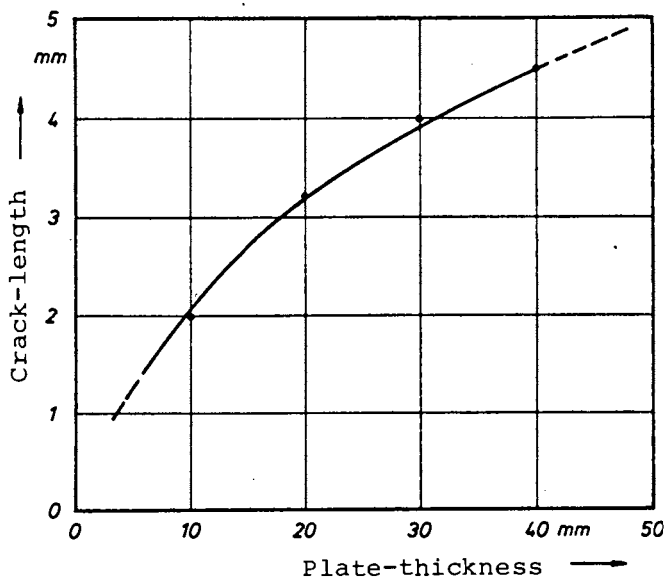


Fig. 2: Holographically detectable cracks without the aid of a liquid-layer (Impact generator: falling steel ball)

In addition it is difficult to obtain a reasonable surface deformation with testpieces thicker than 40 mm (see A 1975). The results with respect to the limited crack geometry and to the limited material thickness could be improved by the aid of a liquid or plastic layer placed upon the testpiece surface. This layer served in amplifying the extremely small wave distortion near small cracks up to a range that could be measured optically. It can be shown theoretically, that the levitation h of the liquid-air interface is

$$h = \frac{4\pi^2}{g} f^2 a_0^2$$

f : Frequency of the generated wave
 a_0 : Amplitude of the generated wave

With the aid of such a liquid-layer it is possible to detect disturbances of the waves resulting from very small material-voids.

The influence of the liquid layer can be seen in fig. 3 and fig. 4. The testpiece was a steelplate with a thickness of 40 mm. The surface-cracks were simulated by rows of 6 holes with a depth varying from 1 mm to 10 mm. One can see in fig. 3 that without the aid of a liquid-layer no distortion of the interference pattern are visible. But if the surface of the testpiece was covered with a thin water film, all cracks are clearly detectable (fig. 4). Similar results were obtained with a much more thicker steelplate, as it is shown in fig. 5 were a 85 mm plate with a liquid layer of water was tested. To examine another liquid-layer than water, experiments were made with gelatine on the surface of the 85 mm thick steelplate. The result is shown in fig. 6, were all cracks are surrounded by two or more interference-fringes. After these positive results experiments were made with even smaller cracks and as fig. 7 and 8 show, surface-cracks down to a length of .5 mm, a depth of only 0,5 mm and a width of 0,1 mm are clearly visible.

In addition, experiments were carried out with material voids which are located below the surface of the testpiece. As can be seen in fig. 9, these voids are also detectable with holographic interferometry. Because the disturbance of the wave at these cracks are big enough to be measured optically, it was not necessary to use a liquid-layer on the testpiece surface.

Further experiments with testpieces of more complicated geometric shape were made. The crack is located near the welding seam of the flange of an elbow. The interference pattern (fig. 10) show that the crack detection will be not influenced by the complicated geometric shape of this testpiece.

7. Next steps

To improve the detection of even smaller cracks, some investigations will be made with a ultrasonic wave generator, which produces waves in the range of a few MHz. More elaborated measurements with testpieces of any geometric shape in connexion with a new holographic difference-method will be tested to suppress the regular interference pattern and to make visible only the disturbance due to the crack.

Dimension of cracks shown in fig. 3 - 6 : length 10 mm (6 holes
1 mm ϕ in each
row)
depth 1 - 10 mm



Fig. 3: Interference pattern of a 40 mm thick steelplate without any liquid layer

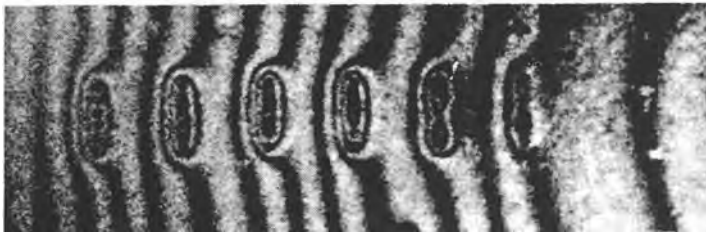


Fig. 4: Interference pattern of the same steelplate as in fig. 3, but with the aid of a thin water-layer. All cracks, even the smallest with a depth of 1 mm are clearly visible



Fig. 5: Interference pattern of a 85 mm thick steelplate. Liquid-layer: waterfilm 2 mm high



Fig. 6: Interference pattern of a 85 mm thick steelplate
Liquid-layer: gelatine 2 mm high

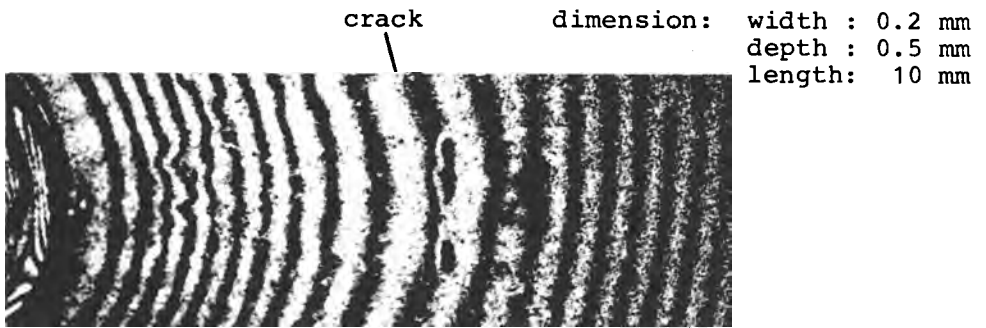


Fig. 7: Interference pattern of a 10 mm thick steelplate
Liquid-layer: gelatine

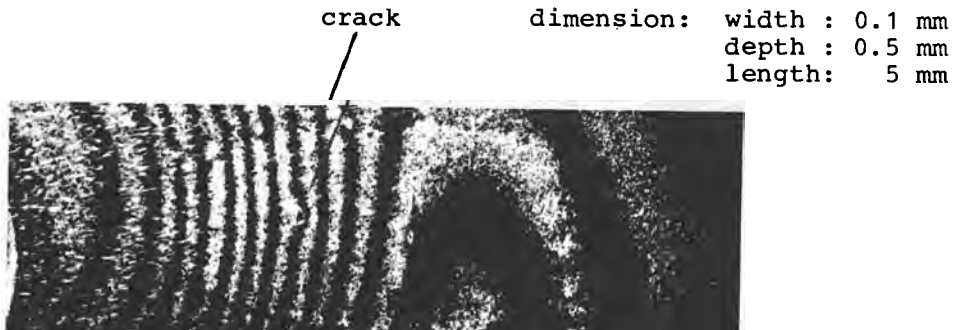


Fig. 8: Interference pattern of a 10 mm thick steelplate
Liquid-layer: gelatine

void

dimension:

depth: 6 mm
Diameter: 6 mm



Fig. 9: Interference pattern of a 40 mm thick steelplate with a void below the surface

welding seam

crack

dimension:

width: 0.3 mm
depth: 2 mm
length: 10 mm

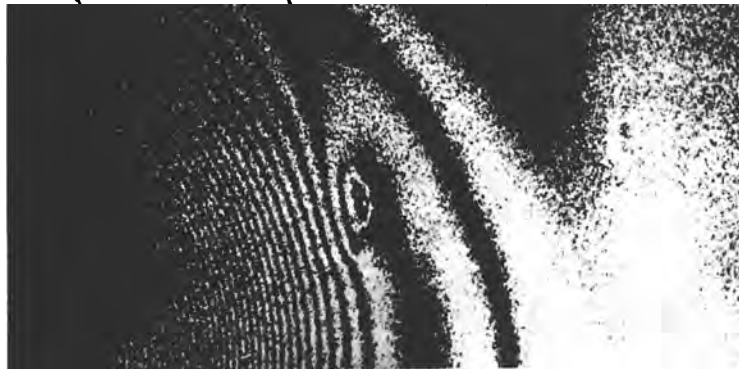


Fig. 10: Interference pattern at an elbow

8. Relations with other Projects

See Annual Reports A74, A75.

9. Reference documents

Annual report A74

Quarterly reports in the series GRS-Forschungsberichte

Report-period Jan. 1976 - March 1976 IRS-F-30

Report-period April 1976 - June 1976 IRS-F-31

Report-period July 1976 - Sept. 1976 IRS-F-33

Report-period Oct. 1976 - Dec. 1976 GRS-F-35

10. Degree of availability

The quarterly reports are available by Gesellschaft für Reaktor-sicherheit (GRS).

<p><u>Title 1 (Original language)</u> Controllo della integrità dei recipienti in pressione mediante monitoraggio dell'emissione acustica.</p>	<p><u>Classification</u> 11.2.3.</p>
<p><u>Title 2 (English)</u> Assessment of nuclear pressure vessel integrity through acoustic emission monitoring.</p>	<p><u>Country</u> ITALY <u>Sponsor</u> ENEL <u>Organisation</u> CISE</p>
<p><u>Date initiated</u> October 1971 <u>Date completed</u> 1978 <u>Last updating</u> March 1977</p>	<p><u>Project Leader</u> G. POSSA - F. TONOLINI</p>

1. General Aim

To develop non-destructive techniques based on acoustic emission monitoring for the assessment of nuclear pressure vessel integrity.

2. Particular Objectives

To develop acoustic emission monitoring systems and methods to be utilized 1) during pressure vessel hydrotests in cold unirradiated conditions, 2) during pressure vessel hydrotests in hot and possibly irradiated conditions, 3) during power operations as a continuous automatic pressure vessel surveillance.

3. Experimental facility and programme

3.1. Experimental facilities: none

3.2. Programme

- 3.2.1. Development of an acoustic emission instrumentation system capable of detection and precise location of emitting defects during pressure vessel cold hydrotests in unirradiated conditions.
- 3.2.2. Development of an acoustic emission instrumentation system for the automatic continuous monitoring of a nuclear pressure vessel during power operation.
- 3.2.3. Installation of a prototype of the instrumentation system developed in 3.2.2. on a power reactor pressure vessel, with the main objective of determining the actual sensitivity in the presence of operational noises.

<u>Title 1 (Original language)</u> Controllo della integrità dei recipienti in pressione mediante monitoraggio dell'emissione acustica.	<u>Classification</u> 11.2.3.
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4. Project status

4.1. Progress to date

- (3.2.1.): 12 channel system for on-line acoustic emission source location designed, manufactured and tested; activity in progress on a 24 channel system.
- (3.2.2.): Prototype of the instrumentation system designed; components purchased or manufactured.
- (3.2.3.): Installation of a two channel instrumentation system prototype on the BWR Caorso pressure vessel to be completed within next May 1977.

4.2. Essential results obtained

Know-how and know-why regarding the technique based on acoustic emission monitoring for defect location in pressure vessels during hydrotests.

5. Next steps

Additional extensive tests of acoustic emission instrumentation in various steel pressure vessel hydrotests (3.2.1.).

Measurements of acoustic emission of the CAORSO BWR pressure vessel during the preoperational tests (3.2.2. & 3.2.3.).

6. Relation to other projects

None

7. Reference documents (Main titles)

- 1) E. Fontana, G. Grugni, B. Pirovano, G. Possa, F. Tonolini: "Controllo non distruttivo di recipienti a pressione mediante analisi dell'emissione acustica nel corso della prova a pressione idrostatica", Energia Nucleare, Vol. 21 n. 10, ottobre 1974.
- 2) E. Fontana, G. Grugni, C. Panzani, B. Pirovano, G. Possa, F. Tonolini: "Acoustic Emission Monitoring During Hydrotests of a Thin Wall Pressure Vessel", Energia Nucleare, Vol. 22 n. 5, maggio 1975.
- 3) E. Fontana et alii: "Acoustic Emission Measurements during the first pressure vessel hydrotest at ENEL - CAORSO BWR", Paper presented to the Conference "Periodic Inspection of Pressure Vessels" - London, Sept. 20/22, 1976.

8. Degree of availability

Various means of cooperation may be envisaged: for example, Italian participation in research programs carried out by partner countries, cost sharing, etc.

9. Budget

The estimate cost of this R&D program is about 200 millions lire per year.

PROJECT TITLE : Study on Evaluation of Structural Safety for Nuclear Pressure Vessels by Acoustic Emission Method	CSNI INDEX NUMBER: 11. 2. 3
SPONSORING COUNTRY: Japan	ORGANISATION: Ship Research Institute, Ministry of Transport
DATE INITIATED: April, 1973 DATA COMPLETED: Progressing (March, 1977)	PROJECT LEADER: A. Kanno

Description :

1. Objectives

The objective of this project is to evaluate the structural safety of nuclear pressure vessels by acoustic emission measuring method. In the practical applications of acoustic emission method, the detailed technical rules are considered to be necessary. This study aims to obtain the basic characteristics of acoustic emission from defected parts of small specimens and pressure vessel models.

2. Project Status

This program consists of two parts, the basic study and the practical study, in the period 1973 to 1978.

In the basic study during the period 1973 to 1975, the acoustic emission source energy of brittle weld crack was estimated relating to the crack area, comparing with the standard acoustic emission source, in which the acoustic emission of dropped steel ball was adopted as a standard acoustic emission. The characteristics of acoustic emission of dropped steel ball were investigated. The propagation of acoustic emission signals in steel plates, the occurrences of acoustic emission in the crack growth during low cycle fatigue bend test of mild steel were also investigated.

In the practical study starting 1975, the model test of acoustic emission was undertaken, making the spherical pressure vessel model and the acoustic emission monitoring system.

3. Essential Result

The characteristics of frequency and amplitude of the standard acoustic emission, concerning to rebound height of dropped steel ball, were experimentally and theoretically analysed. The linear

(To be continued on the next page)

(Continued from the preceding page.)

relation between the acoustic emission energy, which was estimated comparing with the standard acoustic emission, and the crack areas was obtained.

The characteristics of acoustic emission propagation in steel plates, 5mm, 25mm and 80mm thick, were experimentally analysed. Acoustic emission signals, which propagated as the fastest group, attenuated as spherical wave, and that of latter group attenuated in complex modes.

During the fatigue bend test, acoustic emission occurrences above $200\mu\text{v}$ at the preamplifier input were hardly observed from growing crack of the mild steel.

During the hydro-static test of the spherical pressure vessel model, acoustic emission occurrences above $100\mu\text{v}$ at the preamplifier input were hardly observed from the artificial surface notch, about 200mm in length, about 7mm in depth, in the test part (plate thickness; 10mm), and then this vessel was fractured from the surface notch. (Material ; SM 50)

4. Projection

In the practical study, it is scheduled to investigate the acoustic emission signals from the growing cracks in the large model of pressure vessels during cyclic load testing, referring to the acoustic emission characteristics obtained by the material tests.

5. Reports

A.Kannō and M.Sakaki

"Correlation between Acoustic Emission and Crack Size calibrated by Standard Acoustic Emission Source generated by dropped Steel Ball"
The second Acoustic Emission Symposium, Tokyo, September, 1974, (English),
Journal of Non-Destructive Inspection, 23(7) 1974, (Japanese).

A.Kannō and M.Sakaki

"Study on Acoustic Emission Wave Propagation in Steel Plates"
Journal of Non-Destructive Inspection, 24(2) 1975, (Japanese).

A.Kannō, M.Sakaki and M.Shimada

"Acoustic Emission Occurrences during Crack Propagation in Bending Fatigue Test of Mild Steel" Journal of Non-Destructive Inspection, 25(2) 1976, (Japanese).

PROJECT TITLE : Study on flaw size measurement of nuclear reactor pressure vessel using acoustical holography	CSNI INDEX NUMBER: L. W. R. 11. 2. 3
SPONSORING COUNTRY : Japan	ORGANIZATION : Mitsubishi Heavy Industries, Ltd. Takasago Technical Institute
DATE INITIATED : July 1975 DATE COMPLETED : March 1976	PROJECT LEADER : Michio Yamamoto

Description:

1. Objectives

Objectives of this program are to research the way to use acoustical holography for measuring the flaw size which is detected during in-service inspection of nuclear reactor pressure vessel.

2. Project Status

2.1 Program description

In this program three (3) major items which are essential factors when acoustical holography is used to determine the flaw size, are considered.

- (1) Inspection parameters which are the most suitable and give minimum error for reactor vessel inspection.
- (2) Effect of overlay cladding to measured size of flaw when inspection is performed from I. D. surface of reactor vessel.

- (3) Effect of radius of reactor vessel shell to flaw size estimation.

All experimental work had been done using artificial flaws introduced in 150 mm thick steel plate.

2.2 Progress

All experimental and analytical work had been completed.

3. Essential Results

The flaw size may be determined by acoustical holography when appropriate parameters are selected, and the error is approximately $\pm 30\%$ of the size.

4. Projection

Further experimental programs are undergoing to estimate fatigue crack size and to apply acoustical holography to safe-end welds.

5. Report

Final report of this program had been submitted to Science and Technology Agency of Japan.

PROJECT TITLE: Study on flaw size measurement of nuclear reactor pressure vessel using acoustical holography	CSNI INDEX NUMBER: L. W. R. 11. 2. 3
SPONSORING COUNTRY : Japan	ORGANIZATION: Mitsubishi Heavy Industries, Ltd. Takasago Technical Institute
DATE INITIATED : May 1976 DATE COMPLETED : March 1977	PROJECT LEADER: Michio Yamamoto

Description:

1. **Objectives**

Objectives of this program are to research the way to use acoustical holography for measuring fatigue crack size and apply to safe-end welds in nuclear reactor pressure vessels.

2. **Project Status**

2.1 **Program description**

Two (2) major items are included in this program, which are

- (1) To research the measuring error for fatigue crack size estimation in core region of reactor vessel and
- (2) How to apply acoustical holography technique to safe-end welds in reactor vessel nozzles.

2.2 **Progress**

All experimental work had been completed and analytical work is undergoing.

3. **Essential Results**

- (1) Fatigue crack size may be estimated by acoustical holography as well as artificially introduced slit.
- (2) Artificial flaw size introduced in safe-end weld could not be estimated by conventional ultrasonic examination. However, it may be estimated by acoustical holography.

Detailed results are not discussed yet.

4. **Projection**

Further experimental programs are to be planned.

5. **Report**

In course of preparation.

N.V. KEMA

Arnhem, 25 februari 1976

Classification 11.2.3.	
<u>Title 1</u> Automatisch ultrasoon onderzoek van PWR-componenten	<u>Country</u> The Netherlands <u>Sponsor</u> PZEM <u>Organization</u> KEMA, MatEval NDT Co
<u>Title 2</u> Automatic ultrasonic examination of PWR-systems	<u>Projectleader</u> R. Huizing
<u>Initiated</u> 1975 <u>Status</u> Tests are being performed during shut-down February 1976	<u>Scientists</u> H. Jackson (MatEval) V.d. Berg) De Jong) KEMA Boer) Vroman)

1. General aim

Performing inservice ultrasonic inspection on PWR-systems in accordance with the APME-Code and Dutch authorities.

2. Particular objectives

Development of automatic positioning and scanning devices used to carry inspection equipment, such as ultrasonic wheels, to remote areas where manual access is restricted for reliable inspection operations.

The data acquisition equipment is a special purpose mini-computer to provide optimum data; easy command of the data and mechanical systems; and convenient, easily interpreted, real-time display of all essential information.

3. Experimental facilities and programme

Pre-tests have been performed in KEMA- and MatEval (Warrington) laboratories.

The programme consists of:

- ultrasonic inspection of Circumferential welds and nozzle welds in a pressurizer;
- ultrasonic inspection of circumferential welds and nozzle welds of a steamgenerator;
- ultrasonic inspection of circumferential and longitudinal welds in mainloops.

4. Project status

Tests are being performed during shut-down February 1976.

5. Next steps

The main devices to carry the inspection equipment will not be removed and will be used for the next inspection period.

6. Relations with other projects

Not applicable.

7. Reference documents

Not applicable.

8. Degree of availability

Through the organization KEMA.

PROJECT TITLE : DEVELOPMENT OF ACOUSTIC EMISSION MEASUREMENT	CSNI INDEX NUMBER : 11.2.3
SPONSORING COUNTRY : UNITED KINGDOM	ORGANISATION : REML RISLEY
DATE INITIATED : 1969 DATE COMPLETED :	PROJECT LEADER : P BENTLEY

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.

OECD NUCLEAR ENERGY AGENCY - 38, Boulevard Suchet, 75016 PARIS
Telex : 630668 AEN NEA - Tel : 524 82-00

31.719

PROJECT TITLE : DEVELOPMENT OF HOLOGRAPHY AND LASER SPECKLE PHOTOGRAPHY	CSNI INDEX NUMBER : 11.2.3
SPONSORING COUNTRY : UNITED KINGDOM	ORGANISATION : SRD CULCHETH
DATE INITIATED : 1973 DATE COMPLETED :	PROJECT LEADER : D J V MARTIN

Description :

1. General Aim

To determine from measurements made during a non-destructive pressure test whether a vessel has a significant crack.

To evaluate strain by laser speckle photography.

2. Particular Objectives

To characterise holograms of typical vessel regions - membrane, junction, nozzle, etc - with and without cracks.

To holographically inspect regions of a reactor pressure vessel.

3. Experimental Facilities

The photographic equipment - including a pulsed laser - is being assembled. Tests are in hand on: a 5' vessel at Culcheth; a 3" test plate in a 4800 T machine at Risley; a 2" test plate in a machine at Windscale; any thickness of test plate (currently 3") in the hydraulic pressuriser at Springfields. Part through defects in a series of small cylinders are being hologrammed. Proposals for the inspection of a reactor pressure vessel have been issued.

Reference Reports

AEA internal reports

Martin D J V Holographic method giving stress levels and visualisation of defects in thick cylinders. To be published.

Martin D J V Holographic interferometry and laser speckle photography as aids to assessment of pressurised components. Paper No. G3.10 at 3rd Int Conf on Structural Mechanisms in Reactor Technology September 1975

Martin D J V Laser Holographic and Speckle Photography Methods for Defect Detection and Strain Evaluation in Pressure Vessels. To be published.

OECD NUCLEAR ENERGY AGENCY - 38, Boulevard Suchet, 75016 PARIS
Telex : 630668 AEN NEA - Tel : 524 82-00

PROJECT TITLE: Acoustic Emmission - Flaw Relationship for Inservice Monitoring of Nuclear Pressure Vessels	INDEX NUMBER: 11.2.3
SPONSORING COUNTRY: USA	ORGANIZATION: Battelle-Pacific Northwest Laboratory
DATE INITIATED: July 1976 DATE COMPLETED: Continuing STATUS AS OF:	PROJECT LEADER: P. H. Hutton E. B. Schwenk

Description:

1. General Aim:

- a. Characterize AE from a limited range of defects and material property conditions recognized as being of primary concern to reactor pressure vessel integrity.
- b. Characterize AE from innocuous sources (including defects) which can be reasonably expected to exist in a pressure vessel.
- c. Develop criteria for distinguishing significant defects from innocuous AE sources.
- d. Develop an AE-material property-flaw damage model to establish a basis for relating in-service AE to pressure vessel integrity.

2. Particular Objectives:

- a. Procure test material and fabricate specimens.
- b. Complete test instrumentation and establish calibration of complete mechanical-electronic test system.
- c. Start development of AE signatures for A533B steel for:
 - (1) Deformation and cracking under uniaxial load in base metal and in weld metal.
 - (2) Fatigue crack growth in base metal and in weld metal.
 - (3) Oxide deformation and cracking.
 - (4) Innocuous slag inclusion under cyclic load.
- d. Investigate characteristics of kaiser effect in crack growth through plastically deformed material.

- e. Analyze all of the above results for distinction between different AE sources and start development of a fracture mechanics-AE model as a basis for evaluating defect significance by AE.

In the above work, 2.c., d., and e., consider a restricted number of variables in material properties, defect types, innocuous noise sources, growth mechanisms, and environment. Environmental effects will be investigated where considered significant on an individual basis.

- f. Determine the suitability of commercial high temperature sensors with mounting and coupling method for reactor service. Report results to NRC with definition of any further development needed. Initiate further development if needed upon NRC concurrence.
- g. Prepare a topical report containing all the results, analyses and procedures from items 2.c. thru 2.f., recommendations on preliminary feasibility of AE in-service monitoring of reactors and a review and assessment of planned laboratory work to achieve the program objectives.
- h. Coordinate collection and analysis of AE data from HSST specimens, combine these results with the analysis and development of Item 2.e., above. This will involve direct measurements on irradiated fracture specimens plus initiate preparation of a standard method to be used for AE monitoring of HSST testing.
- i. Maintain a library of AE research and application being performed by foreign nations, review and analyze the information and report significant findings to NRC.

3. Experimental Facilities and Programs:

- a. Facilities testing laboratory at PNL; accumulation of AE data from tests at HEDL, and HSST programs.
- b. Program: See item 2 above.

4. Project Status: This is a new project.

5. Next Steps: See item 2 above.

6. Relation with Other Projects: This program has direct relationship to other current programs for development of reactor pressure vessel AE monitoring methods.

7. Reference Documents: Quarterly progress reports and topical reports as they become available.
8. Degree of Availability: Documents will be available from NTIS.

PROJECT TITLE: Improved Non-Destructive Testing of Pressure Vessels	INDEX NUMBER: 11.2.3
SPONSORING COUNTRY: USA	ORGANIZATION: University of Michigan
DATE INITIATED: 8/74 DATE COMPLETED: Continuing STATUS AS OF:	PROJECT LEADER: J. R. Frederick

Description:

1. General Aim:

To assess the capability of the three-dimensional ultrasonic testing system developed during FY 1976 to accurately characterize typical discontinuities with respect to size, shape and orientation within the allowable limits set by the ASME boiler and pressure vessel code Section XI. To develop interactive display techniques and assess change detection testing. To ensure early transfer of this technology to field service by providing consultation and evaluation services for development of a field prototype version of the University of Michigan high resolution UT system and implementation and evaluation of techniques to provide a higher rate of data acquisition.

2. Particular Objectives:

- a. Complete the evaluation of typical discontinuities in reactor pressure vessel base materials, welds, cladding, and austenitic stainless steel piping using the three-dimensional system developed in FY 1976. Emphasis will be placed on naturally occurring and artificially induced discontinuities.
- b. Obtain conventional ultrasonic test results of the same test samples as above in accordance with ASME code Section XI procedures.
- c. Obtain destructive examination results of the same test samples as above for the true discontinuity sizes, shapes and orientations.
- d. Prepare a topical report comparing the results of the above three tasks and assessing the capabilities and limitations of the prototype three-dimensional system as compared to conventional ultrasonic inspection techniques. Included will be a description of modifications necessary for field implementation and the effects of such modifications on system performance.

- e. Develop software for real time display of three-dimensional system data including conventional A-scan, B-scan and C-scan formats. Begin to incorporate the capability for interactive display of discontinuities from different perspectives. Develop initial software for interactive image enhancement.
- f. Perform three-dimensional imaging of fatigue crack samples at successive stages of propagation. Determine the sensitivity of the technique to repositioning errors, the use of substitute transducers and changes in surface finish.
- g. Develop software for the interactive display of change detection data. Determine to what extent interactive display techniques may be used to compensate for positioning errors.
- h. Prepare a topical report on the capabilities of the prototype system to perform change detection testing and assess its applicability to field implementation.
- i. Continue cooperative effort with EG&G initiated during late FY 1976 (with Aerojet Nuclear Company) to employ the techniques of high resolution imaging on the LOFT reactor facility. This will include consultation on system implementation and processing of initial inspection data at the University of Michigan facility during late FY 1977.
- j. Provide information on the developed techniques and methods for field implementation to inspection agencies, utilities and manufacturers to encourage early transfer to field application.
- k. Provide technical assistance and specifications during the design phase of a field prototype system.
- l. Provide research support at the University of Michigan for evaluation of critical parameters peculiar to the method of field implementation. This will most likely include the method of transducer coupling to the extent that it is feasible with existing facilities.
- m. Assist in evaluating system performance and provide consultation on techniques for improving operational performance. This task may include transfer of test data on magnetic tape media for evaluation at the University of Michigan.
- n. Acquire, implement and evaluate a transient digital recording device with the potential of increasing the rate of data collection by nearly three orders of magnitude.

- o. Investigate other methods for increasing the rate of data acquisition such as multi-channel operation and non-coherent pulse repetition.
 - p. Prepare a topical report comparing the results of tasks 2.n. and 2.o.
 - q. Upgrade existing computer system for spotlight scan mode, development of contour-perspective plots and parallel processing.
 - r. Continue to participate in the NDE activities of ASME code Sections III, V and XI for upgrading of present code rules. Provide trip reports, oral reports and consultation on same to the Nuclear Regulatory Commission.
 - s. Perform travel to foreign countries as necessary, to assist in obtaining and evaluating NDE research and development data.
3. Experimental Facilities and Program:
- Facilities: Computer-automated, synthetic transducer array inspection facility.
- Program: See item 2 above.
4. Project Status:
- a. Results to date: Demonstrated feasibility of phase-sensitive synthetic array to improve location and resolution of flaws.
 - b. Essential results: Improved lateral and longitudinal resolution.
5. Next Steps: Complete evaluation of new UT system; build a prototype system to be used in production inspections.
6. Relation to Other Projects: To assist improvement in non-destructive inspection of stainless steel welds and PV steels.
7. Reference Documents: Annual Report.
8. Availability: NTIS.

PROJECT TITLE: Inspection of Nuclear Reactor Welding by Acoustic Emission	INDEX NUMBER: 11.2.3
SPONSORING COUNTRY: USA	ORGANIZATION: General American Research Division
DATE INITIATED: 10/74 DATE COMPLETED: Continuing STATUS AS OF:	PROJECT LEADER: D. Prine

Description:

1. General Aim:

To develop the techniques for continuous monitoring of welding of nuclear components, such as pressure vessels and piping by acoustic emission; to prove the feasibility of the technique under shop conditions and validate the acoustic emission findings by currently accepted NDE techniques; to provide data for ASME code acceptance of the inspection practice; to develop, build and validate prototype monitors for "no-hands" service in shops for nuclear pipe and pressure vessel welding.

2. Particular Objectives:

- a. Perform additional laboratory testing to aid in determination of AE flaw detectability and flaw type discrimination.
- b. Develop correlation between acoustic emission and flaw size by analyzing AE data and cross validating of flaws with other NDE techniques as well as quantitative metallography.
- c. Construct a prototype vessel weld monitor and prove under shop conditions.
- d. Complete shop evaluation of both the piping and pressure vessel AE monitors.
- e. Analyze all the data and experience generated during this program to present a case for NRC acceptance of acoustic emission in-process weld monitoring of nuclear components.

3. Experimental Facilities and Program:

- a. Facilities: Testing set-up at several pipe fabrication shops.
- b. Program: See item 2 above.

4. Project Status:
 - a. Results to date: Demonstrated flaw detection by acoustic emission during welding a variety of piping and in pressure vessel steel in commercial welding shops.
 - b. Essential results: Establishment of positive correlation between acoustically-detected flaws metallography and other NDE results.
5. Next Steps: Test a pressure vessel weld monitor in long production runs. Quantify detection of flaws during pipe and pressure vessel welding.
6. Relation With Other Projects: Studies being conducted in commercial piping and pressure vessel welding shops, not in research labs.
7. Reference Documents: Quarterly progress reports, "Inspection of Nuclear Reactor Welding by Acoustic Emission."
8. Degree of Availability: NTIS.

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1977

VOLUME II
WATER REACTORS
(continued)

11.2.4. Destructive testing

Classification: 11.2.4.

Title: Trykprøvningsfacilitet	Country: DENMARK
Title: Pressure Testing Facility	Sponsor: Res. Establ. Risø
Initiated date: 1977 Status: In progress	Organization: Res. Establ. Risø Scientists: A. Nielsen C. Debel

1. General aim: To provide a facility for testing to failure of intermediate size vessels under controlled conditions.
2. Particular objectives: Testing of experimental vessels particularly under gas pressure in order to facilitate the testing at different temperatures.
3. Experimental facilities and programme: The aim is to provide a facility for fracture mechanics and acoustic emission experiments on vessels.
4. Project status: Designing is in progress.
5. Next steps: Construction.
6. Relation to other projects: The facility should be available for the projects Dynamic Fracture Mechanics and Acoustic Emission.
7. Reference documents: None so far.
8. Degree of availability: In principle no limitations.

N.V. KEMA

Arnhem, 25 februari 1976

Classification 11.2.4	
<u>Title 1</u> Onderzoek naar het scheuruitbreidingsgedrag van onderplaterings-scheurtjes in PWR's	<u>Country</u> The Netherlands <u>Sponsors</u> PZEM, GKN, TNO, KEMA <u>Organizations</u> KEMA, TNO
<u>Title 2</u> Evaluation of the behaviour of undercladding cracks in a PWR	<u>Projectleader</u> R.M. van Kuijk
<u>Initiated</u> 1971 <u>Status</u> Completed 1975	<u>Scientists</u> A.P.A.M. Beyers L.B. Dufour H.C. van Elst

1. General aim

Crack extension behaviour of undercladding cracks in heavy section nuclear steel pressure vessel during service.

2. Particular objectives

Investigation of the crack extension in A508-02 under cyclic load conditions, load and temperature simultaneously. Measurements of crack depths were performed by sampling and statistical approaches.

3. Experimental facilities and programme

The tests have been performed in TNO and KEMA laboratories.

The programme consisted of:

- making a testpiece of A508-2 containing undercladding cracks;
- development of apparatus for realizing simultaneously cycling load and temperature;

- testing two testpieces with undercladding cracks under cyclic loading (1000 times) and constant temperature;
- testing two testpieces with undercladding cracks under cyclic loading (1000 times) and cyclic temperature from 70° - 300°C simultaneously;
- sampling the testpieces after the tests, measurements of crack depths, statistical approaches.

4. Project status

Completed end 1975. It was shown that undercladding cracks grow very little in the base material and no indications were found that the cracks will grow through the cladding.

5. Next steps

None.

6. Relations with other projects

Through TNO with similar projects (TNO, Breda)

7. Reference documents

A serie of about 30 technical and progress reports have been prepared. All these reports are written in Dutch.

8. Degree of availability

Through the organization KEMA.

PROJECT TITLE : STEEL PRESSURE VESSEL INTEGRITY	CSNI INDEX NUMBER : Water reactors 11.2.4
SPONSORING COUNTRY : Switzerland	ORGANISATION : Nordostschweizerische Kraftwerke AG, Baden + Sulzer Bros., Winterthur
DATE INITIATED : 1975 DATE COMPLETED : 1 st quarter 1977	PROJECT LEADER : Nordostschweizerische Kraftwerke AG, Baden

Description :

Fracture mechanics testing programme to achieve a more realistic estimate of critical defect size in a reactor pressure vessel.

The testing programme uses specially instrumented CT-specimens (Compact Tension) of nearly true wall thickness and at temperatures representing operational and materials aspects (i.e. NDTT shift due to radiation embrittlement).

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Telex : 630668 AEN NEA - Tel : 524 82-00

31.719

PROJECT TITLE: Assessment of Reactor Piping Research	INDEX NUMBER: 11.2.4
SPONSORING COUNTRY: USA	ORGANIZATION: Battelle Memorial Inst. Columbus, Ohio
DATE INITIATED: 1976 DATE COMPLETED: Continuing STATUS AS OF:	PROJECT LEADER: E. C. Rodabaugh

Description:

1. General Aim:

To evaluate all significant research on piping systems, to correlate that research with current design practices, and to quantify the piping used in nuclear power plants in terms of reliability.

2. Particular Objectives:

- a. Complete the current effort to complete a document on the assessment of reactor piping research. The chapters are to be entitled: Introduction, Structural Design of Piping Systems, Service Experience, Research on Piping Products Without Controlled Flaws, Research on Piping Products With Controlled Flaws, Reliability of Piping Products and Systems, and Summary. This task is expected to be complete on June 1977.
- b. Initiate a concurrent task on the quantification of the reliability of safety related piping systems. The result will be primary recommendations to improve piping system reliability by providing answers to questions such as (a) where is a failure most likely to occur? (b) Will the failure be a leak or a break? (c) What deterministic steps can be taken to improve the reliability? and (d) Where should future efforts be directed to improve the reliability?

3. Experimental Facilities and Program:

- a. Facilities: Analytical work performed at Battelle-Columbus.
- b. Program: See Item 2 above.

4. Project Status:

- a. Results to date: None; report in draft stages.
- b. Essential results: none yet.

5. Next Steps: See Item 2 above. Report to be completed summer 1977.
6. Relation to other projects: This is part of a coordinated NRC effort to assure the safety of nuclear power reactors.
7. Reference documents: Final report to be published summer 1977.
8. Degree of availability: Available from NTIS.

PROJECT TITLE: Design Criteria for Piping and Nozzles	INDEX NUMBER: 11.2.4
SPONSORING COUNTRY: USA	ORGANIZATION: Oak Ridge National Lab.
DATE INITIATED: 7/75 DATE COMPLETED: Continuing STATUS AS OF:	PROJECT LEADER: S. E. Moore

Description:

1. General Aim:

To conduct integrated experimental and analytical stress analysis studies of piping system components and isolated and closely-spaced pressure vessel nozzles to confirm and/or improve the adequacy of structural design criteria and analytical methods used to assure the safe design and operation of nuclear power plants.

2. Particular Objectives:

a. Publish Reports on the Subjects Listed Below:

- (1) Finite element analysis parameter study of a reinforced nozzle attached to a cylindrical shell under internal pressure.
- (2) Experimental stress analysis of mechanized 10-inch diameter pipe elbows with specified geometric distortions.
- (3) Test report on the cyclic internal pressure fatigue test of an out-of-round machined 10-inch diameter elbow - ORNL model no. ME-2.
- (4) Interpretive report on the experimental elastic-response and fatigue-to-failure tests of five 12-inch nominal size ANSI standard B16.9 TEES.

b. Complete Effort and Publish Reports on the Subjects Listed Below:

- (1) Finite element analysis of two closely-spaced nozzles in a cylindrical pressure vessel under internal pressure.
- (2) Finite element analysis of two closely-spaced nozzles in a cylindrical pressure vessel with in-phase force and moment loads.

- (3) Dimensional study of ANSI standard pipe fittings.
 - (4) Flexibility factor for branch connection.
 - (5) Experimental determination of plastic collapse loads for piping elbows.
 - (6) Finite element analysis parameter study of reinforced nozzle attached to cylindrical shell under force moment loadings.
- c. Isolated and Closely-Spaced Interacting Nozzles in Cylindrical Pressure Vessels.

Conduct Finite element analyses of geometries having two and three interacting nozzles. Issue reports as noted in Tasks 2.a., and 2.b., and document the computer programs. Continue code rules development for single and close-spaced nozzles.
 - d. Elbow End Effects.

Continue the development of stress indices, flexibility factors, and proposed code rules for piping elbow end effects.
 - e. Code Efforts.

Continue close association with PVRC and codes and standards groups, including management of PVRC subcommittee activities, to aid in and to facilitate the implementations of proposed rules and revisions.
3. Experimental Facilities and Program:
 - a. Facilities: Testing laboratories and facilities at Oak Ridge, Tennessee and subcontractor locations.
 - b. Program: See Item 2 above.
 4. Project Status:
 - a. Results to date: See progress reports for details.
 - b. Essential results: To develop analysis methods for piping systems and nozzles and to develop design rules and data for codes and standards.
 5. Next Steps: Essentially as noted in Item 2 above.

6. Relation To Other Projects: This is part of a coordinated effort to assure the safety of nuclear power reactors.
7. Reference Documents: Quaterly progress reports and topical reports, when issued.
8. Degree of Availability: All available through NTIS.

PROJECT TITLE: Development of High Sensitivity Ultrasonic Techniques for In-Service Inspection of Nuclear Reactors	INDEX NUMBER: 11.2.4
SPONSORING COUNTRY: USA	ORGANIZATION: National Bureau of Standards Washington, D.C.
DATE INITIATED: 1975 DATE COMPLETED: Continuing STATUS AS OF:	PROJECT LEADER: M. Linzer

Description:

1. General Aim:

To produce effective in-service inspection devices for early detection of flaws in nuclear reactor components; to improve the sensitivity and flaw discrimination of ultrasonic pulse echoes; to conduct theoretical analyses of the interaction of ultrasonic radiation with materials as a basis for the development of novel field instruments.

2. Particular Objectives:

- a. Concentrate efforts on further development of techniques to improve the signal-to-random noise ratio of ultrasonic signals and apply these to practical problems encountered in nuclear reactor monitoring. Complete the high-speed signal averaging device and increase the pulse compression ratio of the ultrasonic chirp radar system. The combined signal averaging and pulse compression system will be applied to detection of flaws in high-attenuating materials used in nuclear reactor components. Samples to be studied will include PV steels, austenitic stainless steel piping material, heavy sections of attenuating material, and weldments. Both natural and machined flaws will be investigated. Defect location and characterization will be quantified.
- b. Construct a prototype high power insonification system. The maximum power employed is to be dictated by the power handling capacity of the transducer elements. Another area to be explored is power concentration in the material by the use of an electronically-focused concentric annular array. This approach would also have the additional advantage of improving lateral resolution. Finally, sensitivity enhancement by use of acoustic matching elements between the transducer and material will be investigated. The prototype device will incorporate all those components of ultrahigh sensitivity systems which the investigation shows to be useful.

- c. Prepare a topical report to include (a) systems and components description and (b) quantitative capabilities for defect location, characterization and detection by the developed high-sensitivity techniques and systems.
- 3. Experimental Facilities and Program:
 - a. Facilities: Laboratory and test facilities at NBS, Washington, D.C.
 - b. Program: See Item 2 above.
 - 4. Project Status:
 - a. Results to date: Signal averaging device capable of 50 mega hertz ready for debugging. Pulse compression device capable of 8:1 completed, now working on device capable of 20:1.
 - b. Essential results: See progress reports.
 - 5. Next Steps: See progress reports.
 - 6. Relation to other projects: Upgraded UT devices from this program are being incorporated into other NDE projects being sponsored by NRC.
 - 7. Reference Documents: Annual Report when issued.
 - 8. Degree of Availability: Available from NTIS.

PROJECT TITLE: Electrochemical Fracture in Nuclear Power Reactors	INDEX NUMBER: 11.2.4
SPONSORING COUNTRY: USA	ORGANIZATION: American University Washington, D.C.
DATE INITIATED: 1975	PROJECT LEADER:
DATE COMPLETED: Continuing	Dr. B. F. Brown
STATUS AS OF:	

Description:

1. General Aim:

To perform research on inhibitor systems for BWR coolant environments to reduce or preclude stress corrosion cracking; to monitor research programs in stress corrosion cracking and recommend changes in emphasis or direction; recommend corrective actions to be taken to preclude stress corrosion cracking in nuclear systems.

2. Particular Objectives:

- a. Develop and prove the concepts by which inhibitors can be concentrated at the same locations and by the same forces which cause the concentration of noxious anions.
- b. Conduct studies to determine the mechanism by which cadmium ions provide inhibition (as established in FY 76) to stress corrosion cracking of sensitized 304 stainless steel.
- c. Continue the search for other metal ion inhibitors based on descriptive chemistry and on the mechanism of cadmium ion inhibition as it becomes established.
- d. Continue evaluation of other quaternary organic inhibitors.
- e. Evaluate inhibition by dianodic formulations of quaternaries and metal ions.
- f. Attend meetings related to corrosion technology of interest to NRC and report on results of same.
- g. Write a topical report on the results and use of inhibitors to combat intergranular stress corrosion cracking of sensitized 304 stainless steel.

3. Experimental Facilities and Program:
 - a. Facilities: Experimental laboratory at American University.
 - b. Program: See Item 2 above.
4. Project Status:
 - a. Results to date: Have identified several metal ion complexes which show promise for corrosion inhibition. See progress for details.
 - b. Essential results: See progress reports.
5. Next Steps: See Item 2 above.
6. Relation to other projects:
7. Reference documents: Annual reports and monthly progress letters.
8. Degree of Availability: All available through NTIS.

PROJECT TITLE: LWR Pressure Vessel Irradiation Surveillance Dosimetry	INDEX NUMBER: 11.2.4
SPONSORING COUNTRY: USA	ORGANIZATION: Hanford Engineering Development Lab., Richland, Wash.
DATE INITIATED: 1977	PROJECT LEADER:
DATE COMPLETED: Continuing	W. N. McElroy
STATUS AS OF:	

Description:

1. General Aim:

Make measurements in standard reference and controlled LWR-pressure vessel benchmark fields using available multiple foil techniques, quartz sensors, and SSTR's. Preparation of updated and improved ASTM recommended practices for the routine application of validated LWR-dosimetry and surveillance procedures.

2. Particular Objectives:

- a. Define completely the scope of the measurement program for each fiscal year, 1977 through 1981, covering proposed irradiation, calibration, and validation of passive HEDL sensors in PWR (BR-3), NBS, and CEN/SCK benchmark fields. Include the scope in an "Analysis Before Test" document for NRC that also provides the technical approach, interfaces, experimental test plan, analyses, and outline of results for use by NRC. Submit the "Analysis Before Test" document to NRC by March 30, 1977, for permission to continue the program and complete the FY 77 work scope. Work beyond FY 77, associated with the program, will be subject to normal NRC/ERDA concurrence procedures for placing NRC work at ERDA laboratories.
- b. Provide sets of sensors to be incorporated in EPRI-sponsored irradiations in a PWR (McGuire) and BWR (Browns Ferry-3).
- c. Recommend and fabricate state-of-the-art passive radiometric and stable product foils, and recommend surveillance capsule multiple foil sets for approved irradiations in benchmark fields.
- d. Define required updated and improved LWR dosimetry and surveillance ASTM recommended practices; initiate writing of recommended practices.

3. Experimental Facilities and Program
 - a. Experimental Facilities: Power reactors, Browns Ferry-3, McGuire-1, Arkansas Power and Light, Tihange and BR-3 (Belgium), Garigliano (Italy); Test Reactors, ORR/BSR, NBSR, BR-1 (Belgium); Sigma Sigma, Belgium; ISNF.
 - b. Program: See Item 2 above.
4. Project Status:
 - a. Results to date: New project
 - b. Essential results: none reported yet.
5. Next Steps: See Item 2 above.
6. Relation to Other Projects: This program is being coordinated by NRC with EPRI, NBS, the Belgian, British, and Italian governments.
7. Reference Documents: Quarterly progress and topical reports when issued.
8. Degree of Availability: Available from NTIS when published.

PROJECT TITLE: Primary Coolant Pipe Rupture Study	INDEX NUMBER: 11.2.4
SPONSORING COUNTRY: USA	ORGANIZATION: General Electric Company
DATE INITIATED: 1967 DATE COMPLETED: Continuing STATUS AS OF:	PROJECT LEADER: J. Danko D. A. Hale

Description:

1. General Aim:

To improve design of nuclear piping and ensure integrity of the entire piping system; to show that large pipe failures are not credible for these systems; to develop the rate and characteristics of crack growth under BWR environmental conditions for piping materials; to develop an electrochemical field test for detection of sensitized austenitic stainless steel.

2. Particular Objectives:

a. Task K_d

- (1) Analyze and compare data from the 500°F baseline air test to the companion Dresden I BWR crack initiation data and issue a topical report on fatigue crack initiation in a BWR environment.
- (2) The environmental fatigue crack growth tests begun in FY 76 will be completed. Several additional tests at other frequencies and mean stress levels will be conducted as shown in Attachment III, "Proposed Task K_d Test Matrix," of the revised analysis-before-test submitted 8/23/76.
- (3) Electrochemical potential measurements will be made concurrent with the tests of Item 2.a.(2).
- (4) Selected specimens will be metallographically examined to attempt to correlate fracture surface characteristics with macroscopic crack growth behavior.
- (5) A topical report summarizing the results of Items 2.a.(2), (3), and (4) in terms of the effects of frequency and mean stress on crack growth behavior of primary piping materials in a BWR primary water environment will be issued.

b. TASK G

- (1) Write a topical report incorporating all the results through FY 76; make judgment relative to success of correlating DOS with IGSCC and decision for development of a field test for weld qualification.
- (2) The constant load and constant extension rate stress corrosion cracking tests on the ten heats of welded Type 304 and 304L stainless steel pipe will be completed. The electrochemical method for measuring the degree of sensitization of welded stainless steel piping and for stainless steel with simulated weld thermal cycles will be optimized by conducting a number of parametric studies to obtain the greatest sensitivity possible. The Parametric studies will include variations in KSCN additions to the electrolyte (Grain Boundary Activator), time-at-passivating potential, temperature of electrolyte, reactivation sweep rate, sample surface finish, passivating potential, and normalization of the electrochemical data to sample grain boundary area.
- (3) The DOS will be measured in ten heats of welded piping material (seven type 304 and three type 304L) and then correlated with IGSCC results obtained by both constant load and constant extension rate (CXR, rapid screening test) tests in the accelerated BWR coolant environment. The DOS measurements will be obtained at OD and ID regions of the weld haz's at two azimuths of each pipe. Also each weld haz will be profiled for maximum sensitization characterization using a sequential polishing and measurement Technique. Finally, all welded pipe samples will be tested for IGSCC using the CXR method, and these data compared to electrochemical measurements. Data will be analyzed to determine if a correlation exists relating DOS measured on the OD of a pipe weld to predictable IGSCC behavior on the ID.
- (4) Design and construct an electrochemical cell for measuring the degree of sensitization of welded 304 stainless steel pipe for field application. Consider methods for and attempt obtaining the electrochemical measurements directly on the ID surfaces of piping. Develop procedures and techniques in the laboratory on welded pipe sections. The procedures and techniques will be adaptable for use in the field. Capability of the cell for detection of IGSCC susceptibility will be evaluated.
- (5) A topical report will be prepared and summarize all findings and data relative to the DOS and IGSCC resistance of multiple tests.

Of type 304 welds, including the effects of low carbon levels. Included in the report will be (1) procedures for conducting IGSCC tests in an accelerated BWR coolant environment using both constant load and constant extension rate methods, (2) a set of specifications for the fabrication of any electrochemical cell field unit, (3) procedures for conducting the field test, (4) recommendations relative to assessment of DOS measurements in terms of acceptable IGSCC resistance substantiated with data developed during the program.

3. Experimental Facilities and Program:
 - a. Facilities: Two vessels at San Jose for fatigue testing in a simulated BWR water environment. Electrochemical test facilities at Vallecitos Nuclear Center.
 - b. Program: See Item 2 above.
4. Project Status:
 - a. Results to date: See progress reports for details. Shown conservatism of ASME Code design rules for fatigue crack initiation in large component prototypes, and from specimen tests.*
 - b. Essential results: Have helped prove conservative design and safety of piping systems in BWR systems.
5. Next Steps: Complete experimental work in 3B and report.**
6. Relation to Other Projects: Provides fundamental data to support piping reliability analyses, in connection with other NRC programs on vessel integrity, and LOCA for overall evaluation of reactor safety.
7. Reference Documents: "Reactor Primary Coolant System Pipe Rupture Study" Progress Reports, beginning GEAP-4911 April-June 1965, plus topical reports.
8. Availability: Recent issues from NTIS.

*Developed laboratory tests to determine degree of sensitization of stainless steel to stress corrosion cracking.

**External laboratory sensitization test for field applications.

PROJECT TITLE: Residual Stresses at Girth-Butt Welds in Pressure Vessels and Pipes	INDEX NUMBER: 11.2.4
SPONSORING COUNTRY: USA	ORGANIZATION: Battelle Memorial Inst. Columbus, Ohio
DATE INITIATED: 1976 DATE COMPLETED: Continuing STATUS AS OF:	PROJECT LEADER: E. F. Rybicki

Description:

1. General Aim:

To develop an analytical method or methods for calculating the magnitude, direction, and distribution of residual stresses at Girth-Butt welds in pressure vessels and pipes.

2. Particular Objectives:

a. Review of literature and current programs.

Complete the literature survey and review current programs of a related nature. Report these review results and include in the final report.

b. Residual stress data for developing and verifying the mathematical model.

Perform Girth-Butt weld experiments, taking temperature data and evaluating residual stress magnitude, direction and distribution. At least three such tests must be performed, one with two weld passes, one with approximately six passes and one with many passes. From task 2.a., obtain as much applicable data as possible on other programs, such as those at ANL and GE.

c. Mathematical Models.

Utilize the data and information from tasks 2.a. and 2.b. to develop, and then simplify, the methods used to obtain temperature profiles and the resultant residual stress fields.

Prepare a final report, including a documented computer program. Show typical example of usage, as well as comparisons of the analytical results with the experimental results.

d. Residual stresses in weld repair regions.

Evaluate the applicability of the currently available methods to analyze the residual stress fields in repair welds in heavy section steel. This should be coordinated with the HSST program, and the results compared with the experimental results from prolongation weld repairs. To be most useful, this task must be performed expeditiously for two ITV's will be tested near the end of FY 77, and in one, the results are critical.

3. Experimental Facilities and Program:

a. Facilities: Experimental test facilities at BMI, Columbus.

b. Program: See Item 2 above.

4. Project Status:

a. Results to date: Analytical model developed, and initial check on HSST repair weld showed excellent agreement with measurements of weld residual stresses.

b. Essential results: See progress reports.

5. Next Steps: See Item 2 above.

6. Relation to other projects: This is part of a coordinated effort by NRC to assure the safety of nuclear power reactors.

7. Reference documents: Quarterly progress and topical reports.

8. Degree of Availability: All available from NTIS.

PROJECT TITLE: Steam Generator Tube Integrity Program	INDEX NUMBER: 11.2.4
SPONSORING COUNTRY: USA	ORGANIZATION: Battelle Memorial Inst. Pacific Northwest Lab.
DATE INITIATED: 1976 DATE COMPLETED: Continuing STATUS AS OF:	PROJECT LEADER: M. Vagins

Description:

1. General Aim:

- a. Carry out a selected number of burst and collapse tests on baseline and artificially defected tubes representative of the tubes presently installed in PWR steam generators, such tests conducted in environments similar to those found in in-service.
- b. Carefully control the creation of the defects to be similar to those found in PWR steam generators and fully characterize them by use of replication and EDDY current and/or other nondestructive testing methodology.
- c. Develop methodology (technique) for correlating the results of field inspection defect determination with the results of laboratory burst and collapse testing of flawed tubes.
- d. Develop burst and collapse pressure margins of safety as a function of flaw geometry and tube type and operating conditions.
- e. Develop criteria to be applied to the evaluation of remaining life of tubes in which flaws are found during in-service inspections of PWR steam generators.

2. Particular Objectives:

- a. Procure tube material and carry out baseline geometry, material, and strength characterization tests.
- b. Complete bulk of literature survey, summarize, and report findings.
- c. Complete test equipment and instrumentation and establish calibration of all mechanical, hydraulic, and electronic components.

- d. Have vendor complete all defect machining by electrodischarge machining, chemical etching, and/or mechanical cutting (at least for the initial gross material removal for wastage simulation). Establish vendor shipping dates so as to allow free and rapid program continuation.
 - e. Carry out full matrix of tube burst tests and initiate tube collapse tests, both series of tests conducted in environments simulating actual field service.
 - f. Establish criteria for bulge tests and develop test methodology, test equipment, and instrumentation to carry out tests; then carry out tests.
 - g. Develop test procedure for fatigue testing a select number of dented tube specimens under simulated thermal and pressure start-up/shutdown conditions; then initiate tests.
 - h. Develop and maintain an ongoing method of presenting and displaying data generated by the test program so as to expedite analysis of results and possible correlation of test results with in-service generated data.
 - i. Determine the availability of in-service defected tubes and develop procedure for the testing and evaluation of same.
 - j. Carry out analysis of test results, defining margin of safety for artificially defected tubes in simulated in-service environment.
 - k. Carry out study to establish method of inducing cracks and wastage in tubing by means of corrosion and/or corrosion-fatigue under accelerated time conditions; such cracks and wastage to be similar to that found in tubes in-service.
3. Experimental Facilities and Program:
- a. Facilities: Testing laboratories and facilities at PNL, Richland, Washington.
 - b. Program: See item 2 above.
4. Project Status:
- a. Results to date: This is a new program.
 - b. Essential results: None reported as yet.

5. Next Steps: See Item 2 above.
6. Relation to Other Projects:
7. Reference Documents: Quaterly progress reports and topical reports.
8. Degree of Availability: All available through NTIS.

- 11.3. Prestressed concrete pressure vessel
(no entries)

11.3.1. Material properties

PROJECT TITLE : PRESTRESSED CONCRETE PRESSURE VESSEL WITH HOT LINER	CSNI INDEX NUMBER : 11.3.1
SPONSORING COUNTRY : AUSTRIA	ORGANISATION : SGAE, RFB, VÖEST-ALPINE
DATE INITIATED : 1969 DATE COMPLETED : 1978	PROJECT LEADER : Dipl.-Ing. W. Binner

Description :

Conventional concrete reactor pressure vessels have a cold liner covered by a thermal insulation. Such a liner can hardly be inspected and repaired. To fully utilise the safety potential of prestressed concrete vessels for nuclear installations, a concept was elaborated based on a hot liner in direct contact with the reactor coolant.

To demonstrate the feasibility of its concept, a large scale (12 m high, 4 m ϕ) model vessel for 100 bar internal pressure and a liner temperature of 300°C was built at the Austrian Research Center Seibersdorf. An extensive testing programme to prove the reliability of the system for reactor pressure vessels of HTR, LWR and GCFBR power stations is being executed.

<p><u>Title 1 (Original language)</u> Programma di ricerca sul comportamento del calcestruzzo per contenitori di reattori nucleari.</p>	<p><u>Classification</u> 11.3.1</p>
<p><u>Title 2 (English)</u> Research Program on Concrete for Nuclear Reactor Vessels</p>	<p><u>Country</u> ITALY <u>Sponsor</u> ENEL <u>Organisation</u> ENEL</p>
<p><u>Date initiated</u> 1970 <u>Date completed</u> - <u>Last updating</u> April 1977</p>	<p><u>Project Leader</u> P. Bertacchi</p>

Description:

1. General Aim

Determination of concrete properties to provide data for prediction of vessel behaviour.

2. Particular Objectives

- Systematic investigation on the behaviour of concrete subjected to multi-axial stresses.
- Determination of concrete strength after wet and dry thermal treatments
- Study of concrete creep.

3. Experimental Facilities and 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.

<u>Title 1 (Original language)</u> Programma di ricerca sul comportamento del calcestruzzo per contenitori di reattori nucleari	<u>Classification</u> 11.3.1
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Sonic methods have mainly been used with equipment for measuring propagation velocity of constant-frequency signals.

- A large number of specimens have been subjected to thermal treatment at temperatures of 90° and 120° C, both in dry and in moist ambients; moreover, they have been subjected to thermal cycling and to continuous treatment.

The effects of thermal treatment on the bending and compression strength and on Young's modulus (static and dynamic) have been determined.

5. Next steps.

- a.. Study of concrete creep.

The installation of automatic load control equipment at the Niguarda Laboratory, with the possibility of subjecting concrete prismatic and cylindrical specimens up to a max load of 100 tons, makes it possible to start systematic research in order to determine the specific surface of creep and to study the influence exerted on concrete creep by various factors, such as the shape and size of the specimens subjected to loads which are kept constant with time.

- b. Development of novel techniques (optical holography) for determining the state of strain and early stage of fissuring in concrete.

6. Relation to other Projects

Development of advance solutions for pre-stressed concrete pressure vessels (thin-wall solutions) (11.3.2 - 11.3.4)

7. Reference Documents

- R. Bellotti, "Joint Research Programme of ENEL-DSR/CEGB-CERL on concrete for Nuclear Reactor Vessels- Enel Contribution. State of the research as at December 1973. "CRIS n. 2425.
- R. Bellotti, P. Rossi, "New Prospects for Evaluating the Degree of Safety in Concrete Structures Subjected to Multiaxial Stresses". III - 5; Seminar on: Concrete Structures Subjected to Triaxial Stresses - 17 th - 19 th May, 1974 - ISMES - Bergamo.

11.3.2. Stress-strain analysis

Classification	
11.3.2/11.3.4	
<u>Title 1</u> Brudundersøgelse af Betontank-lågmodeller	COUNTRY Denmark SPONSOR DAEC, Risø ORGANIZATION DAEC, Risø
<u>Title 2</u> Overload behaviour and ultimate load capacity of PCRV-closures for BWR-plants.	<u>Project leader</u> S.I. Andersen Scientists:
<u>Initiated</u> (date) <u>Status:</u> concluded	<u>Completed:</u> 1976 <u>Last updating</u> (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

To be published in 1977 in Journal of the Engineering Mechanics Division, Proceedings of the American Society of Civil Engineers.

Structural Failure of Thick-Walled Concrete Elements.

N.S. Ottosen, S.I. Andersen.

Paper H 3/d, 4. Int. Conf. on Structural Mech. in Reactor Technology.
San Francisco (1977).

8. Degree of availability Project information restricted.

<p><u>Title 1 (Original language)</u> Sviluppo di soluzioni avanzate per i recipienti in pressione in calcestruzzo precompresso (soluzioni a pareti sottili).</p>	<p><u>Classification</u></p> <p>11.3.2 11.3.4</p>
<p><u>Title 2 (English)</u> Development of advanced solutions for Pre-stressed Concrete Pressure Vessels (Thin-wall solutions).</p>	<p><u>Country</u> ITALY</p> <p><u>Sponsor</u> ENEL</p> <p><u>Organisation</u> ENEL, ISMES, CISE</p>
<p><u>Date initiated</u> 1970</p> <p><u>Date completed</u> --</p> <p><u>Last updating</u> April 1977</p>	<p><u>Project Leader</u></p> <p>F. Scotto</p>

1. General Aim

To develop advanced economical and safe solutions for PCPVs by better exploiting the resistance of triaxially stressed concrete.

2. Particular Objectives

Reduction of the wall thickness of current PCPVs by applying novel design philosophies and tendon layouts.

Development of a pre-stressed concrete top lid and of a closure-and-sealing system similar to those adopted for steel pressure vessels.

3. Experimental Facilities and Programme

The proposed solutions are investigated by means of suitable design tools, such as physical and mathematical models of the structure and rheological models of the material (concrete).

Most of the tests are carried out at the ISMES Laboratory at Bergamo. Sophisticated small-scale physical models (scale 1:10, 1:20) have been developed, built and tested to cover the entire range of structural behaviour up to collapse.

In order to obtain additional information on the deformation states of the physical models a new approach, based on holographic techniques will be tried.

4. Project Status

a. Progress to date

The following models have been built and tested up to collapse:

- One thick-and three thin-wall models (scale 1:20) for a 300-MWe HTR.
- One thin-wall "continuous" model (scale 1:10) for a 1000-MWe BWR.
- Two thin and one "super-thin" models of the top lid (scale 1:10) for a 1000 MWe BWR.
- One integral model (main structure plus removable lid) scale 1:10 for a 1000 MWe BWR.

Use has been made of mathematical models for the selection of the optimal geometrical solutions by means of an interactive type of man-machine analysis.

A computer program was developed to take into account also the frictional interaction between the lid and the main cylindrical structure.

<u>Title 1 (Original language)</u> Sviluppo di soluzioni avanzate per i recipienti in pressione in calcestruzzo precompresso (soluzioni a pareti sottili).	<u>Classification</u> 11.3.2 11.3.4
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b. Essential Results

The proposed thin-wall solution can lead to a reduction in the PCPV weight from about 37,000 tons to 14,000 tons for a 300-MWe HTR, and from 8000 to 4000 tons for a 1000 MWe BWR.

5. Next Steps

- a. Experimental and theoretical verification of the designs.
- b. Recycling of the results of the physical model tests to the mathematical model.
- c. Testing of the final solution of the thin-walled PCPV by means of one or more physical models (scale 1:10).
- d. Application of thin walled concept to Gas Cooled Fast Breeder Reactors.

6. Relation to Other Projects

- a. "Research Program on concrete for nuclear reactor vessels" (11.3.1)
- b. ENEL-A. B. Atomenergy Agreement concerning "Collaboration for the Development of PCPV for LWRs".

7. Reference Documents

- F. Scotto, "Triaxial State of Stress of Thin-Walled PCPVs for HTGRs; Comparison with a Conventional Thick Solution". IABSE Seminar on Concrete Structures Subjected to Triaxial Stresses, Bergamo, 17-19 May, 1974.
- M. Fanelli et al., "Finite-Element Analysis of Prestressed - Concrete Pressure Vessels". IABSE Seminar, 17-19 May, 1974.
 - E. Fumagalli and G. Verdelli, "Small-Scale Models of PCPVs for High-Temperature Gas Reactors. Modelling Criteria and Typical Results". IABSE Seminar, 17-19 May 1974.
 - F. Scotto, "Thin Walled" Concept and a new top lid applied to the Scandinavian PCRV for a boiling reactor. Paper (H3/4) presented at III SMIRT - London, September 1 - 5, 1975.
 - R. Riccioni, G. Robutti and F. Scotto, "Finite Element Structural Analysis of a P. C. P. V. for a BWR. Paper (H2/7) presented at III SMIRT - London - September 1 - 5, 1975.
 - E. Fumagalli, G. Verdelli, "Research on PCPV for BWR"- Physical model as design tool - Main Results. Paper (H3/5) presented at III SMIRT London - September 1 - 5, 1975.

^ SMIRT - Conference on structural mechanics in reactor technology.

<u>Title 1 (Original language)</u> Sviluppo di soluzioni avanzate per i recipienti in pressione in calcestruzzo precompresso (soluzioni a pareti sottili).	<u>Classification</u> 11.3.2 11.3.4
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8. Degree of availability
Exchange of information can be arranged.
9. Budget
About $60 \cdot 10^6$ Lire/year.

11.3.3. Non-destructive testing, inspection,
surveillance
(no entries)

11.3.4. Destructive testing
(no entries)

11.4. Containment
(no entries)

11.4.1. Concrete structures
(no entries)

11.4.2. Steel structures
(no entries)

11.5. Coolant channels

<u>Classification:</u> 11.5	
Title 1 (Original Language): Untersuchungen zum Einfluß der Größe und Form von Kühlkanalblockaden auf die Kernnotkühlung in der Flutphase eines Kühlmittelstörfalles (PNS 4239 - I.1.3, Jahresbericht A 76)	COUNTRY: BRD
	SPONSOR:
	ORGANIZATION: GfK Karlsruhe Projekt Nukleare Sicherheit
Title 2 (English): Influence of the Size and Shape of Coolant Channel Blockages upon Core Cooling in the Flooding Phase of a LOCA	Project Leader: S. Malang/IRB
Initiated (Date): 1972	Completed (Date): 1979
Status: Continuing	Last Updating (Date): December 31st, 1976

1. General Aim

The influence of coolant channel blockages upon the cooling effect during the reflood phase of a loss of coolant accidents is investigated.

2. Particular Objectives

The objective of the program is to investigate the influence of size and shape of coolant channel blockages on the flow and the heat transfer conditions in the vicinity of the blockages. Not considered is the interaction between deformation and cooling conditions.

3. Research Program

The investigations will be done with a single rod, a row of 5 rods, and a bundle of 5 x 5 rods. The used indirect electrically heated rods have the same geometry like the fuel rods and a stepwise cosine shaped axial power profile.

The experiments are performed in three major steps concerning the shape of blockages:

- a) without blockages (reference tests),
- b) grid plate,

The free cross section for the coolant is varied by different diameters

of the boreholes in the center of the coolant subchannels.

- c) exchangeable balloons attached to the outer rod surface,
The outer diameter, the length, and the axial profile of the balloons are varied.

The mass flow of the flooding water and the back pressure at the outlet of the test section are kept constant for each test. The rod power is following the ANS-Standard for decay heat, where the level is a parameter. Test parameters are:

- a) initial rod temperature,
- b) mass flow of the flooding water,
- c) temperature of the flooding water,
- d) back pressure at the outlet of the test section,
- e) level of the decay heat.

The cladding temperatures and the pressure drops are measured. At three axial levels high speed movies are taken which serve as an important tool for the interpretation of the two phase flow.

4. Experimental Facilities, Computer Codes

The main components of the test rig are:

- water tank,
- pump,
- throttle valve,
- lower plenum,
- test section with electrically heated rods,
- upper plenum with pressure-regulating valve,
- thyristor controlled power supply,
- digital data acquisition system,
- high speed film camera with a flash light equipment.

An available computer code was improved and adjusted for the calculation of the heat transfer coefficient using the measured clad temperatures.

5. Progress to Date

After the shake-down period of the test loop and the data acquisition system the experiments with a row of 5 heater rods were started.

The first test series without blockages has been completed. It serves as baseline experiments.

Tests with a grid plate to simulate coolant channel blockages have been started. In the first step which has been completed the reduction in the coolant channel cross section area was 62 %.

6. Results

The analysis of the experiments was started. A comparison between the tests with free coolant subchannels and those with subchannels partially blocked by a grid plate show that at the same mass flow rate the heat transfer was always better behind the blockages. However, in front of the grid plate the heat transfer was nearly uninfluenced. That result is not relevant for rod bundles in which only a part of the subchannels is blocked. For other types of subchannel blockages the result is not applicable without modifications.

7. Next Steps

The tests with a row of 5 heater rods and a grid plate simulating coolant channel blockages are continued. In the next step the influence of balloons attached to the outer surface of the rods is investigated. Tests with a 5 x 5 rod array are prepared.

8. Relation with other Projects

FLECHT-Tests, performed by Westinghouse
Low pressure tests, performed by KWU.

9. References

Semiannual Reports of the Nuclear Safety Project.

10. Degree of Availability of the Reports

Unrestricted Distribution.

<u>Classification: 11.5</u>	
<u>Title 1 (Original Language):</u> Blockierte Kühlkanäle (RS 194 - I.1.3, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
<u>Title 2 (English):</u> Blocked Cooling Channels	Project Leader: H. Hein
<u>Initiated (Date):</u> 1. 1. 76 <u>Status:</u> Continuing	<u>Completed (Date):</u> 31. 12. 77 <u>Last Updating (Date):</u> 31. 12. 76

1. General Aim

Investigation on the effect of ballooned fuel claddings during a LOCA on the emergency cooling of a BWR.

2. Particular Objectives

Based on the measured heat transfer rates in the blocked region, the influence of the flow distribution on the heat transfer will be evaluated upstream and downstream of the blocked region as a function of the geometry of the blockage.

3. Research Program

- 3.1 Planning and preparation of blockage tests
- 3.2 Modification of the test facility for BWR geometry blockage tests
- 3.3 Assembly of the bundle in the rod bundle container
- 3.4 Pre-tests of mass distribution measurement techniques in the core
- 3.5 Performance tests
- 3.6 Evaluation of tests

4. Test Facilities

The BWR tests will be performed in a double-bundle test facility which was installed within the scope of Task RS 36. During test performance, the cooling channels of one bundle will partly be blocked by sheets. The second, parallel bundle remains unblocked.

The following parameters will be investigated: initial temperature, pressure, power, coolant mass-flow and degree of blockage.

5. Progress to Date

The tests will be performed on the 2nd BWR double bundle under the scope of R & D Task RS 36 C. The double bundle was prepared for the installation of blockage sheets and the type of blockage to be used was determined. The blockage provides for local blockages of 75 % and 39 % respectively. The blockage plates were fabricated.

The next step was the assembly of the blockage sheets in the 2nd BWR double bundle. The ξ -value of the blockage was determined for Reynold's numbers between 10^4 and 10^5 , followed by the assembly of the double bundle in the supporting structure (cage) and rod bundle container. The instrumentation lines were led from the container and sealed.

The flooding and spray systems were connected to the feed lines. Partial start-up of the plant and check-out of the measurement technique was initiated.

The construction of the test rig for developing and evaluating special measuring techniques is completed. The main components such as pump and power supply have been ordered.

6. Results

Measurement of the resistance coefficient of the blockage in the bundle resulted in a ξ -value of 5 ± 0.1 within the Reynold's number range of 10^4 to 10^5 .

The plant has nearly been completed so that start-up could be initiated.

7. Next Steps

- start-up of plant
- performance of 10 tests using the first blockage type
- evaluation of tests
- decision on 2nd blockage type
- tests using the 2nd blockage type
- erection of the model test rig
- performance of measurement technique tests.

8. Relation to Other Projects

Preparatory work has been performed within the scope of R & D Tasks RS 36, RS 36 A, RS 36 B, RS 36 C.

Experiments pertaining to fuel rod performance are being conducted by both the project "Nukleare Sicherheit" at GfK as well as within R & D Task RS 107 and RS 185.

9. References

10. Degree of Availability

<p><u>Title 1 (Original language)</u> Effetti dello scoppio di tubo a pressione CIRENE.</p>	<p><u>Classification</u> 11.5</p>
<p><u>Title 2 (English)</u> CIRENE power channel pressure tube burst tests.</p>	<p><u>Country</u> ITALY <u>Sponsor</u> : CNEN <u>Organisation</u> : CISE</p>
<p><u>Date initiated</u> October 1970 <u>Date completed</u> 1978 <u>Last updating</u> May 1977</p>	<p><u>Project Leader</u> G. Possa</p>

1. General aim: to investigate the consequences of the hypothetical explosion of a power channel pressure tube for the CIRENE reactor.
2. Particular objective: to measure the pressure waves produced by explosion in the D₂O tank and to determine the associated stresses on major mechanical structural components.
3. Experimental facility and programme
 - 3.1. Experimental facility
 - BETULLA: facility located in CCR Euratom at Ispra including a pressure vessel (with nearly CIRENE dimensions) for explosion containment.
 - 3.2. Programme
 - 3.2.1. Preliminary burst tests to individuate most relevant plant parameters.
 - 3.2.2. Burst tests in full height scale with a dummy explosion tube (full length rupture).
 - 3.2.3. Burst tests as in 3.2.2. with explosion tube and adjacent target channel simulating in full scale (dimensions and structural materials) the CIRENE power channel.
 - 3.2.4. Numerical simulation with existing computer codes of the transient triggered by explosion.

Remark Burst tests for determination of the consequences of a hypothetical pressure tube rupture due to hot spot have been canceled.
4. Project status
 - 4.1. Progress to date
 - (3.2.1.): completed
 - (3.2.2.): completed
 - (3.2.3.): completed
 - (3.2.4.): in the initial stage

<u>Title 1 (Original language)</u> Effetti dello scoppio di tubo a pressione CIRENE.	<u>Classification</u> 11.5
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4.2. Essential results

- Basic understanding of the dynamics of underwater explosions in a confined volume.
- Understanding of the rupture process of a pressurized Zircaloy tube artificially defected, under typical conditions of a pressure tube reactor.
- Gathering information on explosion stresses in reactor structural components.

5. Next steps

Programme item 3.2.4.

6. Reference documents

- 1) M. Famiglietti, A. Parmeggiani, G. Possa, L. Galbiati: "Pressure Burst Due to Power Channel Explosion in a Pressure Tube Reactor", presented at the 3rd SMIRT Conference, London 1-5 Sept. 1975.

7. Degree of availability: to a limited extent.

12. QUALITY ASSURANCE
(no entries)

12.1. Formulation of quality assurance system

<u>Classification:</u> 12.1	
<u>Title 1 (Original Language):</u> Statusbericht Qualitätssicherungssystem - Darstellung des Istzustandes - (RS 124 - II.3.1, Jahresbericht A 76)	<u>COUNTRY:</u> BRD
	<u>SPONSOR:</u> BMFT
	<u>ORGANIZATION:</u> KWU Erlangen
<u>Title 2 (English):</u> Status Report of the Quality Assurance System	<u>Project Leader:</u> Dr. Kaden
<u>Initiated (Date):</u> 1. 8. 75	<u>Completed (Date):</u> 31. 1. 77
<u>Status:</u> Completed	<u>Last Updating (Date):</u> 31. 12. 76

1. General Aim

In a study the present state of the quality assurance system for design, manufacturing and examination, procurement, shaping erection and commissioning of the components (NDES) including nuclear auxiliary and ancillary components and the containment will be described.

2. Particular Objectives

All components were considered, which are used in the nuclear steam generating system.

3. Research Program

- a) Listing of various steps of the work for the representation of the present state.
- b) Detailed description of the activities, the boundary conditions and the quality assurance securing.

4. Experimental Facilities

No experimental facilities necessary.

5./6. Progress to Date and Results

The progress report was revised and discussed with members of the IRS. The revised version was completed.

7. Next Step

The work has been completed.

8. Relation with Other Projects

9. References

10. Degree of Availability

12.2 Fabrication methods
(no entries)

- 12.3. Non-destructive testing, inspection, surveillance
(for pressure structures and components: see
under relevant sections 7 and 11)

PROJECT TITLE : Stress Wave Emission Studies of Materials and Structures	CSNI INDEX NUMBER : 12.3
SPONSORING COUNTRY : Australia	ORGANISATION : Australian Atomic Energy Commission, Private Mail Bag, Sutherland, NSW, 2232
DATE INITIATED : January, 1973 DATE COMPLETED : Continuing	PROJECT LEADER : Dr. K. Brown Materials Division

Description :

Objective:

To study the stress wave emissions (SWE) arising from specific deformation and fracture events in materials and structures in an attempt to improve the usefulness of SWE as a non-destructive testing tool.

Proposed Investigations:

- 1) The techniques of characterisation and analysis of SWE signals will be progressively refined.
- 2) Deformation and fracture phenomena in well-characterised zirconium and steel specimens will be studied.

OECD NUCLEAR ENERGY AGENCY - 38, Boulevard Suchet, 75016 PARIS
Telex : 630668 AEN NEA - Tel : 524 82-00

14.560

Titre Détermination fine de la dimension de défauts par mesure de temps de parcours ultrasonores.	Pays : FRANCE Organisme directeur : CEA/DSN
Titre (anglais) Accurate flow size determination by measurement of ultrasonic delay time.	Organisme exécuteur : Ecole Centrale Responsable : A.C.PROT (DSN-SETSSR)
Date de démarrage : 01/01/77 Date prévue d'achèvement : 31/12/77 Etat actuel : A lancer Dernière mise à jour : 23/11/76	Scientifiques : P.AZOU D. de VADDER

Objectif général :

Il s'agit d'une étude analytique associée à des essais de laboratoire, destinés à démontrer la faisabilité d'une méthode de dimensionnement fine des défauts décelés par ultrasons.

Objectifs particuliers :

Montrer analytiquement et vérifier expérimentalement que les dimensions d'un défaut peuvent être atteintes par mesure des temps de parcours de l'onde ultrasonore au cours du balayage du défaut par le traducteur ultrasonore.

Titre Utilisation de calculateurs dans les systèmes de protection.	Pays : FRANCE Organisme directeur : CEA
Titre (anglais) Computer protection system	Organisme exécuteur : CEA/LETI (Grenoble) Responsable : Mr. TOURNIER
Date de démarrage : 1/1/75 Date prévue d'achèvement : 1/1/79 Etat actuel : Etude en cours Dernière mise à jour : 21/12/76	Scientifiques :

Objectif général :

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

Objectif particulier :

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

Prochaines étapes :a) Elaboration des règles pour la programmation d'un tel système.

Elles doivent contenir et la modularisation stricte des programmes à utiliser et une liste des éléments de programmation à ne pas utiliser à cause de la fiabilité des programmes.

b) Programmation des autotests des calculateurs .

Elle doit être effectuée en étudiant et optimisant l'efficacité de la détection des défaillances. Pouvoir vérifier si toutes les défaillances simples peuvent être détectées avec ce programme de test, elles doivent être simulées ou provoquées au niveau du matériel.

.../...

- c) Etudes sur la fiabilité des modules et composants à utiliser et surtout sur la sécurité et disponibilité du système entier choisi. Ces études montreront sans doute encore des points faibles dans le système qui doivent être éliminés par la suite et ils devront fournir la preuve de la capacité du système à remplir les exigences demandées.
- d) Un prototype du système entier informatique doit être réalisé et entièrement testé à l'aide d'un dispositif de test, par exemple un calculateur hybride (3) qui permettrait d'effectuer, en principe toutes les combinaisons des valeurs d'entrée, mais en fait, à cause du grand nombre des combinaisons (10^{64}) seulement une petite partie de ces possibilités.
- e) Statistique des défaillances déjà apparues qui permettrait ensemble avec une étude des défaillances (arbres de défaillance, ambiguïté des défaillances, probabilité d'occurrence des défaillances particulières, etc...) d'encore améliorer le système entier.

Documents de référence :

"Propositions d'un système de protection utilisant des calculateurs. Applications au contrôle des réacteurs nucléaires", - Note Technique LETI/MCTE N° 1206, HD.MAIER, P.DARIER.

Titre Surveillance en continu par émission acoustique du circuit primaire des réacteurs à eau.	Pays : FRANCE Organisme directeur : CEA
Titre (anglais) Continuous surveillance of water reactors primary circuits using acoustic emission	Organisme exécuteur : CEA/DTECH Responsable : CONTRE (DTECH-STA)
Date de démarrage : 01/76 Date prévue d'achèvement : 12/80 Etat actuel : En cours Dernière mise à jour : 1/4/77	Scientifiques : R. SAGLIO ASTY

Objectif général :

Définir et construire un matériel d'émission acoustique spécifique adapté à la surveillance en continu du circuit primaire des réacteurs à eau, c'est-à-dire susceptible de fonctionner dans l'ambiance d'un réacteur en fonctionnement.

Objectifs particuliers :

Permettre en 1977 la surveillance du dôme du pressuriseur de FESSENHEIM 1.

Installations expérimentales et programme :

Equipements électronique :

- Microprocesseur
- Echelle de comptage
- Electronique d'acquisition.

Titre Déttection des fissures dans les rebords internes de tubulures.	Pays : FRANCE Organisme directeur : CEA/DSN
Titre (anglais) Crack detection of the nozzle inner edges.	Organisme exécuteur : CEA/DTECH-STA Responsable : A.C.PROT (DSN-SETSSR)
Date de démarrage : 01/01/77 Date prévue d'achèvement : 31/12/78 Etat actuel : A lancer Dernière mise à jour : 23/11/76	Scientifiques : R.SAGLIO (STA)

Objectif général :

Le rebord interne des turbulures est une zone particulièrement sollicitée dans une cuve de réacteur. L'étude consiste à rechercher la méthode optimale pour détecter et éventuellement dimensionner les défauts qui peuvent prendre naissance pendant le fonctionnement.

Objectifs particuliers :

- Evaluer de manière critique les méthodes actuellement envisagées en particulier ultrasons et courants de Foucault.
- Etudier la faisabilité d'un contrôle en ultrasons focalisés.
- Etudier la possibilité d'application aux inspections périodiques de cuves de réacteurs nucléaires.

Installations expérimentales et programme :

Utilisation des moyens actuels de la STA et en particulier de la maquette de piquage SENA.

Prochaines étapes :

Démonstration de la possibilité de contrôle par traducteurs ultrasonores focalisés ou/et courants de FOUCAULT (fin 1977)

Titre Développement de méthodes de contrôle par ultrasons par l'extérieur des circuits primaires PWR.	Pays : FRANCE Organisme directeur : CEA
Titre (anglais) Development of ultrasonic testing methods from the outside of PWR primary circuits.	Organisme exécuteur : CEA/DTECH Responsable : M. CONTRE (DTECH-STA)
Date de démarrage : 01/76 Date prévue d'achèvement : 10/80 Etat actuel : en cours Dernière mise à jour : 1/4/77	Scientifiques : M. SAGLIO

Objectif général :

Permettre le contrôle par l'extérieur du circuit primaire des réacteurs PWR et plus particulièrement des cuves.

Objectifs particuliers :

Etudier les méthodes de contrôle par ultrasons propres à permettre un suivi précis des défauts éventuellement détectés par l'intérieur, sans nécessiter le démontage des internes.

Installations expérimentales et programme :

- Examen des modifications des plans de Génie Civil permettant cette inspection par l'extérieur.
- Etude des méthodes de contrôle
- Eventuellement étude d'un avant projet de machine d'inspection.

Etat de l'étude :

1) Avancement à ce jour :

- Etude des problèmes d'immersion locale - réalisation de joints d'étanchéité.
- Diminution des encombrements des traducteurs par utilisation de miroirs.

2) Résultats essentiels :

Mise au point d'un prototype expérimental de faisabilité de contrôle par l'extérieur en immersion locale.

.../...

Prochaines étapes :

Miniaturisation de la cuve d'immersion locale
Etude de traducteurs adaptés aux géométries et permettant de
satisfaire aux exigences d'encombrement.
Avant projet de machines d'inspection.

Titre Programme européen de contrôle de tôles fortes par ultrasons.	Pays : FRANCE Organisme directeur : CEA/DSN
Titre (anglais) European program of ultrasonic testing of heavy section steel-plates.	Organisme exécuteur : CEA/DTECH-STA Responsable : A.C.PROT (DSN-SETSSR)
Date de démarrage : 01/03/77 Date prévue d'achèvement : 01/07/78 Etat actuel : A lancer Dernière mise à jour : 23/11/76	Scientifiques : R.SAGLIO (STA)

Objectif général :

Participation à un programme européen de contrôle de tôles de forte épaisseur du programme HSST, mises à disposition par les USA.

Objectifs particuliers :

Détection des défauts volontairement introduits dans les tôles en utilisant la technique des traducteurs focalisés et éventuellement l'holographie acoustique.
 Un tel essai devrait constituer une base pour la qualification de la méthode mise au point au CEA, puisqu'il sera suivi de la découpe des tôles et d'une corrélation entre examen ultrasonore et macro ou micrographies.

Installations expérimentales et programme :

Les installations sont celles actuellement disponibles à la section des techniques lancées (STA).
 - Cuve de contrôle par immersion (locale ou non).
 - Pantographe
 - Appareils et traducteurs à ultrasons.

Etat de l'étude :

Avancement à ce jour :

Les types de contrôles sont définis. Les tôles doivent être mises à disposition du 28 février au 30 juin 1977.

.../...

Prochaines étapes :

Contrôle des 3 tôles HSST (2ème semestre 1977).

Titre Moyen d'inspection de circuit primaire de réacteurs.	Pays : FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Nuclear reactor primary circuit inspection means.	Organisme exécuteur : CEA - DPR - STEPPA
	Responsable : A.C.PROT (DSN - SETSSR)
Date de démarrage : 1/1/77 Etat actuel : A lancer	Date prévue d'achèvement : 31/12/78 Dernière mise à jour : 22/11/76 Scientifiques : J.P.VERTUT (DPR)

Objectif général :

Les dispositifs mis au point pour le contrôle de cuves ne constituent qu'une partie de la solution des problèmes d'inspection.

Il reste à définir des moyens propres à atteindre de manière autonome des endroits contaminés et peu accessibles. Cette étude vient en appui de celle commencée en 1976 par la section EMH.

Objectifs particuliers :

Il s'agit essentiellement d'étudier un dispositif autonome susceptible de se rendre dans des endroits inaccessibles ou difficilement accessibles. (soudure de tuyauteries, boîte à eau de G.V., cuve de réacteur rapide) pour effectuer des inspections de divers types : ultrasons, courants de Foucault, visuel par télévision etc...) et d'en démontrer la faisabilité.

Etat de l'étude :

Avancement à ce jour :

Des schémas de principe ont été étudiés par le STEPPA au cours de l'année 1976 et serviront de base à l'étude envisagée.

Prochaines étapes :

- Etude d'un dispositif autonome (été 77)
- Maquette de faisabilité. (fin 77)

Titre Etude des filtrages optiques de films radiographiques	Pays : FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Study of optical filtering of radiographic films.	Organisme exécuteur : CETIM
	Responsable : A.C.PROT (DSN-SETSSR)
Date de démarrage : 19/5/76 Etat actuel : en cours	Date prévue d'achèvement : 19/11/77 Dernière mise à jour : 23/11/76 Scientifiques : FLAMBARD PARASKEVAS

Objectif général :

Recherche d'un procédé de filtrage optique permettant d'augmenter le contenu d'informations délivrées par les films radiographiques.

Objectifs particuliers :

Amélioration du contraste dans la détection soit de fissures mal orientées soit de présence d'un bruit de fond (pièce radioactive et/ou en ambiance radioactive).

Etat de l'étude :

Avancement à ce jour :

L'étape est en cours de finition. Un document sera diffusé qui permettra après analyse (étape 2) de lancer l'étude proprement dite.

Prochaines étapes :

- 1) Etablissement d'un document faisant le point des connaissances en la matière (fin 76)
- 2) Choix des orientations (fin 77)
- 3) Lancement des études correspondantes.
- 4) Choix du procédé à développer
- 5) Etude de ce procédé.

Titre Etude de l'influence des divers revêtements sur le contrôle ultrasonore par l'extérieur des circuits primaires de réacteurs.	Pays : FRANCE Organisme directeur : CEA/DSN
Titre (anglais) Study on the effect of coatings on the ultrasonic testing of primary circuits of nuclear reactors from the outside.	Organisme exécuteur : CEA/DTECH-STA Responsable : A.C.PROT (DSN-SETSSR)
Date de démarrage : 01/01/77 Date prévue d'achèvement : 31/12/78 Etat actuel : A lancer Dernière mise à jour : 23/11/76	Scientifiques : F.DUBOIS (STA)

Objectif général :

Il s'agit d'évaluer l'influence des peintures actuellement utilisées pour la protection des cuves de réacteur, sur les possibilités de contrôle par ultrasons.

Objectifs particuliers :

- Il est nécessaire d'évaluer les modifications introduites par l'environnement (température, irradiation) sur les qualités d'adhérence et de résistance à l'abrasion de peintures afin de connaître l'influence possible sur le couplage des traducteurs ultrasonores.
- Eventuellement cette étude peut déboucher sur un "meilleur choix".

Prochaines étapes :

- Réalisation d'éprouvettes métalliques représentatives.
 - Essais ultrasonores avant influence de l'environnement.
 - Essais en température et sous irradiation.
 - Essais ultrasonores comparatifs.
- (Mise en route du programme fin 1977)

Titre Détectabilité par ultrasons des défauts en compression.	Pays : FRANCE Organisme directeur : CEA/DSN
Titre (anglais) Ultrasonic detectability of flaws under compression.	Organisme exécuteur : CEA/DTECH-STA Responsable : A.C.PROT (DSN-SETSSR)
Date de démarrage : 01/07/77 Date prévue d'achèvement 31/12/78 Etat actuel : A lancer Dernière mise à jour : 23/11/76	Scientifiques : R.SAGLIO (STA)

Objectif général :

Etude de l'influence du champ de contraintes sur la détectabilité des défauts lors d'un contrôle par ultrasons.

Objectifs particuliers :

Il s'agit d'étudier, sur éprouvettes de traction représentatives (forte épaisseur), l'influence du champ de compression sur la détectabilité de défauts connus (détectés par ultrasons en l'absence de contrainte) et d'apprécier comment varie le dimensionnement de ces défauts en fonction de la contrainte appliquée.

Prochaines étapes :

- Détection de défauts réels introduits volontairement dans des éprouvettes de traction.
- Etude de l'évolution de ces défauts au cours d'essais de fatigue par traction-compression.
- Dimensionnement pendant les phases de traction puis compression.

Relation avec d'autres études :

Etude probabiliste de la rupture d'une cuve de réacteur nucléaire.

Titre Utilisation de la neutronographie pour l'inspection en service des réacteurs nucléaires et le contrôle non destructif des composants du coeur.	Pays : FRANCE Organisme directeur : CEA/DSN
Titre (anglais) Use of neutron radiography in in-service inspection nuclear reactor and non destructive testing of core composants.	Organisme exécuteur : CEA/DSN-SEESNC Responsable : M.HOUELLE
Date de démarrage : 01/01/73 Date prévue d'achèvement : 31/12/78 Etat actuel : Etude en cours Dernière mise à jour : 08/11/76	Scientifiques :

Objectif général :

Le but de ce programme est d'effectuer des essais de contrôle par neutronographie de composants de réacteurs. Le mini-réacteur pour neutronographie MIRENE, mis en service en 1976, permettra de mener à bien cette action. L'installation est équipée de 2 faisceaux collimatés de neutrons qui serviront à effectuer les essais d'examens de pièces.

Objectifs particuliers :

Valeur du procédé pour :

- 1) L'examen de pièces épaisses en acier.
- 2) L'examen d'éléments combustibles de réacteurs à eau.
- 3) L'examen de matériels divers (ensembles colles).

Installations expérimentales et programme :

Réacteur pour neutronographie MIRENE.

Etat de l'étude :

- 1) Avancement à ce jour :

Le réacteur MIRENE a divergé en novembre 1976 et sera disponible pour des examens de pièces à partir de décembre 1976.

- 2) Résultats essentiels :

Les essais d'examens effectués antérieurement sur une installation semblable à MIRENE ont donné les résultats essentiels suivants : Ce type de réacteur installé près de deux réacteurs de puissance, PHENIX et RAPSODIE, permet un contrôle neutronographique satisfaisant du combustible nucléaire. Des essais effectués avec cette même installation ont montré qu'il était possible de contrôler des pièces en acier de 5 cm d'épaisseur.

.../...

Prochaines étapes :

Essais d'examen de pièces test en acier et de pièces diverses de réacteur électrogène, susceptibles d'être démontées et transportées pour le contrôle dans une installation du type MIRENE.

Relation avec d'autres études :

Comparaison avec les autres méthodes de contrôle non destructif, en ce qui concerne l'examen de composants de réacteur et l'examen d'objets et de matériels divers.

Documents de référence :

- "Source utilisant la bouffée de neutrons produite par un saut de réactivité dans un réacteur", M.HOUELLE - supplément au Bulletin d'Information de l'ATEN N° 90 (juillet - août 1971).
- "Radiography with Neutrons", M.HOUELLE, C.MERCIER, H.REVOL - British Nuclear Energy Society - Conference at the University of Birmingham - 10/11 september 1973.
- "Neutronographies de pièces épaisses en acier", M.HOUELLE - Rapport SEESNC N° 123 - février 74.
- "Rapport provisoire de sûreté du miniréacteur pour neutronographie MIRENE", - Rapport SEESNC N° 122, 1974.

<u>Classification:</u> 12.3	
<u>Title 1 (Original Language):</u> Schallemissionsmessungen an bruchmechanischen Proben (Ermüdungsrisse) (RS 191-II.3.2, Jahresbericht A 76)	COUNTRY: BRD
	SPONSOR: BMFT
	ORGANIZATION: Battelle-Institut e.V.
<u>Title 2 (English):</u> Acoustic Emission Measurements on Fracture Toughness Specimens (Fatigue Cracks)	<u>Project Leader:</u> Dr. J. Eisenblätter, H. Jöst
<u>Initiated (Date):</u> January 1, 1976 <u>Status:</u> Continuing	<u>Completed (Date):</u> March 31, 1977 <u>Last Updating (Date):</u> December 1976

1. General Aim

Correlations are to be established between the measured values of AE and the type and size of and stresses on various defects with a view to using acoustic emission (AE) measurements for the inspection of pressure vessels.

2. Particular Objectives

At first all the investigations are conducted on one type of defect (fatigue crack). The examinations include the materials of the pressure vessel of light-water reactors, i.e. the ferritic steel 22NiMoCr37 and the cladding material austenite X10CrNiNb189 as well as materials of the sodium-cooled fast breeder reactor, i.e. the austenites X6CrNi1811 and X6CrNiMo1713, the ferritic steels 8CrMoNiNb910 and X12CrMo91, and "Incoloy 800" (X10NiCrAlTi3220).

To be able to determine the types of acoustic waves which are excited at the surface of thick-walled structural parts when the AE source is located at the surface opposite the transducer or within the wall, the propagation of AE signals is measured.

3. Research Program

- 3.1 Preparation of specimens and construction of the experimental setup
- 3.2 Design of the data recording system
- 3.3 Investigations on fracture toughness specimens
- 3.4 Reduction of data and determination and optimization of the required correlations between measured and desired values
- 3.5 Investigation of the acoustic wave propagation in thick-walled structural parts

4. Experimental Facilities

Ad 3.2: Construction of a measuring facility for calibrating narrow- and broad-band AE transducers by means of the reciprocity method with two transducers, which permits absolute calibration (voltage/sound pressure).

Assembly of the measuring system for AE data such as count rate, total count and amplitude and energy distribution.

Assembly or construction of experimental facilities for detecting and monitoring crack propagation during fracture toughness testing by optical measurement of the crack growth, measurement of the crack opening displacement (COD) and examination of the fracture surfaces.

Ad 3.3: A servo-hydraulically controlled test unit (load capacity 220 Mp) which has been made suitable for AE measurements by special interference noise eliminating measures is available for the experiments.

Ad 3.5: Assembly of a measuring facility for determining wave propagation by means of conventional instruments such as pulse generator, amplifier, oscillograph and multi-channel source location unit. A steel plate with polished surface and with the dimensions 1800 x 900 x 80 mm has been prepared; in addition, a large vessel is available for the experiments.

5. Progress to Date

Ad 3.1: Preparation of modified compact tension specimens from 22NiMoCr37 and heat treatment of these specimens in order to obtain the required microstructures: 100 percent low temperature bainite; 40 percent bainite, balance ferrite and pearlite (grain size ASTM 7-8), 40 percent bainite, balance ferrite and pearlite (ASTM 2-3).

The hardness values determined and first investigations on the heat-treated specimens showed that the materials did not meet the usual requirements with respect to mechanical parameters. For this reason some specimens with the structure of 100 percent low temperature bainite were subsequently subjected to an additional heat treatment in order to achieve higher toughness values.

Design and construction of a special chucking device without pin joints (Fig. 1) to avoid the friction noise caused by the pins of usual chucking devices for compact tension (CT) specimens.

Ad 3.2: Evaluation of methods for the exact calibration of AE transducers, using two different procedures. A broad-band transducer made of lithium sulfate hydrate was calibrated in an under-water sound-field which was measured by a calibrated hydrophone and by the reciprocity method (cf. 4.). Routine calibration of a large number of narrow- and broad-band transducers consisted in comparing these transducers with the above-mentioned broad-band one.

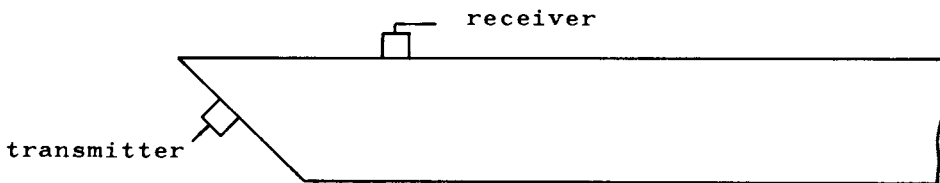
During the fracture toughness experiments the AE signals were measured by narrow- and broad-band transducers and partly stored on magnetic tape. The AE count rate was directly recorded with coarse amplitude distribution analysis, while the magnetic tapes were evaluated subsequently. Some experiments also included source location measurements, measurements of the amplitude distribution and frequency spectra,

both under cyclic and tensile loading.

During each experiment the crack length was determined with the methods outlined in Section 4.

Ad 3.3: The load and the load frequency were varied during the experiments in order to be able to study the resulting effects on AE. A total of 17 fracture toughness specimens (modified 2" CT specimens) made from steel 22NiMoCr37 of different micro-structure were investigated during cyclic and tensile loading, using two temperatures (RT and 60°C).

Ad 3.5: Transmission of confined transversal and longitudinal wave pulses of the frequency of 1, 2 and 4 MHz from an oblique end surface of the test plate (cf. drawing below).



Recording of body and surface waves on the surface.

Investigation of the following special cases:

- near field of ultrasonic probes (distance-dependent phase velocity)
- transversal waves from ultrasonic probe with angles of incidence in the region of the critical angle of total reflection of 33.2° (strong angle-dependence of the phase shift and of the shift of the reflected wave beam)
- transversal waves from normal point sources in the angular range between 34° and 37° (strong angle-dependence of the phase of the incident wave)

6. Results

Ad 3.2: When the elastic properties of the transmission medium (steel) are known, the reciprocity method enables absolute calibration of the transducer sensitivity (in dB resp. $1 \text{ V}/\mu\text{bar}$).

Figs. 2 and 3 show the sensitivity values of various commercial broad-band transducers (Fig. 2) and of several Battelle-developed and commercial narrow-band transducers (Fig. 3). Comparing the results achieved by calibrating broad-band and narrow-band transducers of various manufacturers shows that

- all these transducers are considerably less sensitive than stated by the manufacturers,
- the sensitivity decreases with increasing frequencies also in the case of broad-band transducers
- the resonant transducers designed by Battelle-Frankfurt, which have a natural frequency of 70 and 140 kHz, are by far the most sensitive ones.

It should be noted, however, that this calibration relates only to longitudinal waves.

Ad 3.3: Fig. 4 shows the fracture surfaces of the materials at different quenched and tempered conditions; the differences in toughness are clearly obvious. The fracture surfaces in Figs. 4a and 4b (40 % low-temperature bainite, balance ferrite and pearlite, ASTM 2-3 or 7-8) and Fig. 4c (100 % low-temperature bainite) are characteristic of brittle material behavior, while Fig. 4d shows the fracture surface of a tough material (additional heat treatment). The great difference in toughness values is also evident from the diagram in Fig. 5 which shows the load versus the crack opening displacement (COD).

Figs. 6 and 7 depict the AE count rate recorded at four respectively two different discriminator thresholds and the COD during rupture of a brittle specimen (corresponding to Fig. 5, curve a) and a tough specimen (corresponding to Fig. 5, curve b). Two regions in which AE occurs can be clearly distinguished. These two regions are separated at the upper load P_0 during cyclic loading. AE in the lower load region, which has its maximum at about the medium load P_m , is definitely caused by friction between the crack surfaces. The increase in AE above P_0 , which increases until fracture occurs, is attributed to micro-cracking processes at the crack

tip. The amplitudes of these AE signals are considerably higher than those of AE originating from friction processes.

Fig. 8 shows a distinct increase in the AE count rate with decreasing load frequency during cyclic loading, which clearly demonstrates that the AE below P_0 results mainly from friction (slip-stick effect). The load frequency was varied between 0.0016 and 1.66 Hz in the experiment.

Ad 3.5: The amplitude of the surface waves is about 50 dB lower than that of the incident body waves. Also in the special cases mentioned in section 5 intensive surface waves were not observed.

7. Next Steps

Ad 3.1: Procurement of the materials used for the sodium-cooled fast breeder reactor and preparation of specimens; construction of chucking devices

Ad 3.2: Calibration of transducers with shear and surface wave

Ad 3.3: Performance of the remainder of the planned experiments

Ad 3.4: Evaluation of the results with a view to establishing correlations between measured AE values and characteristic crack parameters

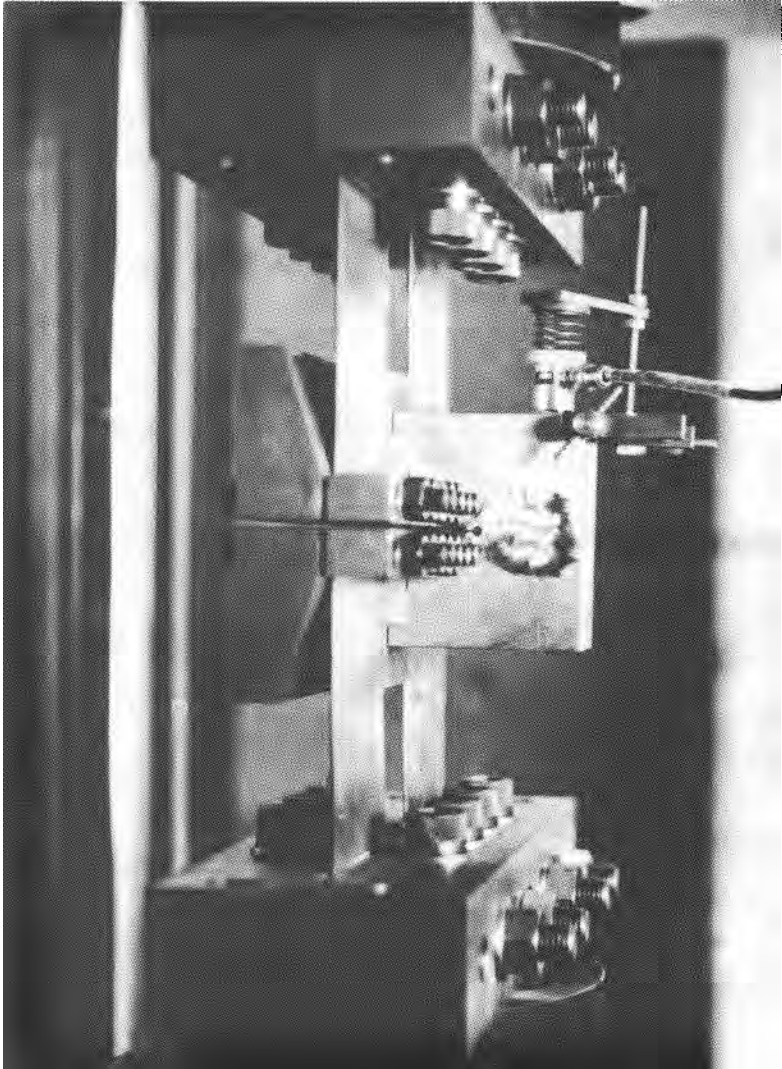
Ad 3.5: Additional investigations on the test plate, the waves being transmitted from a point source (as a function of the distance between transmitter and receiver), location of AE sources; investigations on a vessel

8. Relation to Other Projects

Continuation of the work under projects RS 0031 to RS 0031 D; cooperation with IzfP, Saarbrücken, and IfaM, Bremen (RS 196)

9. References

10. Degree of Availability of the Reports



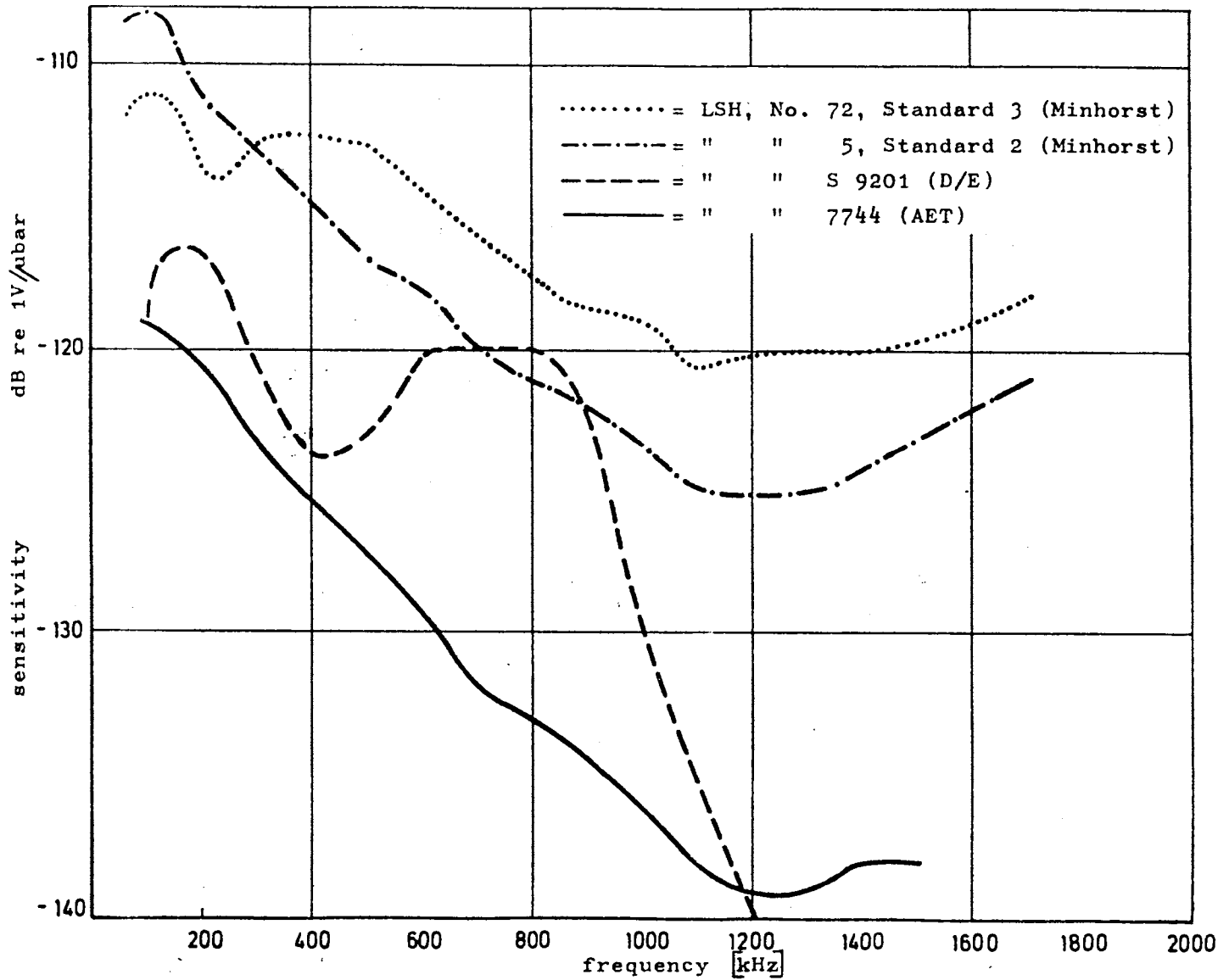


Fig. 2: Reciprocity calibration of broad-band transducers

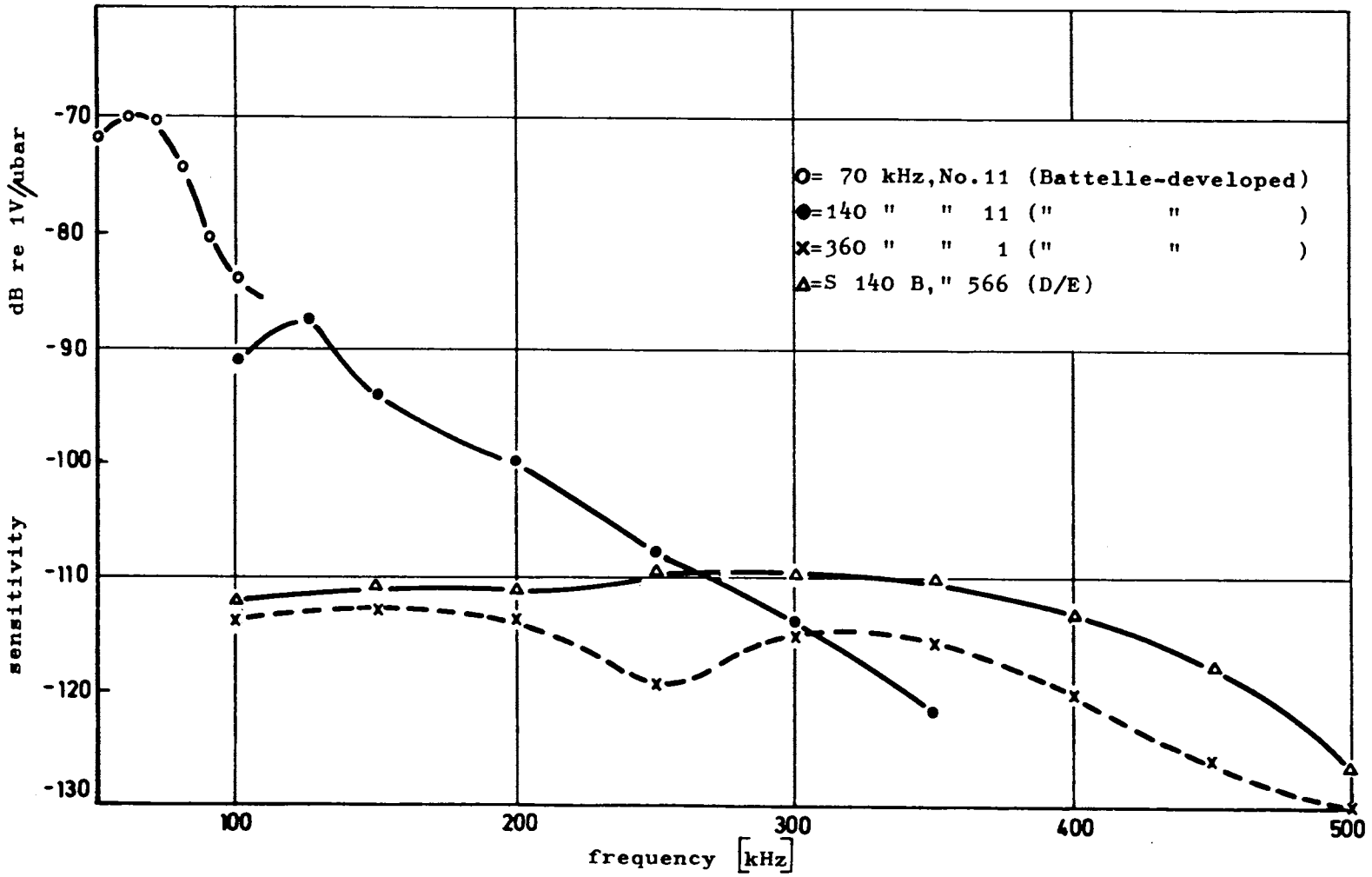


Fig. 3: Reciprocity calibration of narrow-band transducers

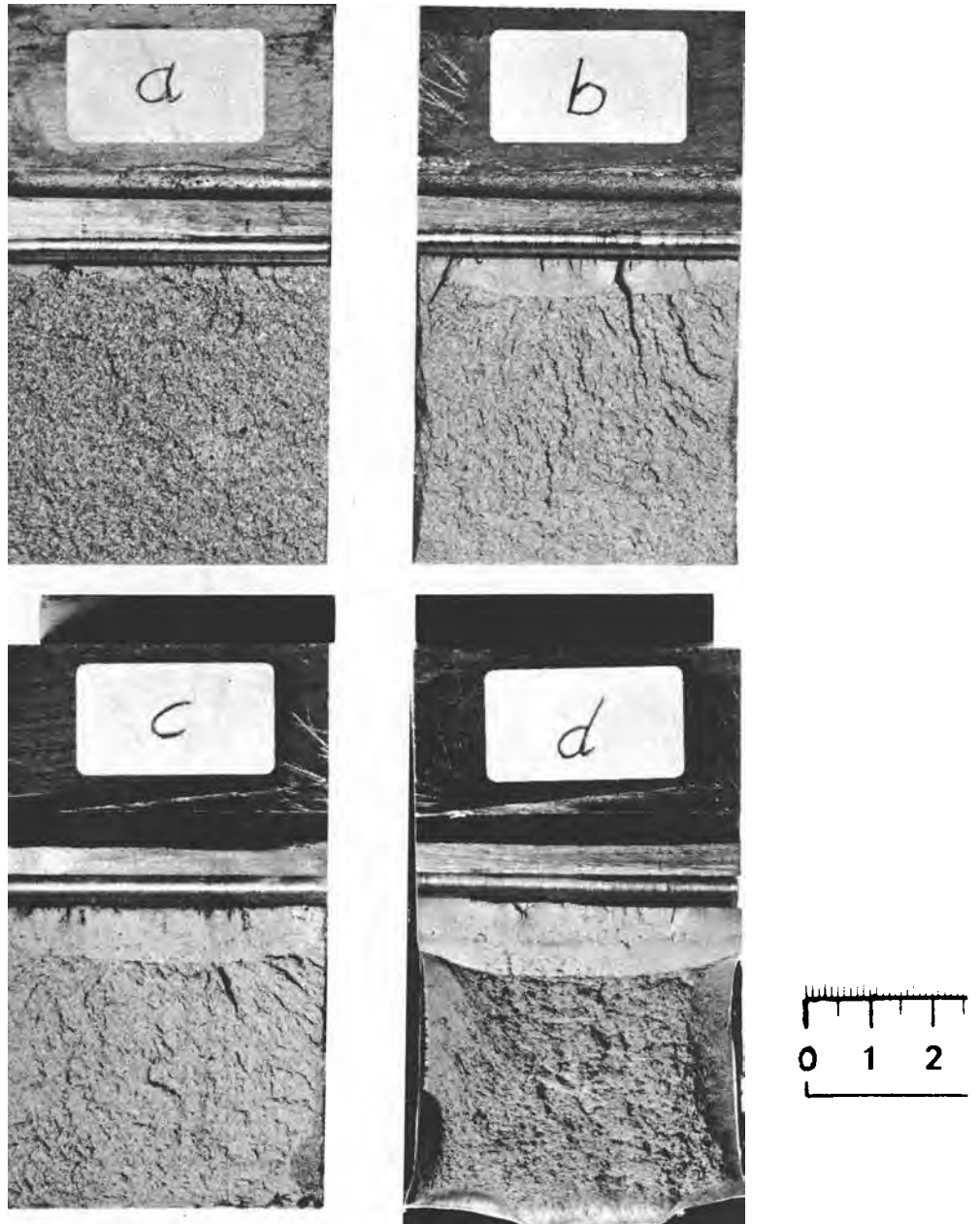


Fig. 4: Fracture surfaces of specimens from 22NiMoCr37 after different heat treatments

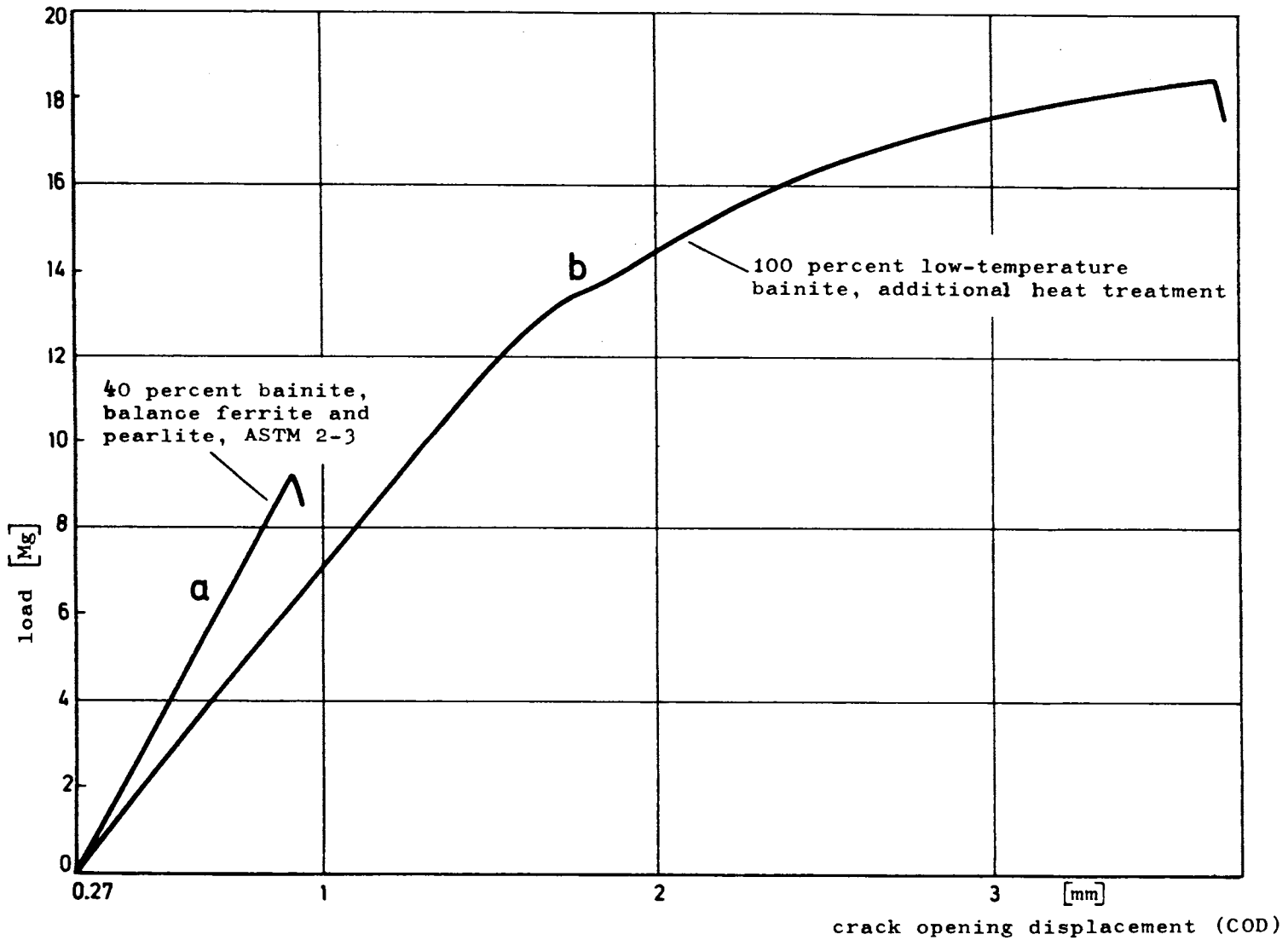


Fig. 5: Load-COD diagrams

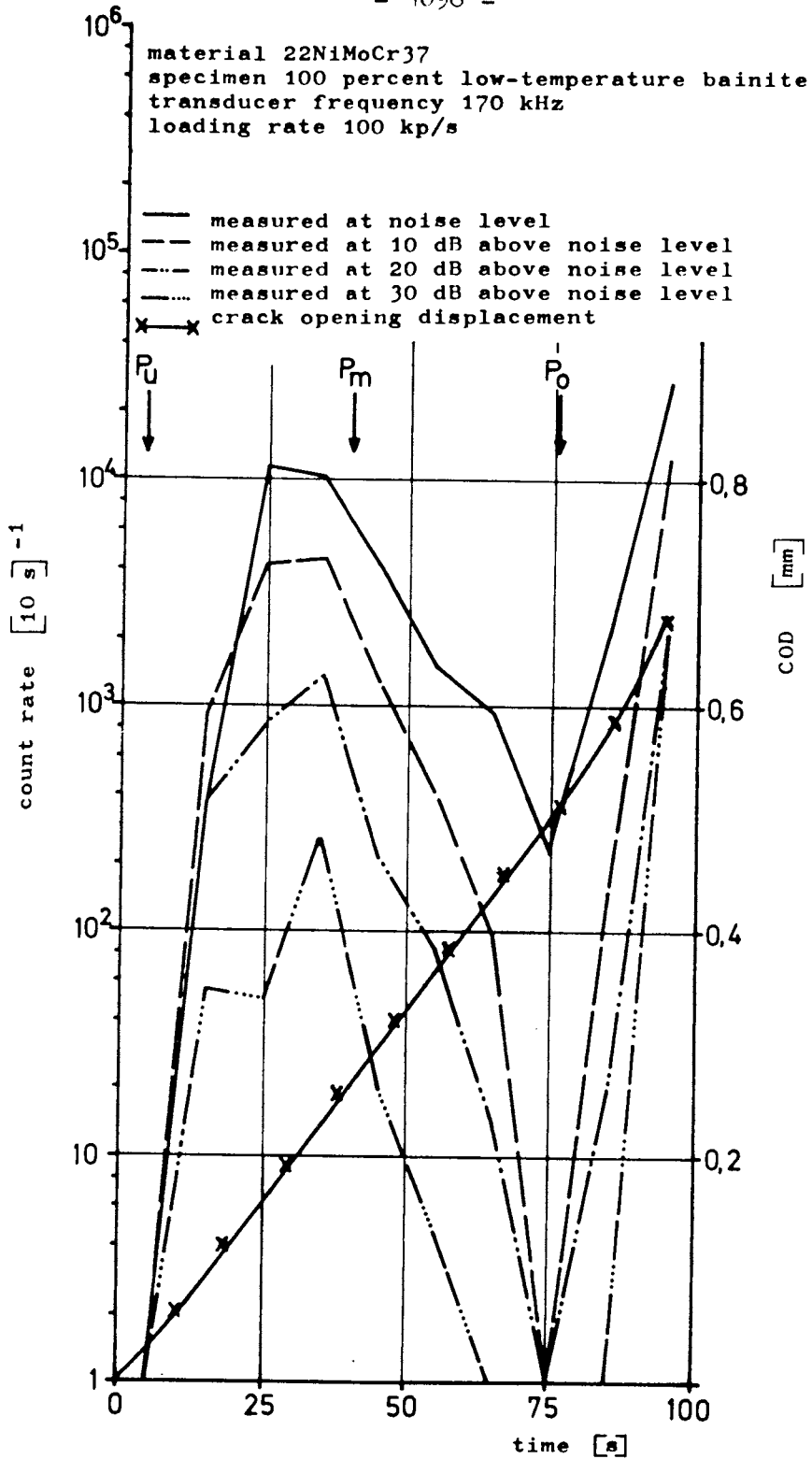


Fig. 6: AE and COD over time during tensile test of a pre-cracked specimen

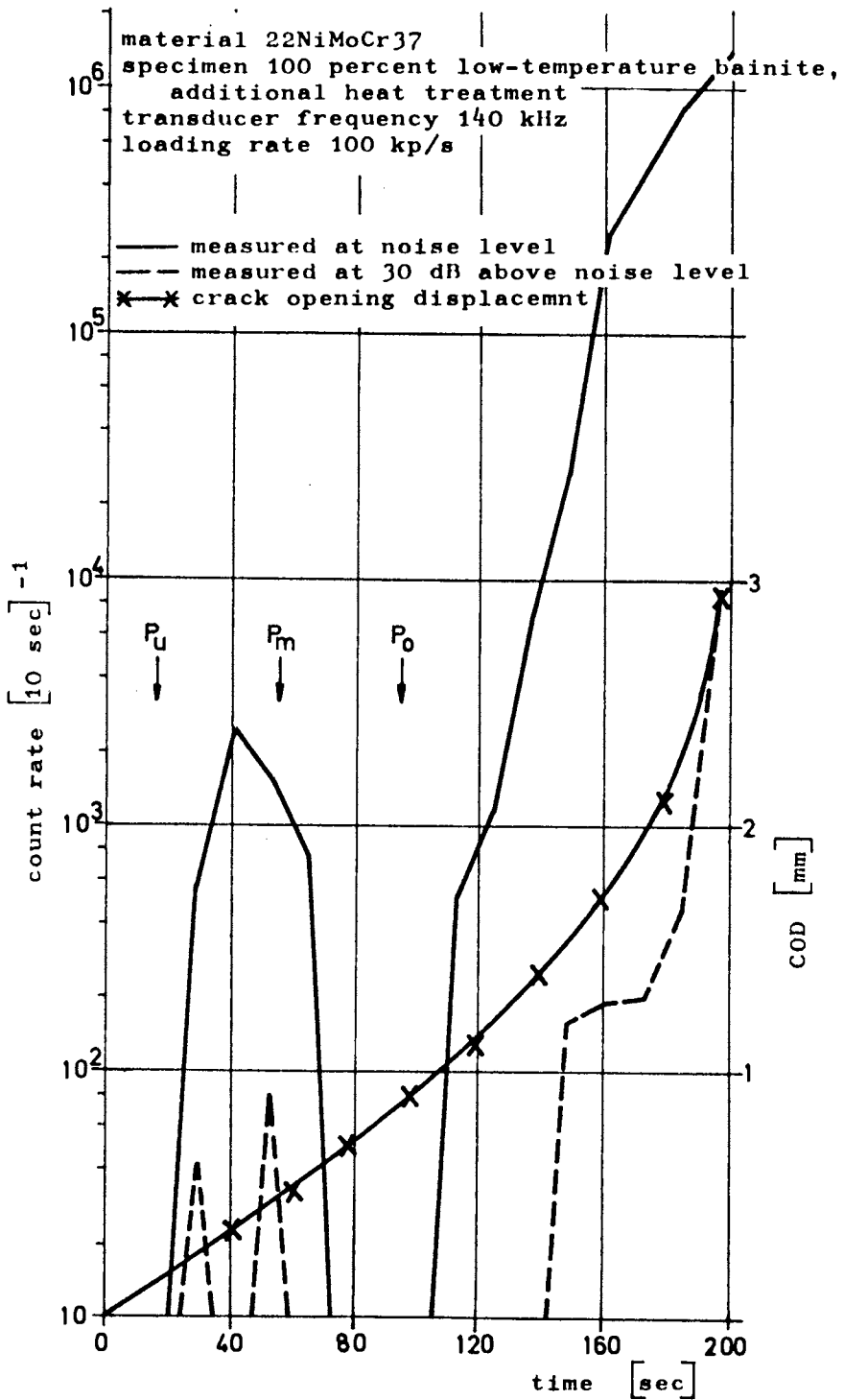


Fig. 7: AE and COD over time during tensile test of a pre-cracked specimen

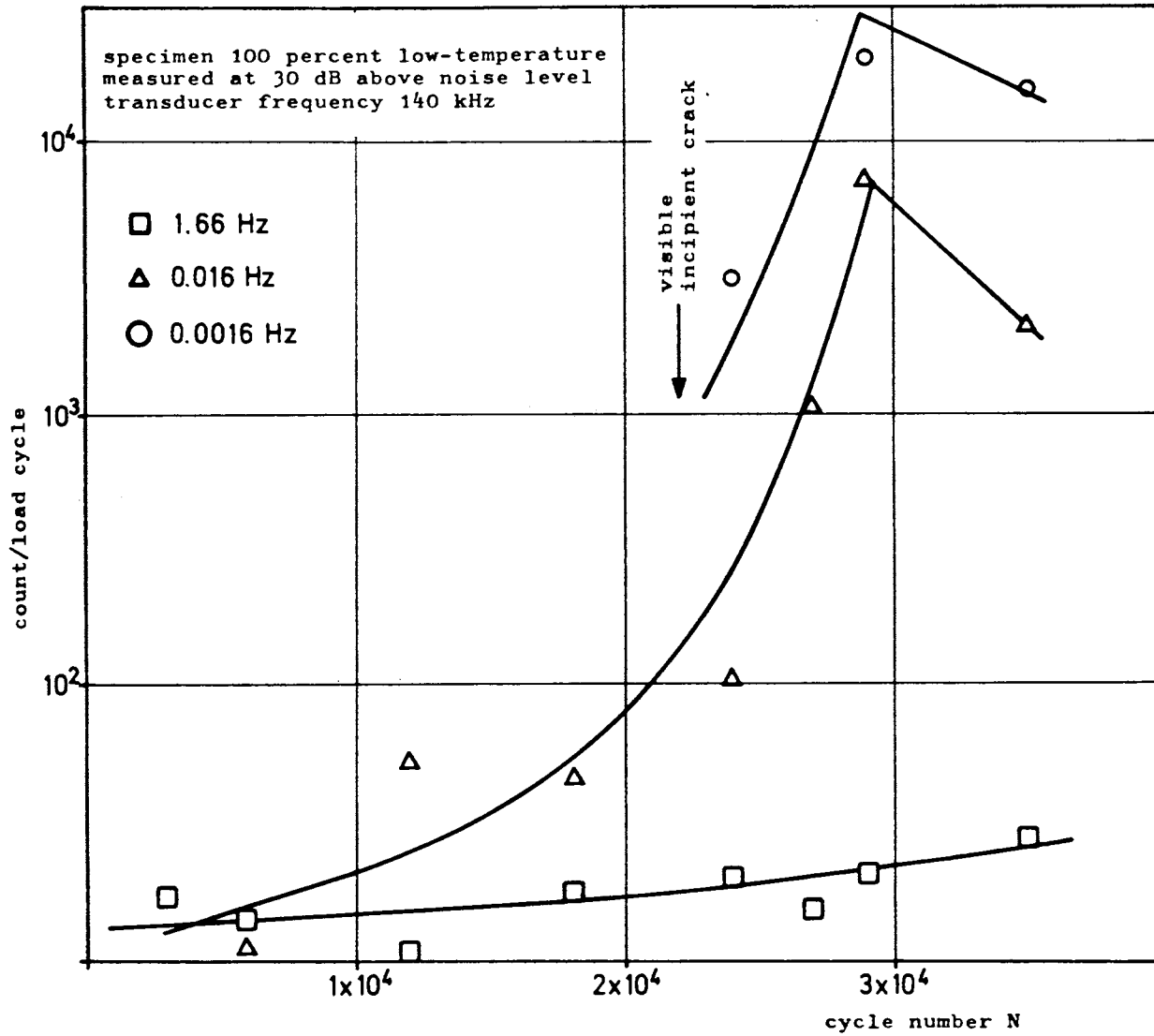


Fig. 8: AE at varying load frequency during fatigue

<u>Classification:</u> 12.3	
<u>Title 1 (Original Language):</u> Vorversuche zur Leckageüberwachung mit Hilfe der Schallemissionsanalyse (RS 193-II.3.2, Jahresbericht A 76)	COUNTRY: BRD
	SPONSOR: BMFT
	ORGANIZATION: Battelle-Institut e.V.
<u>Title 2 (English):</u> Preliminary Investigations On On-Line Leak Detection By Acoustic Emission	<u>Project Leader:</u> Dr. P. Jax
<u>Initiated (Date):</u> January 1, 1976	<u>Completed (Date):</u> January 31, 1977
<u>Status:</u> finished	<u>Last Updating (Date):</u> December 1976

1. General Aim

The general aim of the program is the development of a continuously operating surveillance system for leaks in the primary system of a nuclear power plant with the aid of acoustic emission. The system should be capable of constantly inspecting all components of the primary system of the plant during operation and locating the position of the leak with sufficient accuracy.

2. Particular Objectives

Under this research program it was to be investigated on a test loop whether acoustic emission is suited to detect leaks in the presence of background noise similar to that occurring under actual operating conditions. In particular, it was to be investigated what factors of influence on acoustic emission have to be taken into account (e.g. pressure, temperature, position), whether there is a relationship between leak size and leak type, and what possibilities of locating leaks exist in principle.

3. Research Program

Various defined leaks were to be produced on a test loop of KWU, Erlangen, and the acoustic emission involved was to be analyzed.

The research program covered the following investigations:

- 4 different leak types (crack-type leak, flange-type leak, valve leak, and borehole-type leak); pressure about 150 bar, temperature 300°C
- 2 leak sizes for the crack-type leak and the valve leak (about 35 and 180 kg/h) and systematic variation of the leak size for the flange-type and borehole-type leak; pressure about 150 bar, temperature 300°C
- Variation of several parameters affecting the background noise, such as the flow rate in the measuring section and the delivery of the circulating pump
- Systematic variation of temperature (320 to 90°C) or pressure (160 to 86 bar) for two leak types (borehole-type leak and flange-type leak) or one leak type (borehole-type leak)

On the basis of AE signals stored on magnetic tape, detailed analyses were to be made with respect to

- acoustic emission intensity and amplitude distribution,
- frequency spectrum,
- correlation between the various parameters.

4. Experimental Facilities

The test loop used was that of the experimental refuelling-machine facility made available by Kraftwerk Union AG (KWU), Erlangen. In this facility pressure and temperature can be controlled separately. The loop was connected with an austenitic tube (length about 10 m, outside diameter 267 mm) which constituted the actual measuring section. The devices for simulating the different leak types were mounted to this tube by means of nozzles and flanged connections.

5. Progress to Date

For the leakage experiments, transducers were mounted either directly or via flexible waveguides at five measuring points differently spaced from the leak. The signal level, represented by the rms value, was continuously recorded. For detailed analysis, the AE signals from all measuring points were stored on magnetic tape in analog form.

The research program outlined in Section 3 was carried out, and the planned analyses of the signals stored on magnetic tape have been largely completed.

Parallel to the leakage experiments, the attenuation of acoustic signals in the measuring section was determined at different frequencies. These measurements were made both with continuous noise sources (compressed air-stream, sinus signal fed into a transmitter) and with pulsive signals (square pulse fed into a transmitter).

6. Results

Acoustic emission was found to be a sensitive method for the detection of leaks. All leaks with leakage rates from about 20 kg/h to 250 kg/h were indicated in spite of the background noise of the experimental facility. As can be extrapolated from the results, the detection limit of AE, $N \approx 2$ to 6 kg/h, is markedly below the leakage rates examined. For borehole-type leak and flange-type leak a simple relationship was found to exist between the low-frequency (60 to 150 kHz) AE intensity R and the leakage rate L determined by means of an orifice-plate circuit ($R \sim L^b$, $b = 1.5$ or 1). Within a specific scattering range, it is thus possible already at the present state of research to make inferences as to the size of a leak.

Amplitude Distribution of the AE Signals

In all cases a continuous acoustic signal level is emitted, which differs basically from the burst signals observed in the case of crack growth and friction processes. At a given frequency, the sound pressure amplitude A varies around an rms value U according to a uniform statistical distribution function \mathcal{G} . Thus, we have

$$\mathcal{G} = 2/U \cdot \left(\frac{A}{U}\right) \cdot \exp\left(-\left(\frac{A}{U}\right)^2\right) \quad (1)$$

The physically reasonable measurement parameter is the rms value of amplitude U that can be measured by means of a voltmeter and from which, at a given frequency, all other parameters of acoustic emission intensity can be derived by means of equation (1). In general, it is sufficient for characterizing the leakage noise to measure U as a function of frequency and time.

Frequency Spectrum

Analyses of the frequency spectra of acoustic emission confirmed that marked changes in the ratio of the rms values of the investigated frequency ranges 300 to 900 kHz and 60 to 150 kHz may occur, depending on leak type, leak size and temperature. In a first approximation it was found for the borehole-type leak that at constant pressure and temperature conditions the higher frequency components become more intensive with decreasing diameter of the borehole (decreasing leakage rate). A similar relationship of the frequency spectrum with the leakage rate was not detected for the flange-type leak. There is evidence that the frequency spectrum depends in particular on the temperature of the medium circulated in the test loop. In addition, it was found that a minor variation of the frequency spectrum at the beginning of a leakage experiment correlates with a small increase in temperature at the site of leakage (heating by the medium).

In summary it may be stated that, apart from the results obtained for the borehole-type leak, it is not possible as yet to give more details about the relationships between frequency spectrum and the experimental parameters.

A comparison of the frequency spectra recorded at various measuring points showed a systematic relationship to the distance of the transducer from the site of leakage. With increasing distance the higher-frequency components (400 to 900 kHz) of acoustic emission are more strongly attenuated, i.e. there is a marked shift of the measured frequency spectrum in favor of the lower-frequency components. It appears feasible to achieve a more exact location of leaks with the aid of this effect, which is due to the attenuation of the signals during their propagation.

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7. Next Steps

Preparation of the final report.

8. Relation with Other Projects

Parallel to the AE measurements in the ultrasonic frequency range (50 to 800 kHz) described in the present report, the Allianz-Zentrum für Technik, München, measured the frequency components of leakage noise below 30 kHz (research project RS 97).

It was to be clarified in this way whether the audible frequency range or the ultrasonic frequency range is better suited for leak detection.

9. References

10. Degree of Availability

<u>Classification:</u> 12.3				
<u>Title 1 (Original Language):</u> Verbesserte phänomenologische Beschreibung von Schall-emissionssignalen und ihre Analyse auf eine bessere Fehlerbewertung. (RS 196 - II.3.2, Jahresbericht A 76)	<u>COUNTRY:</u> BRD			
	<u>SPONSOR:</u> BMFT			
	<u>ORGANIZATION:</u> Fraunhofer-Ges., Izfp, IfaM, IFKM			
<u>Title 2 (English):</u> Better phenomenological description of acoustic emission signals and their analysis with respect to a better evaluation of defects.	<u>Project Leader:</u> Dr. J. Lottermoser Dipl.-Phys. T. Fischer Dr. B. Voss			
	<table border="0" style="width: 100%;"> <tr> <td style="width: 50%;"><u>Initiated (Date):</u> 1.2.1976</td> <td style="width: 50%;"><u>Completed (Date):</u> 31.1.1979</td> </tr> <tr> <td><u>Status:</u> continuing</td> <td><u>Last Updating (Date):</u> December 1976</td> </tr> </table>	<u>Initiated (Date):</u> 1.2.1976	<u>Completed (Date):</u> 31.1.1979	<u>Status:</u> continuing
<u>Initiated (Date):</u> 1.2.1976	<u>Completed (Date):</u> 31.1.1979			
<u>Status:</u> continuing	<u>Last Updating (Date):</u> December 1976			

1. General Aim

The general aim of this project is

- the determination of the correlations between acoustic emission (AE) parameters and defects (stress intensity factor) and optimization of mathematical models for the description of these correlations,
- characterization of AE sources (cracks) in base metals and welded joints,
- determination of the wave modes which are responsible for the propagation of AE signals in thick-walled components,
- correction of AE parameters for signal disturbances on their path in a specimen or component and the transducer,
- better analysis of AE signals including determination of energy, amplitude, spectra and the statistical distributions of these parameters.

2. Particular Objectives and Research Program

This work is done especially for applications on the reactor pressure vessels. The herein used materials are 22 NiMoCr 3.7, 20 MnMoNi 55 and austenitic steels, mainly X6 CrNi 18.11. Base metals as well as welded joints are to be investigated.

The work shall be done in three steps:

- laboratory tests on modified CT and SEN specimens,
- laboratory tests on specimens with production defects,
- application of the developed methods and models to large scale tests.

The research program is including the simulation of AE signals and the characterization of acoustic emission transducers by their spectra and directivity pattern.

3. Experimental Facilities

- hydraulic testing machines modified for low noise,
- special grips for SEN and CT specimens without frictioning parts,
- analog and digital devices for signal analysis in the frequency range from 0.1 to 2 MHz,
- computer for signal analysis (especially frequency analysis) and determination of statistical distributions.

4. Progress todate

- 4.1. Modification of testing machines for low noise. Construction of special grips for CT specimens.
- 4.2. Frabrication of specimens (including welding) and metallurgical analysis.
- 4.3. Measurement of the directivity pattern of AE transducers.
- 4.4. AE measurements on CT specimens (50 mm)

- 4.5. Optimization of fracture mechanic models to the geometries and materials used in these experiments.
- 4.6. Measurements for determining the propagation of ultrasonic pulses in 100 mm thick plates and smaller specimens including welds.
- 4.7. Simulation of AE pulses with a laser.

5. Essential Results

- 5.1. In fatigue of CT specimens with large $\Delta K \approx 1500 \text{ N/mm}^{3/2}$ ($K =$ stress intensity) we have mainly acoustic emission from friction of the crack flanks. For smaller values of $\Delta K \approx 700 \text{ N/mm}^{3/2}$ there is no more AE of friction.
- 5.2. The micro-structure of the ES-welded joints was mainly heterogeneous due to the high heat input involved during ES-welding (fig. 1). The base material is well tempered martensite. In the outer regions, the heat affected zone contains pearlite and ferrite in a lamellar constellation. Closer to the weld seam, the propagation of pearlite and upper bainite increases. The region of the HAZ next to the seam has coarse grain. It consists exclusively of upper bainite with rather large stripes of cementite inside the ferrite grains. Also the weld zone itself is purely upper bainite, but this has a crystallized dendritic structure. From measurements of hardness it can be expected that the region next to the base material is the weakest. The coarse grain region next to the weld is sensitive to cleavage fracture.
- 5.3. In a tension test on a CT specimen we have generally two maxima of AE energy rate:
 - a first peak is due to friction of crack flanks and depending on their roughness (fig. 2 for a welded joint with friction and fig. 3 without friction); the AE signals can be characterized by the distribution of energy per event given in picture 4a (specimen of base material).

- a second peak is due to plastic deformation, micro-crack formation and macroscopic crack growth. These AE signals can be characterized by the distribution of energy per event given in picture 4b (specimen of base material).

By means of frequency and pulse area analysis, different groups of signals were detected. According to the time of their appearance they were correlated to plastic deformation, microvoid formation, macroscopic crack growth and friction (fig. 5).

Samples having the fatigue crack in different regions of the weld seam and in the HAZ all showed cleavage fracture at room temperature. At 60 °C, shear lips with a thickness of a few mm appeared.

- 5.4. The volume of the plastic zone in front of the crack tip is given by an expression

$$V_{pl} = a_4 K^4 + a_6 K^6$$

This expression is calculated by taking account of the plane strain and plane stress conditions in the materials. The coefficients a_4 and a_6 are dependent on the thickness of the CT specimen and the yield point of the material. For a specimen of thickness 50 mm we shall always see a dependence of V_{pl} on K with 4th power but for smaller specimens (e.g. thickness = 7 mm) and lower yield point there should be an influence of both parts of the above expression.

- 5.5. For the pressure vessel we have to assume that the major AE sources are near to the inner surface of the vessel wall because the stresses are highest at the inner side. In general the acoustic emission signals are detected at the outer surface of the vessel wall. For this geometry we have done experiments which are described in picture 6. The results are to be seen in picture 7. We see that the amplitude of the surface wave is dependent on the roughness of the outer surface of the vessel wall. Large-scale experiments are mainly done in the frequency range of 100 - 300 kHz, i.e. with wavelengths of cm. So we cannot assume for reactor pressure vessels that the surface wave is important for the propagation of acoustic emission signals.

- 5.6. we have simulated an acoustic emission source with a laser heating up the surface of a steel or aluminium plate. The generated acoustic

pulse has a short duration (< 500 ns) and the directivity pattern given in picture 8.

6. Next Steps

- Continuing AE measurements on CT and SEN specimens with signal analysis,
- determination of the correlations between AE parameters and stress intensity factor and development of models,
- measurement of the directivity pattern of natural AE sources (cracks).

7. Relation With Other Projects

These works are done on the base of the results of the projects RS 31 to RS 31/3 and in cooperation with Battelle-Institut at Frankfurt (RS 191).

8. References

E. Waschkiel:
Schallausbreitung bei der Schallemissionprüfung.
(1977)

9. Degree of Availability of the Reports

The report cited above is available from IzfP, Saarbrücken.

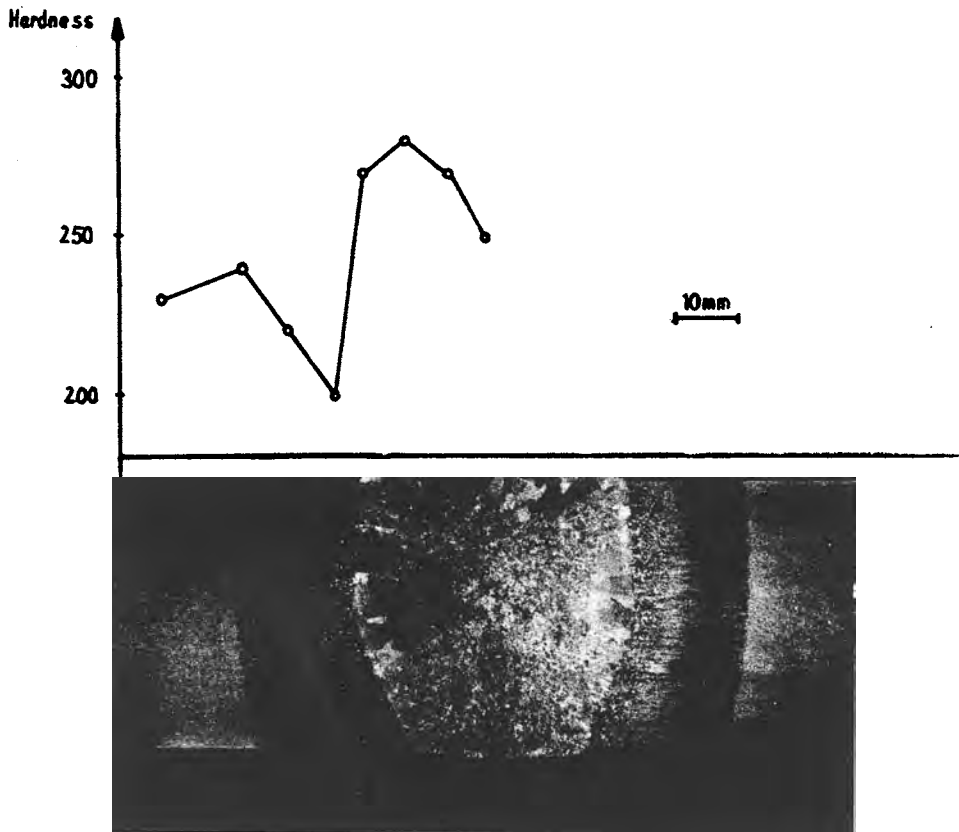


Fig. 1: photo-micrograph and hardness of a welded joint.

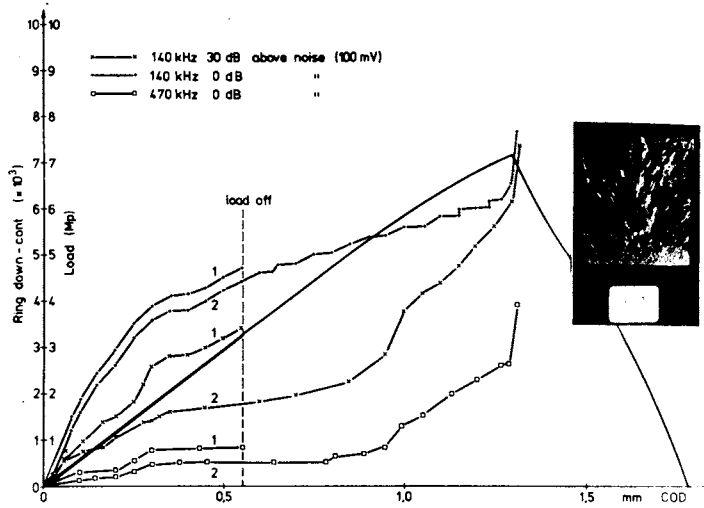


Fig. 2

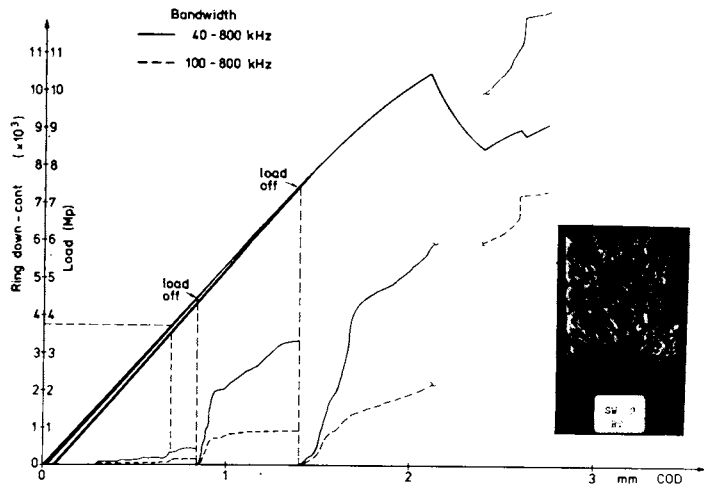


Fig. 3

Fig. 2: Ring-down-count and load against crack-opening for CT specimen (welded joint); rough surface of the crack.

Fig. 3: Ring-down-count and load against crack-opening for CT specimen (welded joint); smooth surface of the crack

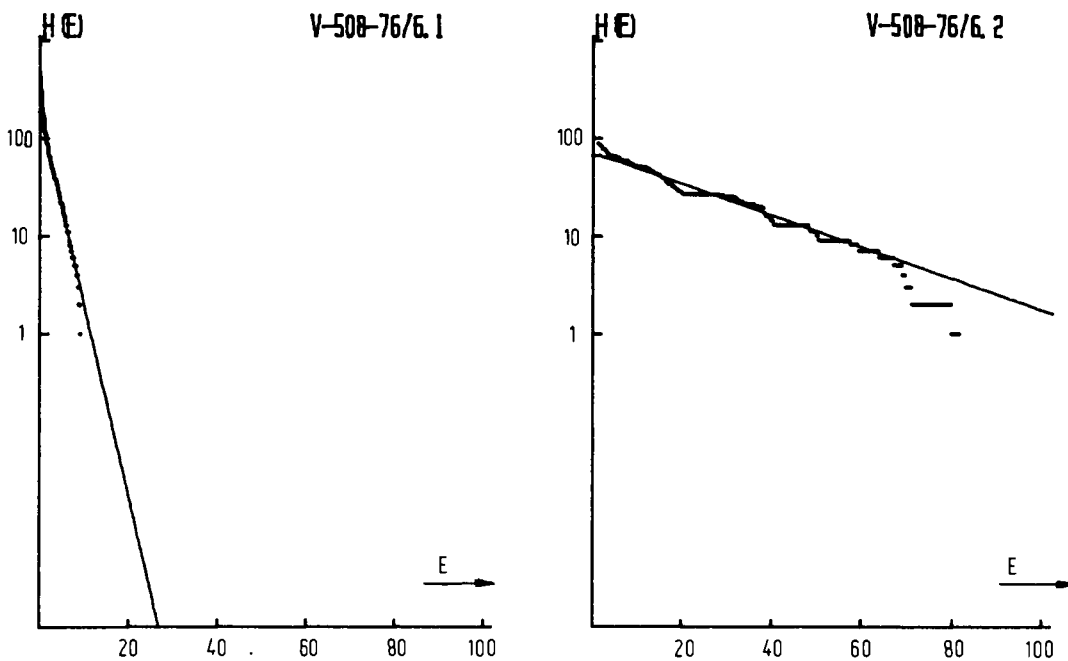


Fig. 4: Number H of AE events with energies equal to or greater than E (on arbitrary scale) in a tension test on a CT-specimen (base material 22 NiMoCr 3.7)
 a) friction of the crack-surface
 b) micro-cracking and macroscopic crack-growth.

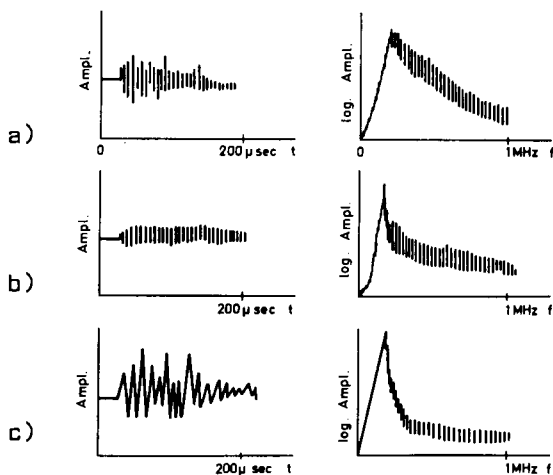
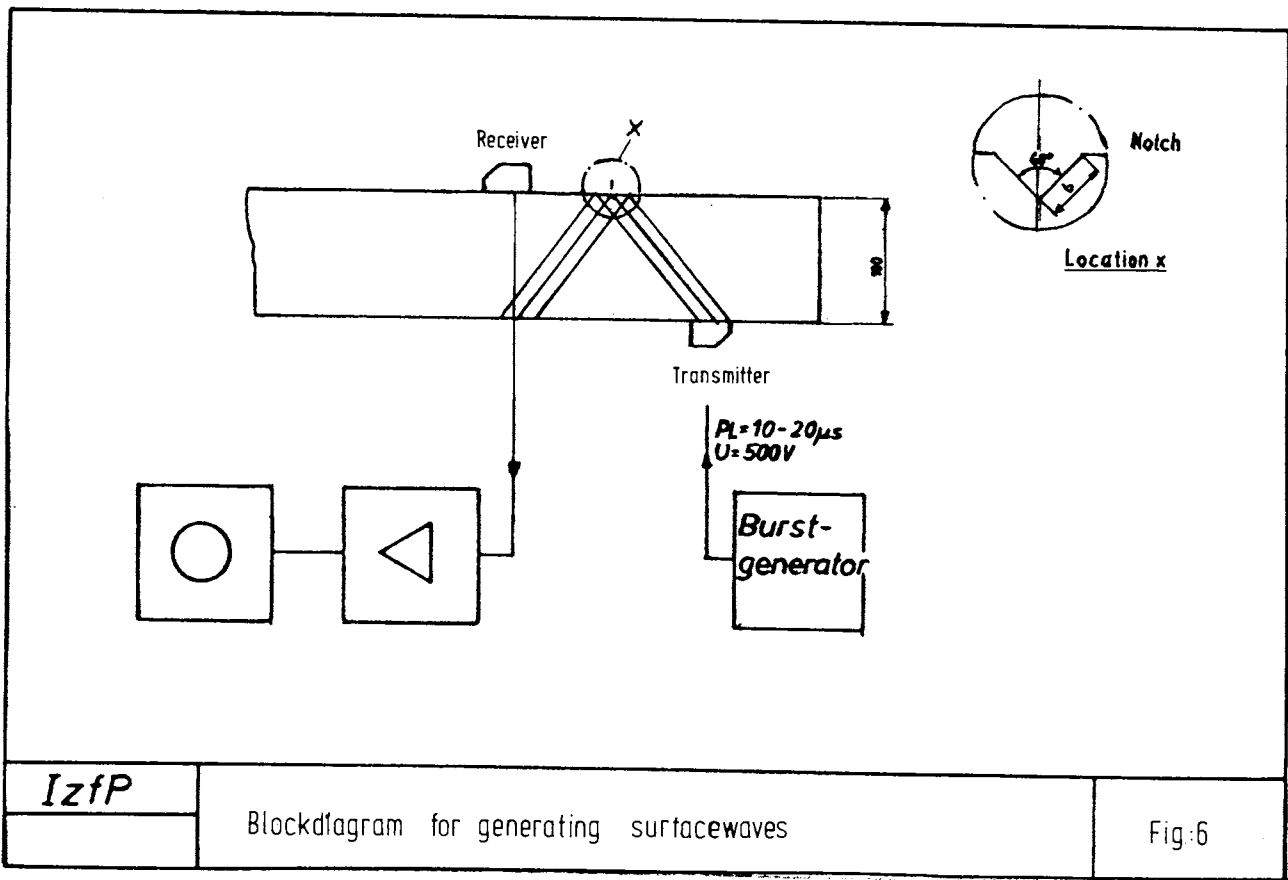
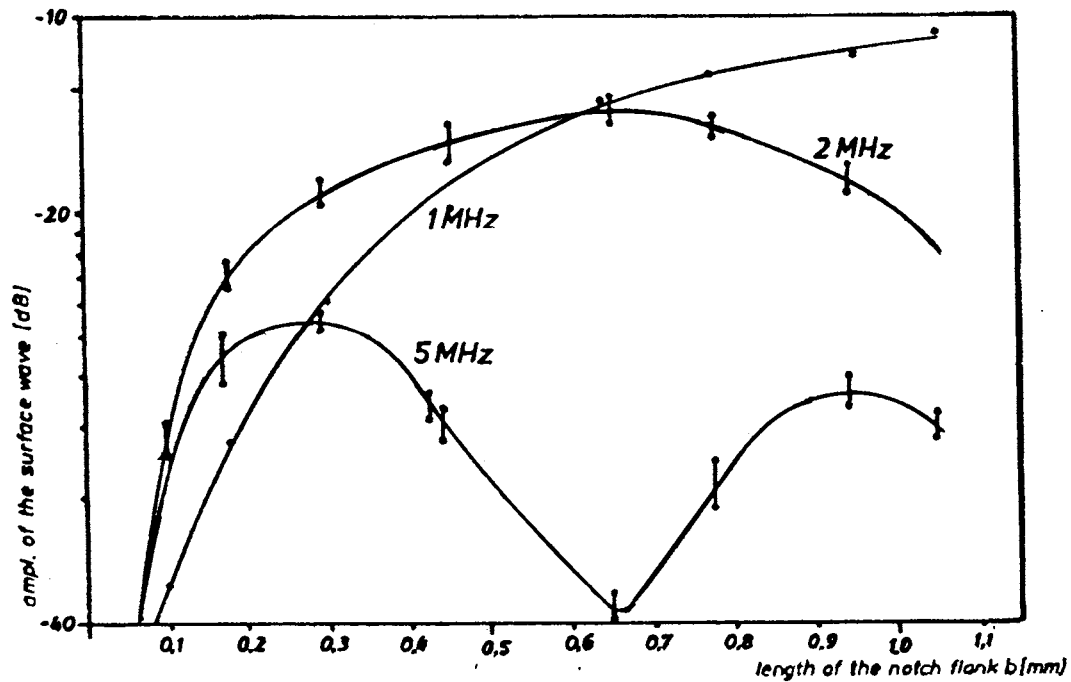


Fig. 5: typical AE signals and spectra:
 a) friction of the crack surface
 b) micro-cracking
 c) macroscopic crack growth





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Relative ampl. of the generated surface wave
(in reference to the incident transverse beam)
as function of the length of the notch flank.

Fig.: 7

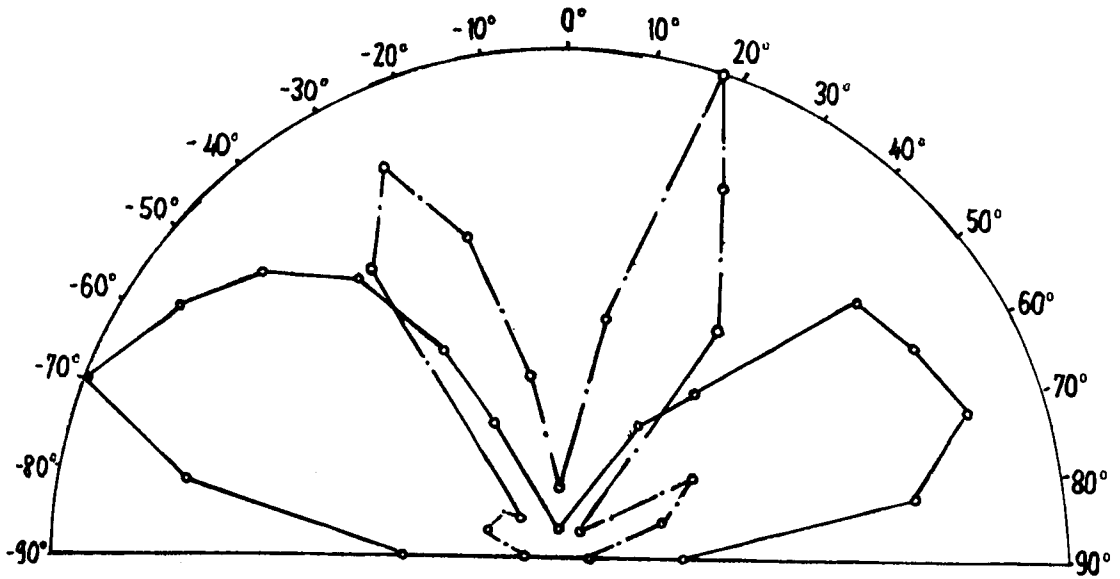


Fig. 8: Directivity pattern of a laser-generated acoustic source

— longitudinal wave
-·- transverse wave

<u>Classification:</u> 12.3	
<u>Title 1 (Original Language):</u> Zerstörungsfreie Prüfung des Gefügezustandes mittels Ultraschallrückstreuung (RS 102-16/1 - II.3.2., Jahresbericht A 76)	COUNTRY: BRD
	SPONSOR: BMFT
	ORGANIZATION: Fraunhofer-Ges. München, IzfP
<u>Title 2 (English):</u> Non-destructive structure evaluation by means of scattered ultrasound	Project Leader: Dr. G. Deuster Dr. K. Goebbels
<u>Initiated (Date):</u> 01.05.1973	<u>Completed (Date):</u> 31.12.1977
<u>Status:</u> Continuing	<u>Last Updating (Date):</u> December 1976

1. General Aim

Among other things the material constants of steels are structure-dependent. Therefore the determination of the structure of steels is an essential task and has to cover the whole volume, especially for big components like reactor pressure vessels. Under this category the structure of welds is to regard, too, and the evaluation has to be done in a ndt way. Ndt methods for these requirements are not yet technical standard, but in an advanced stage of development.

Another important point of interest is the detection of defects - both real and latent - as flaws in the presence of a high coherent noise level, the accumulation of impurities in certain material areas and the proof of inner tensions.

2. Particular Objective

Measurements of scattered ultrasound can be used for qualitative and quantitative determination of metal structures (grain size, impurity distribution). Compared to attenuation measurements, they present e.g. the following advantages:

- no need of plane-parallel, polished test specimens,
- evaluation of the structure inside the material,
- quantitative separation of absorption and scattering coefficient from attenuation coefficient,
- quantitative grain-size determination, independent of other methods.

The included impurities, generally differing in size from the grains, become detectable when measuring scattered ultrasound at appropriate frequencies.

Inner tensions cause changes of the elastic constants of the material, which lead to changes of the velocity of elastic waves. For that reason, velocity measurements are done using the pulse echo overlap method.

The reduction of the coherent noise level e.g. at the US examination of coarse-grained materials and clad reactor pressure vessels, is possible when using signal averaging methods. In this respect several averaging methods (variation of probe position, excitation frequency of beam angle) are under consideration.

3. Research Program

In 1976, there were the following points of special interest:

- a) test of the new device for scattering measurements in an application test with about 200 steel specimens;
- b) determination of the impurities in several reactor pressure vessel specimens;
- c) measurements on austenitic materials and welds for better understanding of the steel structures and for obtaining a better signal-to-noise level in ultrasound examinations;
- d) measurements with several averaging methods applying the basic laws of the US scattering and the results of the measurements mentioned under c).

4. Experimental Facilities, Computer Codes

- For very accurate measurements of ultrasound velocities and attenuation an experimental device (MATEC) is used.
- Scattering measurements are made with two apparatus:
 - at first a device built by Koppelman¹⁾, recording the analog scattering signal on a XY recorder.
 - A second apparatus was built in the institute, marked especially by an analog-digital converter (100 MHz sample rate) and the on-line evaluation of the scattering measurement in a computer.
- For the averaging measurements the US pulse generating system differs from that one of the second apparatus (see above); the signal processing system is the same.

5. Progress to Date

The application test has been performed completely in 1976; the qualitative evaluation of the scattering curves allows to determine whether the structure is homogeneous or inhomogeneous; from measurements on austenitic materials there could be developed several averaging methods to improve the signal-to-noise ratio.

6. Results

- In the application test about 200 steel specimens are examined. The scattering curves were evaluated and the scattering coefficient and the grain size were determined from the attenuation coefficient. The error bonds are plus minus half an ASTM grain size class.
- The determination of impurity areas within the specimen was done by evaluating the scattering curves only in a qualitative manner. If there are any inclusions the sound reflexions outside this area are superposed to the scattering curve.
- To improve the signal-to-noise ratio when testing coarse-grained materials, especially austenites and its welded joints several averaging methods were applied. By varying the probe position, the excitation frequency or the beam angle, the interference pattern with respect to its maximum-minimum distribution over the

time axis is varied, too. If there is any signal from a defect it will stay with nearly the same amplitude at the same place in the pattern. By the superposition of several interference structures, the useful signal grows more than the noise whereby the signal-to-noise ratio is improved.

When applying the basic laws of the US scattering theory, the averaging curves can be interpreted in a quantitative manner; the flaw size of the defects can be determined by using the DGS method after Krautkrämer.

7. Next Steps

Beside a further examination of the several averaging methods there will be a main effort in improving the local resolving power when examining welded joints by means of focusing probes. In 1977 the project will be finished.

8. Relation with Other Projects

No essential relations with other projects are present, only the work concerning austenitic steels in RS 102-16/1 and RS 143 are tuned together.

9. References

K. Goebbels: Materialprüfung 18 (1976) 3, 86-88

S. Kraus, K. Goebbels:

Signalmittlungsverfahren zur Unterdrückung des kohärenten Untergrundes bei der Ultraschallfehlerprüfung grobkörniger Werkstoffe.

Izfp-Bericht 760717-TW

<u>Classification: 12.3</u>	
<u>Title 1 (Original Language):</u> Anpassung und Weiterentwicklung von elektrischen, elektromagnetischen und magnetischen Prüfverfahren für den Einsatz an Reaktoren (RS 102-18 - II.3.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: Fraunhofer-Ges., IzfP
<u>Title 2 (English):</u> Adaption and continuation of the development of electrical, electromagnetic and magnetic test methods for the use at reactors	<u>Project Leader:</u> Dipl.-Phys. R. Becker Dr. W. Mohr Dipl.-Phys. G. Dobmann
<u>Initiated (Date):</u> 1.8.1973 <u>Status:</u> Part 1 finished, parts 2,3,4 continuing	<u>Completed (Date):</u> 30.9.1976, Part 2,3,4: 30.9.1977 <u>Last Updating (Date):</u> 31.12.1976

General aim

In the field of the electrical, magnetic and electromagnetic test methods the project has the general aim

- to transfer the status of knowledge into the concepts of techniques and devices,
- to expand the foundations in mathematics and physics as much as necessary,
- to establish rules and models for interpretation of measurement results,
- to adjust the techniques to the specific problems in reactor testing.

Based on the research results, special developments orientated towards application are to follow as individual projects.

1. Eddy Current Test Method

1.1. Particular Objectives and Research Program

We are working at two kinds of application:

- a) testing of installed heat exchanger tubes,
- b) testing of the cladding, the basic material, the weld and austenitic/ferritic junctions of the reactor pressure vessel for location of cracks near the surface.

In both cases a great number of parameters give measuring effects. The signals of defects which generally are the relevant parameters can be superimposed by signals of other parameters, so that the presence of the defects cannot be recognized or the indication of the defect depth can be falsified. Therefore the signals from the disturbing parameters must be suppressed and only the signals from the defects must be given out with the indication of the defect depth.

To solve this task the multifrequency method is to apply where is worked with several frequencies simultaneously. This method is to develop further with regard to the practical application and to test in several test situations which are simulated by models.

1.2. Experimental Facilities and Computer Codes

- a) Evaluation and application of computer programs to determine the impedance of the test coil as a function of the electrical, magnetic and geometrical properties of the specimen, as well as a function of the dimensions of the coil; the specimen can be a tube, a rod or a slab consisting of several layers.
- b) Construction of a laboratory test device which works simultaneously with up to 4 frequencies in the range of 10 - 500 kHz.
- c) Application of this device for the above mentioned test situations and optimization of the evaluation method to obtain the read-out value.

1.3. Progress Today

- a) Improvement of the test device (in 1.2.b) concerning stability and handling.

b) Continuation of the experiments with tubes, claddings and welded joints.

c) Completion of the computer program in 1.2.a).

1.4. Results

The project has been finished. In laboratory tests it could be demonstrated that the multifrequency method is adequate to solve the difficult problems in testing heat exchangers and other components of nuclear plants. A test device exists, which is the starting point for the construction of a prototype suited for the "in-situ" application. Just so there is a computer code which can be used to optimize the test equipment (frequency, design of the coils, etc.) for several test situations.

1.5. Relations With Other Projects

Concerning the testing of claddings and welded joints there is a relation to project RS 89.

1.6. References

Final report, in preparation.

2. Electrical Resistance Probe Method

2.2. Particular Objective

The method is used for crack-depth measurements of surface cracks in metallic materials. By contacting the body with potential probes an electrical flow field is impressed. The associated potential field, measured on the surface is a function

- of the electrical conductivity,
- of the distance of the potential probes and
- of the surface geometry.

Every surface crack disturbs this geometry and therefore the potential distribution. The normalized voltage U/U_0 measured above the crack is

only a function of the depth of the crack while U_0 is the voltage in the undisturbed case. The theoretical formulation of the problem leads to a Neumann boundary value problem.

2.3. Research Program

2.3.1. Numerical solution of two-dimensional electrical potential field problems.

2.3.2. Experimental tests of the numerically computed crack-depth functions for real reactor components.

2.4. Experimental Facilities, Computer Codes

2.4.1. Implementation of a finite-element FORTRAN IV program for numerical solution of Neumann boundary value problems.

2.5. Progress Today

The finite element program was improved, it allows the solution of potential problems for two-dimensional geometries with arbitrary boundaries and variable sources of the field.

2.6. Results

Application of the program on surface crack geometry in screw bolts of reactor pressure vessel yields to good agreements with model measurements. For CT-standard test specimens of fracture mechanics the form and manner of the electrical sources could be optimized for studying crack growth measurements.

2.7. Next Steps

The treatment of surface cracks in claddings and welded joints at a nozzle in reactor pressure vessel.
Development of a finite element program for three-dimensional geometries.

2.8. Relation with Other Projects: none

2.9. References

Dobmann, G.:

Potentialsondenverfahren. Berechnung und Messung zweidimensionaler Potentialverteilung bei homogener Anregung.

Izfp-Bericht 750102-TW

Zwischenbericht Potentialsondenverfahren. Numerische und experimentelle Untersuchungen zum Teildurchströmungsverfahren.

Izfp-Bericht 760214-TW

2.10. Degree of Availability of the Reports

Materialprüfung 18 (1976) 9, pp. 342-344

3. Magnetic Leakage Flux Method

3.1. Particular Objective

For ferromagnetic materials which are flowed through by means of direct magnetic field produced by a yoke or by means of a magnetic field of a direct current, the magnetic leakage flux measured with Hall elements over and near a surface crack is a function of the permeability and of the geometry of the crack. The magnetography method allows documentation and signal processing of the leakage flux signals.

3.2. Research Program

3.2.1. The development of a magnetography signal measure and interpretation unit.

3.2.2. Theoretical examinations about the influence of surface crack parameters as crack-length, -depth and width and of the permeability by solving numerically a Fredholm's integral equation.

3.3. Experimental Facilities, Computer Codes

3.3.1. Fundamental experiments with magnetography are made.

3.3.2. A FORTRAN IV program for iterative solution of integral equation is implemented.

3.4. Progress Today

First numerical results.

3.5. Results

The first iterative solution shows the influence of crack width in the distance of the local extremas in the vertical magnetic flux component.

3.6. Next Steps

The plan is the non-destructive testing of welded joints in reactor containments with magnetography-leakage-flux method. For this application a signal processor must be built.

3.7. Relations With Other Projects: none

3.8. References

Höller, P.:

Elektrische und magnetische Verfahren zur zerstörungsfreien Werkstoffprüfung.

In "Neuzeitliche Verfahren der Werkstoffprüfung", Verlag Stahleisen mbH, Düsseldorf 1973, Seite 139

4. Guided Ultrasonic Waves Electrodynamically Excited

4.1. Particular Objectives and Research Program

A technique will be elaborated which allows to inspect heat exchanger tubes for longitudinal and transverse cracks by guided waves. The guided waves will be excited and received contactlessly by electrodynamic transducers.

The contactless electrodynamic excitation shall also be used for free ultrasonic waves in order to develop an angle probe for austenitic specimens.

4.2. Experimental Facilities

The equipment for tube testing has been in essential the same as in 1975 with the difference that the transducer coils are shortened to improve the local resolution of flaw echoes. For the excitation of free ultrasonic waves a pulse magnetization technique was developed which allows the generation of a high short-time magnetization power in electromagnets of small size.

4.3. Project Status

4.3.1. Progress Today

Since the dispersion of guided waves generated in thin-walled tubes is very small in the frequency range of interest, short transducer coils with large spatial bandwidth could be developed in order to improve the axial local resolution of flaw echoes.

For the contactless electrodynamic generation of free ultrasonic waves meander-transducers with their fitting electromagnets were developed to be used as angle probes for ferritic specimens, the angle of incidence of the ultrasonic waves being adjustable by electronic frequency change. To suppress the side lobes in the radiation patterns some tapering techniques were tested.

4.3.2. Essential Results

Concerning tube waves the axial resolution of flaw echoes could be trebled. Concerning the contactless electrodynamic generation of free ultrasonic waves a dynamic range of 60 dB could be reached at an angle of incidence of 45° . By frequency change it was possible to vary the angle of incidence from about 25° to 50° . A Dolph-Chebyshev-tapering involved a suppression of the side lobes of about 15 to 25 dB in the radiation patterns.

4.4. Next Steps

The development of a method for tube testing by guided waves is carried

on in another project. The dynamic range of free ultrasonic waves generated by contactless electrodynamic transducers shall be improved in order to be available for austenitic specimens, too.

<u>Classification: 12.3</u>	
<u>Title 1 (Original Language):</u> Automatisierung und EDV der US-Impulsechoprüfung (RS 102-17 - II.3.2, Jahresbericht A 76)	<u>COUNTRY:</u> BRD
	<u>SPONSOR:</u> BMFT
	<u>ORGANIZATION:</u> Fraunhofer-Gesellschaft, Izfp
<u>Title 2 (English):</u> Automation of Pulse-Echo-Testing and Electronic Data Processing of the Results	<u>Project Leader:</u> O.A. Barbian
<u>Initiated (Date):</u> 15.10.1974 <u>Status:</u> Continuing	<u>Completed (Date):</u> 31.12.1977 <u>Last Updating (Date):</u> December 31, 1976

1. General Aim

During the production of reactor pressure vessels both, the basic materials and the single components undergo a nondestructive and fully volumetric test. After the completion of the reactor a basic test is made, and then, at certain intervals, fully automatic in-service inspections take place under more complicated conditions. This is to locate and evaluate defects. Evaluation is effected in a computer after the reconstruction of a threedimensional flaw picture dependent upon the different testing methods. These flaw pictures are compared with the specifications and standards.

2. Particular Objectives

The aim of the investigations is the reconstruction of flaw pictures true as to locus and geometry. Only such reconstructions allow the application of fracture mechanical criteria or empirical rules for the evaluation of nondestructively detected flaws.

3. Research Program

The following steps have to be carried out:

- utilization of information included in echo-dynamics
 (local curve of transit time, local curve of amplitudes),

- utilization of information included in phase and amplitude of the signal,
- adjustment of reconstruction models to natural defects must be better than the former one (for example finite cylinder, infinite cylinder, sphere, ellipse, circle),
- consideration of subliminal connexions of defects by means of neighbouring relations,
- considerations, respectively suppression of transfer fluctuations, coupling failures, form echoes, and redisturbances;
- the results are to be made available in a compressed manner and for all testing techniques simultaneously, i.e. as threedimensional flaw picture, which allows sections in any direction.

4. Experimental Facilities, Computer Codes

For the realization of this concept it is necessary to work with probes, which have a high sensibility and dynamic range. Moreover, the test systems have to work with different ultrasonic test methods as, i.e., tandem, pulse-echo, V-transmission, focus-tandem, focus-pulse-echo, etc. An ultrasonic hardware has to be available, which allows the transfer of the necessary signals (for example, echo-amplitude + echo transit time, resp. the envelope of an echo-signal and, if necessary, the complete high-frequency signal) to the computer. This work is done by IzfP as well as the experimental data recording at artificial and natural defects. The software, necessary for data processing has to be developed by the associated institute IPW in Freiburg.

5./6. Progress to Date and Results

For the reconstruction of flaws in the far-field of piston diaphragms mathematical models have been developed which use the signal locus curves for evaluation. These models are able to consider simultaneously the indications of different ultrasonic test methods, as for example tandem and pulse-echo. By that means the distinction between plane and voluminous defects is possible. For plane defects the mathematical model has been expanded already so much that defect size and inclination can be

reconstructed from the amplitude locus curves.

The IPW in Freiburg has started the development of the software for the mentioned evaluation and reconstruction. The following program parts are completed already:

- **determination** of the local curves of amplitudes and transit time,
- compensation of the amplitude locus curves by parabolas,
- reconstruction of defect locus in B-picture.

Concerning the ultrasonic testing electronic (hardware), the logarithmic amplifier, the multiplexer, the filters and the transmitter are completed and tested both, with bursts and with original ultrasonic signals. Moreover, the time synthesizer and the peak detector are finished and tested with bursts.

7. Next Steps

The ultrasonic hardware components will be completed and built up. Hardware and software will be tested with real and natural signals and - if necessary - improved.

8. Relations with Other Projects

This programme is partly based on the results of the reactor safety research RS 27/1 and RS 27/2 and will be carried through in coordination with the still running projects RS 2703 and RS 169.

9. References

<u>Classification:</u> 12.3	
<u>Title 1 (Original Language):</u> Holografische Interferometrie (RS 102-22 -II.3.2, Jahresbericht A 76)	COUNTRY: BRD
	SPONSOR: BMFT
	ORGANIZATION: FhG-IfaM
<u>Title 2 (English):</u> Holografic Interferometry	<u>Project Leader:</u> Dr. W. Jüptner
<u>Initiated (Date):</u> 1.7.1975 <u>Status:</u> completed	<u>Completed (Date):</u> 30.9.1976 <u>Last Updating (Date):</u> December 1976

1. General Aim

Using holographic interferometry, the field of deformations of workpiece surfaces is registered by interferometric comparison of two states stored in a hologram. The appearing interference fringe system yields quantitative informations about the deformation in three axes. To produce the two stages the unloaded object has to be holographed, consequently it is subjected to the normal load and holographed again.

The field of deformation, obtained in this manner, can be used for calculation of the states of stress even of geometrically complicated workpieces. Furthermore by qualitative and quantitative evaluation weak points become evident. In fracture mechanical tests informations about the stress intensity could be obtained from experimentally determined plastic zones, and thus about the influence of crack displacement.

2. Particular Objective

It appears reasonable to apply the above mentioned advantages of holographic interferometry for safety controls of reactor components. However, before this method can be employed for components with possible interior cracks or other damages, on the basis of this research project the following problems have to be solved:

- a) Can defects be detected in components with thick walls?
- b) Which are the conditions for critical stresses to be determined from the holographic fringe system?

- c) Is it possible to recognize critical deformations in the region of weld seams?
- d) Which equipment is suitable for measurements without vibration protection?

These points shall be investigated in this project.

3. Research Program

- 3.1 Investigations to produce holograms without vibration isolation
 - 3.1.1 Holographic equipment with an output coupled laser
 - 3.1.2 One-beam holographic equipment
 - 3.1.3 Holographic equipment with a phase reference mirror
- 3.2 Deformation measurements of a specimen with internal cracks
 - 3.2.1 Construction of a tensile testing machine for holographic applications
 - 3.2.2 Experimental evaluation of the deformation field.

4. Experimental Facilities

- 4.1 Holographic equipment
including an Ar-Ion Laser with an acousto-optical
output coupler
- 4.2 Holographic equipment
Calculator Tektronix Scientist 909
Tensile testing machine 120 ton

5. Progress to date

- 5.1 Investigations to produce holograms without vibration isolation
An important condition during the taking of a hologram is that for the period of the exposure of the hologram, the relation of the two lightways with respect to the object and to the reference beam is only allowed to change by less than one tenth of the light wave length; that means less than approximately 10^{-4} mm:

$$S_{ob} - S_{ref} < 0,1$$

S_{ob} : Change of object beam path

S_{ref} : Change of reference beam path

This condition may be satisfied in different ways, depending on the case and whether vibration isolation of the object is possible:

- a) The hologram can be taken using a powerful laser, so that the object movement is small enough during the exposure / 1 /.
- b) The laser output can be controlled by the object movement. This leads to a stroboscope-like exposure, but without a well-defined frequency.
- c) The phase difference can be held constant while taking the hologram by a strong coupling of the two light waves. This means that the reference-wave will be modulated by the object / 2, 3, 4, 5 /.

Based on a theoretical evaluation the following methods were investigated experimentally.

5.1.1 Stroboscope-like exposure

The condition to use this method requires that the mean dislocation of the object from a reference position (not necessarily the rest position) decays with time. A hologram can be taken by exposing the photographic plate only while the object is in the reference position. To do this the position of the object must be exactly determined and when in the reference position, the laser light should be switched on.

This type of exposure is in effect "stroboscope-like". For the experiments a cw-Ar-laser with an acousto-optic output coupler inside the resonator was used. The laser is equipped with highly reflecting mirrors, so that without a signal the energy is stored in the resonator. A standing acoustical wave in the modulator deflects the beam according to a control signal and in this way the stored energy of the resonator is discharged in a pulse.

For determining the reference position of the object an acceleration gauge with low mass was fixed to the object and the signal processed by a special newly developed electronic equipment. When the acceleration signal (or respectively the derived path signal) reaches a limit which is variable depending on the application of the system, the modulation signal and subsequently the laser is switched off. By the aid of this device the holographic plate is only exposed while the object is in a region of 0.1 around the reference position. This technique was demonstrated by taking a hologram from a ship propellor without using vibration isolation.

5.1.2 Coupling of the light wave fields by a stiff connection between the object and the holographic plate.

It is easier to get a constant phase difference between the reference and the wave at the location of the holographic plate, when the object and the hologram are rigidly connected together. Additionally if the object is illuminated through the photo-plate, the illumination wave is both the reference beam and the object beam after the reflection on the object.

By this method the object-plate system is decoupled from movements of the light source.

First experiments were made with simple models. In this case, the object with the holographic plate, was put on a normal tripod standing directly on the floor. The system was disturbed by the normal vibrations. A steel sheet which was vibrating in natural modes was then holographed.

5.1.3 Coupling of the object and the reference beam by a phase mirror.

In another method one can use a reference mirror fixed to the object in order to diminish the phase difference between the object wave and reference wave caused by the movement of a non-vibration isolated object during the time of exposure / 6 /.

If the light source impinging on the object and the reference beam are the same, the phase differences related to vibration movements will theoretically disappear.

This was proved with two interesting examples; firstly, the deviation of the working table of an electron beam welding machine under tensile load was investigated. To do this, the mirror of the reference beam was put onto the table. Then the table was moved and the interference fringe system was regarded in real-time.

Furthermore, the deformation of a welded sample was measured in a Testatron tensile test machine. Although it was necessary to expose the holograms for about 10 s, the good quality of the holograms proved the efficiency of the method.

5.2 The deformation measurements were made with fracture mechanics specimen of the steel 22 NiMoCr 3 7. The sample was chosen in such a way as to allow a comparison of the experimental values with those of theoretical predictions from fracture mechanics. The dimensions of the specimen were $30 \times 120 \times 520 \text{ mm}^2$, and the dimension $B = 30 \text{ mm}$ corresponds to the thickness of the HDR.

The experiments should show, whether defects could be detected by holographic interferometry, and whether the danger of the defects could be quantified with the aid of fracture mechanics.

The measurements were made in specially adapted holographic equipment with a newly developed 100 to-tensile-testing machine.

5.2.1 Construction of a tensile-test system for holographic investigations.

The forces needed to simulate conditions prevailing in practice are in the order of 1 MN. Testing machines with such high loading were not available in IfaM at the time of the planning of these experiments.

Therefore a hydraulic testing system was designed and built in IfaM. It consists of three major

components:

- a) Four hydraulic cylinders with a maximum pressure force of 0.3 MN per cylinder.
- b) Two support crosses for the loading transfer from the cylinders to the sample.
- c) The hydraulic aggregate.

The whole system was tested up to a load of 0.9 MN, which corresponds to the fracture loading of the notched sample, with very good results.

5.2.2 Experimental evaluation of the deformation field.

For these measurements, the specimen ($30 \times 120 \times 520 \text{ mm}^3$) was prepared with internal cracks of variable length from 8 to 80 mm. With each crack length the specimens were loaded in steps of 0.05 MN ($= 14 \text{ N/mm}^2$ related to the undamaged cross section) up to a load of 0.5 MN = 140 N/mm^2 . This load approximates to that on the material in the HDR at a test pressure of 100 bar. The holographic-interferometric measurements were made using the double exposure method.

6. Results

6.1 Investigations in taking holograms without vibration isolation.

In order to conduct measurements directly at the reactor, it was necessary to investigate and refine the methods of taking holograms without the need for vibration isolation.

The following methods gave good results with respect to the taking of holograms and the making of interferometrical measurements:

6.1.1 Stroboscope-like exposure

6.1.2 One-beam holography

6.1.3 Phase reference mirror method

If the object is moving around a test position, it seems to be possible with these methods to compensate nearly all vibrations. But in all cases the techniques require more refinement before practical application is possible.

Furthermore, the investigations as part of this research work have shown, that it should be possible to measure deformations on reactor components by the means of holographic interferometry. During measurements on the pressure vessel, the holographic equipment is stiffly connected to the object. In this case the relative movements between the pressure vessel and the measuring equipment are small.

6.2 Deformation measurements with a flawed tensile test specimen.

The evaluation of the double-exposed holograms showed that when the crack in the middle of

the specimen reaches 10 mm (= 17 % of the cross section). In this case the deformation reaches 0.5 μm . With the largest experimentally investigated crack length of 80 mm, the measured results were compared to those from theoretical calculations. The measured deformation was approximately 30 % smaller than the calculated one. This result can be considered to be good in the light of the approximation necessary for the calculation.

6.3 Evaluation of the results.

The described investigations in the region of deformation measurements postulate a new application for holographic interferometry. Metallic materials were not previously investigated in this manner. The measurements of the internally flawed specimens have shown the physical influence at the unbroken surface as a result of the defects. The results lead to the conclusion that it should be possible, with improved methods of evaluating, to detect small cracks using this method. Apart from this positive aspect, it must be stated, that with the present state-of-the-art of holographic interferometry, it is not yet practical to detect critical defects by this means. However other applications may be possible. The investigations have also shown, that it should be possible to quantify the stress intensity in the region of a crack when the methods of holographic interferometry are promoted in this direction. Since the stress intensity in front of the defect is a value to describe the static and dynamic strength of the material, the potential danger of a defect can be quantified without other expedients. The future research should be directed to an investigation of this field of application.

9. References

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- / 3 / Corcoran, V.I.,
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- / 6 / Jüptner, W. Berichte zum Forschungsvorhaben RS 102 - 22
- / 7 / Jüptner, W. Untersuchungen zur zerstörungsfreien Prüfung von Reaktorkomponenten
mit Hilfe der holografischen Interferometrie.
Abschlußbericht zum Forschungsvorhaben.

<u>Classification:</u> 12.3	
<u>Title 1 (Original Language):</u> Ermittlung von Schweißbeigenspannungen mit Hilfe der Röntgenografie im ambulanten Einsatz (RS 13o - II.3.2, Jahresbericht A76)	<u>COUNTRY:</u> BRD <u>SPONSOR:</u> BMFT <u>ORGANIZATION:</u> Allianz, Isma- ning
<u>Title 2 (English):</u> The Determination of Welding Residual Stresses with the Aid of Ambulant X-Ray Diffraction	<u>Project Leader:</u> Dr. Christian Dr. Elfinger
<u>Initiated (Date):</u> 1.6.1974 <u>Status:</u> Finished	<u>Completed (Date):</u> 31.5.1976 <u>Last Updating (Date):</u> June 1976

1. General aim

The only method known today of determining the absolute value of residual stresses in metallic materials by non-destructive means is the x-ray diffraction stress measuring procedure. The aim of the work in progress is to clarify to what extent it is possible to incorporate the measurable residual stresses in the surface or in the surface-adjacent zones of welded joints into theoretical reflections on strength factors, and to perfect measuring methods and apparatus in such manner as to facilitate their ambulant application in respect of reactor components.

2. Particular objectives

The work is divided into two sections:

- a) Investigations into the evolution of surface stresses and stress gradients in and in the vicinity of submerged arc welded joints.
- b) Reconstruction of the x-ray diffraction measuring equipment for ambulant use.

3. Experimental facilities

For the x-ray diffraction measuring on specimen welds, two centre-point

free x-ray goniometers of Messrs. Siemens were available. For the ambulant measurements on major components, a newly developed goniometer was used.

4. Progress to date

Determination of the distribution of welding residual stresses laterally to the welded joint and the stress gradients dependent on the depth in submerged arc specimen welds has been completed. On the basis of model conceptions, endeavours were made to discuss the formation of welding residual stresses in the weld material and the heat-affected zone.

The new design and construction of a mobile goniometer has been completed and the first test measurements carried out (Fig. 1).

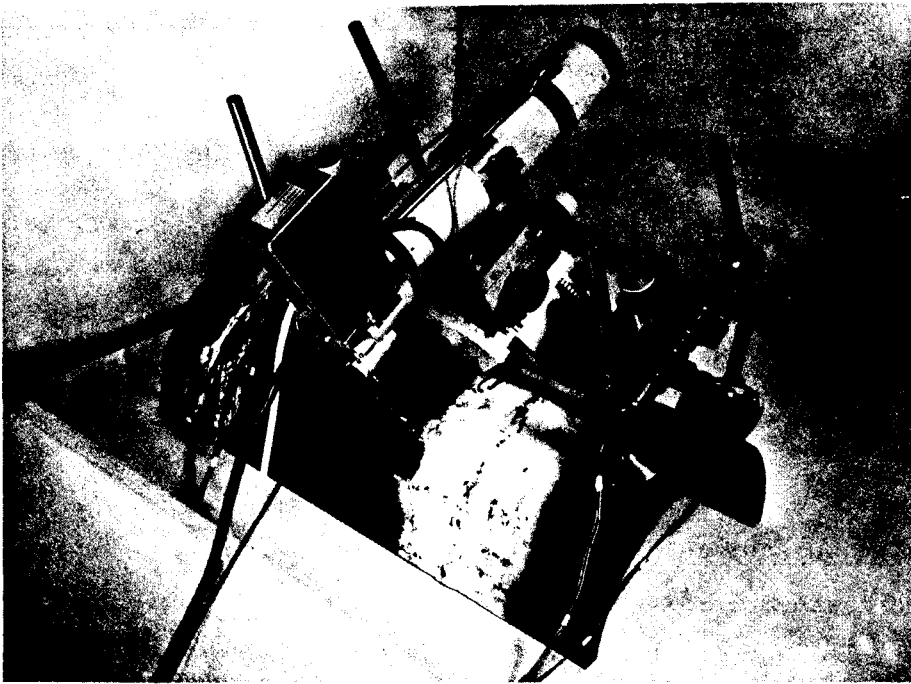


Fig. 1:

5. Results

The fundamental question of the work in hand, namely to what extent it is possible to employ x-ray diffraction measuring in determining residual stresses in weld seams, can be answered in the affirmative. In untreated

weld seams, the maximum stress peaks occur in the weld seam surfaces, the stress gradient being dependent on the depth whilst in surface-adjacent zones it is almost nil or very low. Thus, in spite of the shallow penetration depth of the x-rays, material zones are reached, whose residual stress state is characteristic for the weld seam to be measured. In mechanically treated weld seams, electrolytical removal of a maximum of 0,3 mm suffices to also penetrate into zones, which are unaffected by the treatment and reveal the characteristic residual stress state of the original weld seam.

The newly developed x-ray goniometer for mobile use shown in Fig. 1 permits easy application of the x-ray welding residual stress measuring procedure on components of any geometry down to a curvature radius of 150 mm. By virtue of the rigid connection between component and measuring equipment, it is possible for the first time to quantitatively employ the x-ray measuring method on major objects.

The test measurements conducted also yielded new data and information on the formation of welding residual stresses in weld material and heat-affected zones.

For the newly developed goniometer, patent (No. 2633144.6) and registered design (No. 762344.5) have been applied for at the German Office of Patents in Munich.

The work is completed and concluded.

6. References

Christian H., Elfinger F., Klug W.: Ein Röntgengoniometer für den mobilen Einsatz. Materialprüfung 18 (1976) Nr. 10, S. 388 - 390

<u>Classification:</u> 12.3	
<u>Title 1 (Original Language):</u> Weiterentwicklung von Ultraschall-Prüfverfahren und dazu erforderlichen Einrichtungen für wiederkehrende Prüfungen an Reaktordruckbehältern (RS 2703 - II.4.2, Jahresbericht A 76)	COUNTRY: BRD FRG
	SPONSOR: BMET
	ORGANIZATION: M.A.N., Nürnberg
<u>Title 2 (English):</u> Onward Development of Ultrasonic Inspection Techniques and Allied Equipment for In-Service Inspection of Reactor Pressure Vessels	<u>Project Leader:</u> Dipl.-Ing. J. Lindner
<u>Initiated (Date):</u> 1.2.1976 <u>Status:</u> Dec. 1976 Cotninueing	<u>Completed (Date):</u> 1.2.1978 <u>Last Updating (Date):</u> Dec. 1976

1. General Aim

Basing on the work carried out under the Federal Research Programme RS 27, RS 2701 and RS 2702 the target of the present project is to achieve further progress in the development of inspection techniques and equipment for in-service inspection of pressure vessels and, on completion of the work described herein, to establish a means of remote-operated volumetric inspection of the entire reactor pressure vessel according to present-day technological standards. This development project relates to ultrasonic inspection techniques only. Volumetric inspection of the entire pressure vessel is a target which calls for all participant firms to make further progress in the development of inspection systems and inspection techniques, manipulators and electronic equipment. Participants in this programme are the Federal Institute for Materials Testing and the firms of Kraftwerk Union AG, Krautkrämer GmbH and Maschinenfabrik Augsburg-Nürnberg AG (M.A.N.).

2. Particular Objectives

The programme is sub-divided into the following areas:

1. Basic studies of ultrasonic inspection methods and optimization of detection and analysing techniques.
2. Conceptual design of inspection systems to improve the probability of fault detection in different inspection areas:
3. Development, fabrication and trials of internal and external inspection manipulators with a view to achieving the goal of complete volumetric inspection of the reactor pressure vessel by automated equipment.
4. Onward development of components and methods to improve data logging, data processing and display/printout systems.

3. Research Programme

The research programme embraces the following work:

3.1 General

The general aim of the research programme is to make further progress in the development of ultrasonic inspection techniques with a view to increasing the probability of fault detection and improving the interpretation of the inspection results. This includes research into techniques for the analysis of indications, e.g., focus technology or line holography, and preparations for practical application. The tasks call for extensive theoretical and experimental research.

3.2 Inspection Systems

Special inspection systems must be developed for the inspection of close-to-surface zones and of lugs and supports welded to the inside and outside of the reactor pressure vessel. The

task involves testing a method of monitoring the stability of the probes during the course of inspection work. The task also includes the development of a probe with adjustable sound beam parameters for special inspection applications, e.g., for inspecting the ligaments in areas of penetration holes for control rods.

3.3 Manipulators

New manipulators are being developed and current manipulators are being improved for remote-controlled inspection of the following areas of reactor pressure vessels:

- Ligaments in areas of control rod penetration holes in the bottom closure and closure head
- External inspection of transition weld between cylindrical area of vessel and spherical bottom (BWR)
- Inspection of inner radii of small nozzles
- External inspection of flange at bottom of vessel and at closure head
- Extended external inspection of cylindrical area of vessel with automatic transfer of inspection car from one longitudinal rail to another. Motions under computer control.

3.4 Data Logging

For processing the data of automatic ultrasonic inspection the capacity of data logging equipment must be increased to accommodate the increased flow of data. Existing programmes are being increased to improve software and the interpretation quality of display and printouts.

4. Experimental Facilities, Computer Codes

The research programme envisages laboratory tests to establish the capability of inspection equipment for individual inspection techniques as well as the provision of working models of manipulator components. Various computer codes are used to establish the best parameters of inspection systems. The task includes the provision of further computer codes for evaluation and display/printouts of the data obtained during inspection.

5. Progress to Date

Actual test results from a large vessel wall specimen are available for a closer examination of perhaps significant indications with the aid of special methods of analysis. The following four methods of analysis were tested:

Single-probe dual-frequency echo dynamics
Tandem echo dynamics
Tandem echo dynamics with focus head
Line holography

Conceptual data and designs are available on the inspection systems to be developed for the individual areas. Prototypes are being built to test the capabilities of these systems. For the probe with adjustable sound parameters there are certain dimensional limitations because the probe is also proposed to be used in areas of restricted space.

Conceptual designs have been prepared for the various manipulators. Detailed designwork has been started.

For data logging and data display/printouts, work has commenced on the further development of software for on-line

data processing and on increasing the data flow to all measuring channels.

6. Results

Most of the work is still in progress and no detailed results are as yet available.

The tests on the methods of analysis have shown that according to present knowledge focus technology and line holography appear to be particularly suitable methods for the analysis of indications. Computer codes are available for calculating the three-dimensional tandem echo dynamics.

7. Next Steps

Work under this programme will be continued in all areas with special emphasis on:

- Testing of new inspection systems on test specimens
- Study of various parameters as to the quality of indications for focus technology
- Testing of working models of new manipulator elements
- Onward development and testing of improved data display/printout methods.

8. Relation with Other Projects

Reference is made to what has been said in Report A 74 with regard to RS 2702.

9. References

During the period under review the following summarized reports have been issued in the German language:

<u>Author</u>	<u>Title</u>	<u>Date</u>
BAM	Three-dimensional Echo Dynamics for Tandem Technology	September 76
BAM	Fault Analysis on a Large Vessel Wall Specimen	October 76

10. Degree of Availability of the Reports

A relatively small number of reports have been issued.
A limited number are still obtainable from the author or
from GRS, Köln.

<u>Classification:</u> 12.3	
<u>Title 1 (Original Language):</u> Entwicklung und Bau einer Ultraschall-Prüf- und Auswertelektronik einschließlich Vorverstärkerkasten RS 169 - II.3.2, Jahresbericht A 76	<u>COUNTRY:</u> BRD
	<u>SPONSOR:</u> BMFT
	<u>ORGANIZATION:</u> Krautkrämer GmbH
<u>Title 2 (English):</u> Development and construction of an electronic test and evaluation system with ultrasound including pre-amplifier box	<u>Project Leader:</u> Ing.grad. G. Gutmann
<u>Initiated (Date):</u> July 1, 1975	<u>Completed (Date):</u> December 1977
<u>Status:</u> Continuing	<u>Last Updating (Date):</u> December 1976

1. General Aim

Development of ultrasonic evaluation electronics meeting today's technical safeguarding requirements, universally applicable with all types of reactors.

Due to the experiences made so far and to the technical development as it is foreseeable in the future, the following demands were made on this electronic evaluation system:

2. Particular Objectives

Maximum number of channels: 60

Maximum speed of the manipulating system to be reached: 100 mm/s

Scanning rate: 1 test per mm

And, the installation is to be controlled continuously for function and stability.

3. Research Program

- Drawing up of the detailed operating program with arrow diagram.
- Development of some new evaluation modules
 - a) min/max-value-memory
 - b) noise investigations
- Development and construction of the electronic evaluation system corresponding the demands made.
- Testing the interplay between the modules developed.
- Projection works with regard to the construction of the prototype.
- Construction of the prototype.
- Investigations for noise suppression at the prototype of multiple wire systems.

4. Progress to Date

The works of the development program listed under 3. have all been accomplished but the last two points

- Construction of the prototype
- Investigations for noise suppression at the prototype of multiple wire systems.

The works on these two points had to be deferred because of details demanding clarification before starting on the definite construction of the electronic system, such as number of channels and possibly integration of modules and/or drafts of the research plan RS 102-17.

5. Results

All electronical modules, especially needed for this electronic evaluation system have been developed and tested as function sample. The demands to be met on function, exactitude, and stability of the modules have been fulfilled.

While carrying on construction and development of the general aim the following principal requirements were satisfied:

- a) Maximum number of channels: 60
- b) Maximum speed the manipulating system can realize: 100 mm/s
- c) Scanning rate: 1 test per mm
- d) All-automatic and permanently effective control of function and stability.
- e) Maximum stability tolerance permissible of ± 1.5 dB.

The aims set in respect of function and stability have been reached.

Production can be started following clarification of the definite extension steps on the prototype.

6. Next Steps

- Synchronisation with research plan RS 102-17 and start of the layout works for the construction of the prototype.
- Construction of the prototype
- Investigations for noise suppression at the prototype of multiple wire systems.

<u>Classification:</u> 12.3								
<u>Title 1 (Original Language):</u> Erhöhung der Auffindwahrscheinlichkeit von Fehlern durch Verbesserung des Signal-Rausch-Verhältnisses bei der Ultraschallprüfung (RS 190)	COUNTRY: BRD							
	SPONSOR: BMFT							
	ORGANIZATION: Battelle-Institut e.V.							
<u>Title 2 (English):</u> Increasing the Probability of Flaw Detection by Improving the Signal-to-Noise Ratio in Ultrasonic Testing	Project Leader: R. von Klot							
	<table border="0" style="width: 100%;"> <tr> <td><u>Initiated (Date):</u> 1.1.1976</td> <td><u>Completed (Date):</u></td> </tr> <tr> <td></td> <td>31.8.1976</td> </tr> <tr> <td><u>Status:</u></td> <td><u>Last Updating (Date):</u></td> </tr> <tr> <td>Continuing</td> <td>December 31, 1976</td> </tr> </table>	<u>Initiated (Date):</u> 1.1.1976	<u>Completed (Date):</u>		31.8.1976	<u>Status:</u>	<u>Last Updating (Date):</u>	Continuing
<u>Initiated (Date):</u> 1.1.1976	<u>Completed (Date):</u>							
	31.8.1976							
<u>Status:</u>	<u>Last Updating (Date):</u>							
Continuing	December 31, 1976							

1. General Aim

The project is aimed at investigating whether it is possible in ultrasonic testing to improve the signal-to-noise ratio by application of the signal averaging technique.

2. Particular Objectives

It is to be investigated whether signal averaging can improve the signal-to-noise ratio of the echoes from flaws close to claddings (e.g. in reactor pressure vessels and primary pipe systems).

3. Research Program

The investigations were carried out on a clad specimen with a bore (diameter 5 mm) near the cladding and on a large vessel wall test specimen with two cracks originating directly below the cladding (crack depth 7.2 mm and 14.7 mm).

4. Experimental Facilities, Computer Codes

A commercial materials testing instrument and an averaging computer were used for the investigations. The exponential average value is computed by the following algorithm:

$$A_n = S_n/G + A_{n-1} \cdot (1 - 1/G)$$

S_n	n-th input signal	A_n	stored result after n sweeps
G	weighting factor	A_{n-1}	stored result after n-1 sweeps

5. Progress to Date

The echo amplitude and the signal-to-noise ratio were measured. The velocity of the probes, the direction of the probe velocity relative to the defect (parallel \parallel and perpendicular \perp), and the weighting factor were varied in these measurements.

6. Results

For the perpendicular direction of the probe velocity, the amplitude of the echo from the defect first remains constant and then drops rapidly as the weighting factor increase (Fig. 1). For the parallel velocity direction almost no drop occurs up to $G = 2^{10}$.

The signal-to-noise ratio first rises and then, after a maximum, declines as the weighting factor increases (Fig. 2). With growing probe velocity the maximum is shifted toward smaller values of the weighting factor and thus is reduced.

In the vessel wall test specimen the echo amplitudes for probe velocities of 2.0 mm/s and 4.0 mm/s are almost identical.

The signal-to-noise ratio passes through a maximum as a function of the weighting factor (Fig. 3). The maximum increases with rising probe velocity, but is not shifted. The signal-to-noise ratio can be improved by up to 10 dB.

7. Next Steps

The probe velocity is to be increased to about 50 mm/s by modification of the averaging computer. Defects close to claddings will be analyzed by single and tandem probes from the clad side and from the reverse side.

8. Relation with Other Projects

9. References

R. von Klot, A. Sahm: Erhöhung der Auffindwahrscheinlichkeit von Fehlern durch Verbesserung des Signal-Rausch-Verhältnisses bei der Ultraschallprüfung, RS 190, Report of August 1976 by Battelle-Institut e. V., Frankfurt am Main, to the Federal Ministry of Research and Technology (in German).

10. Degree of Availability of the Reports

The above-cited report is available upon request from Bundesministerium für Forschung und Technologie, Referat 313, Postfach 120 370, D-5300 Bonn 12.

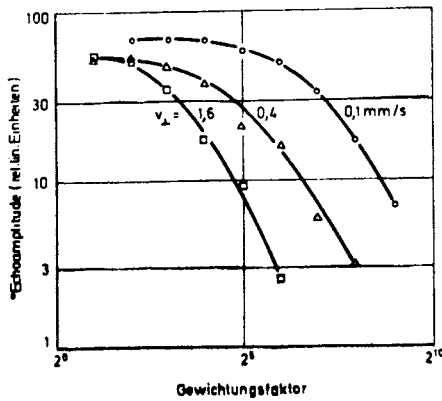


Fig. 1: Amplitude of the echo from the bore in the specimen as a function of the weighting factor for different probe velocities v_{\perp} . Shear waves 4 MHz.

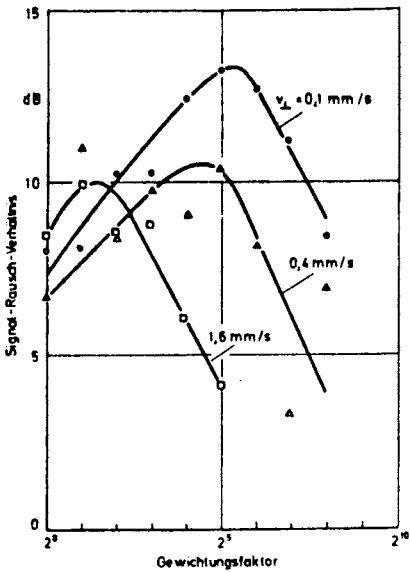


Fig. 2: Signal-to-noise ratio of the echo from the bore in the specimen as a function of the weighting factor for different probe velocities v_{\perp} . Shear waves 4 MHz.

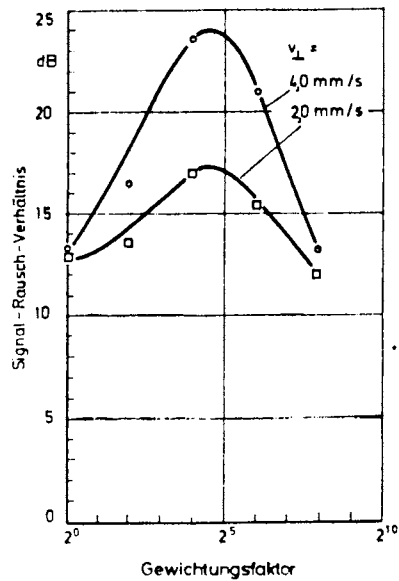


Fig. 3: Signal-to-noise ratio of the echo from the crack (depth 7.2 mm) in the vessel wall test specimen as a function of the weighting factor for different probe velocities v_{\perp} . Shear waves 1 MHz.

<u>Classification: 12.3</u>	
<u>Title 1 (Original Language):</u> Studie über die Anwendung des Wirbelstromimpulsverfahrens zur Qualitätssicherung von Kernkraftwerkskomponenten (RS 229 - II.3.2, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: BAM, Berlin
<u>Title 2 (English):</u> Study on the application of pulsed eddy current methods to quality control of nuclear power plant components	<u>Project Leader:</u> Dr. G. Wittig
<u>Initiated (Date):</u> 1.10.76 <u>Status:</u> Continuing	<u>Completed (Date):</u> 31.12.1977 <u>Last Updating (Date):</u> December 1976

1. General Aim

Study of the possibilities of applications of the pulsed eddy current method to nondestructive testing of nuclear reactor components. The evaluation of the signals generated by the test system has to deliver informations about present defects with a high degree of statement power in a simple manner. The obtained results shall permit a critical evaluation if experimental investigations will result in successful solutions of detailed testing problems.

2. Particular Objectives

Ascertainment of the state of knowledge about the theoretical foundations and the foundations in the field of technical applications. Critical examination of the evaluation of test and investigation results from practical use of the method. Obtaining of statements about possibilities of application of the pulsed eddy current method for important test problems in the field of nuclear reactor components.

3. Research Program

3.1. Compiling of informations

3.2. Working of statements about possibilities of application of the method

3.3. Estimations of further developments

4. Experimental Facilities, Computer Codes

Program for numerical calculation of distributions of magnetic fields and eddy currents.

5. Progress to Date

Ad 3.1. Evaluation of present and procured literature.

Visit to KWU with discussions about test problems in question.

Ad 3.2. Numerical calculation of temporal and local distribution of the magnetic field and the eddy current density in a half-space with electrical conductivity of austenitic steel. Excitation by a plane magnetic field of variable duration.

6. Results

Program for calculation of field and eddy current distributions.

The project is in an initial phase in this time.

7. Next steps

Ad 3.1. Continuation of the evaluation of reports and of the procurement of informations about test problems.

Ad 3.2. Further calculations of field and eddy current distributions.

8. Relation with other Projects

- RS 89 - II.3.2

- RS 102- 18 - II.3.2

9. References

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10. Degree of Availability of the Reports

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<u>Classification:</u> 12.3	
<u>Title 1 (Original Language):</u> Mehrfrequenz-Wirbelstromprüfung Phase 1: Aufbau eines Mehrfrequenz-Geräteprototyps (RS 231 - II.3.2, Jahresbericht A 76)	<u>COUNTRY:</u> BRD
	<u>SPONSOR:</u> BMFT
	<u>ORGANIZATION:</u> Fraunhofer-Ges., IzFP, Saarbr.
<u>Title 2 (English):</u> Multifrequency - eddy current testing Phase 1: Construction of a multifrequency prototype	<u>Project Leader:</u> Dipl.-Phys. R. Becker
<u>Initiated (Date):</u> 1.9.1976	<u>Completed (Date):</u> 29.2.1977
<u>Status:</u> Continuing	<u>Last Updating (Date):</u> December 1976

1. General Aim

The project has the aim of building up a prototype for the inspection by means of the multifrequency eddy current method. Starting from the results of RS 102-18, section "eddy current", the equipment shall be applicable to the inspection in production lines as well as to the recurrent inspection of reactor components of the type BWR, PWR and FSR. The concept of the equipment shall harmonize with the conditions of the large application field. Over all, great importance must be spent upon safety against disturbances and upon easy handling.

2. Particular Objectives

The equipment consists of 2 parts:

- the transmitter-receiver-unit:

The frequencies are given after multiplexing each after the other into the coil. The switching is so fast, that an inspection velocity of 3 m/s with a local resolution of 1 mm is achieved. The advantage of this proceeding compared with the simultaneous method is the small electronic expense and the flexibility concerning the quantity of frequencies and the frequency range.

Respectively, in one of 3 ranges the frequencies can be tuned continuously and in any combination:

range 1: 10 kHz - 600 kHz
range 2: 1 kHz - 60 kHz
range 3: 0.1 kHz - 6 kHz

Besides the oscillators the transmitter-receiver unit consists of the coil driver, the amplitude-phase-meter, the compensation network and the display to figure the test information (impedance of the coil). The coil is driven over a cable, which has a length of up to 30 m and can be an absolute or a differential type. In both cases the compensation is made in an automatical manner.

- the evaluation unit:

This performs the separation of the parameters and produces the read-out value, which is settled of the disturbing signals and is only a function of one parameter to be chosen. The evaluation is realized by means of digital computer circuits and allows the simultaneous determination of several read-out values coordinated to different parameters.

3. Research Program

The electronic circuits and components are to develop and to integrate within the prototype described in 2.

4. Progress to Date and Results

The block diagram of the equipment is worked out upon the experiences with the laboratory test device, which is arisen in the preceding project RS 102-18.

The building up of the transmitter-receiver unit is short of conclusion.

5. Next Steps

Construction of the evaluation unit will be started and connection of the both units will be realized.

<u>Classification:</u> 12.3	
<u>Title 1 (Original Language):</u> Untersuchungen zur Leistungsfähigkeit der akustischen Holographie, vor allem im Vergleich zu fokussierenden Prüfköpfen bei der ZfP (RS 0230 - II.3.2, Jahresbericht A 76)	<u>COUNTRY:</u> BRD
	<u>SPONSOR:</u> BMFT
	<u>ORGANIZATION:</u> BAM
<u>Title 2 (English):</u> Investigations of the efficiency of acoustical holography especially in comparison to focussed beams in NDT	<u>Project Leader:</u> Dr. J.Kutzner Dr. H.Wüstenberg
<u>Initiated (Date):</u> 1. 10. 1976 <u>Status:</u> Continuing	<u>Completed (Date):</u> 31. 3. 1979 <u>Last Updating (Date):</u> December 1976

1. General Aim

Development of simplified methods of
 - acoustical holography with numerical reconstruction
 - flaw sizing by scanning with focussed beams.
 Application of the developed methods to the analysis of ultrasonic indications under practical circumstances.

2. Particular Objectives

-

3. Research Program

3.1 Construction of probe systems

Determination of the decisive criterions (aperture length, resolution etc.) to construct optimal probe systems.
 Application of the developed probe systems on test-blocks.

3.2 Software-Development

Elaboration of theoretical foundations (normal incidence, oblique incidence, tandem-technique, distance of scanning points), development of the reconstruction- and interface-software (AD-converter, plotter) for scanning in one- and two-dimensions.

- 3.3 Holography in tandem-technique
Investigations of the efficiency of this method, experiments on test-blocks.
- 3.4 Construction and testing of focussing probes
Checking of the available construction method, manipulator for focussing probes, adaption of the focussing probes on the curvature of the surface.
- 3.5 Experiments with the probe-systems developed for holography- and focussing systems.
Testing of the developed systems on the test-blocks, which were produced in other research programs of reactor security, estimation of the efficiency of the developed techniques.

4. Experimental Facilities, Computer Codes

- To 3.1 Manipulator system, electronic equipment for acoustical holography
- To 3.2 Reconstruction software (FORTRAN), driver-software for AD-converter, CRT-terminal and plotter (Assembler)
- To 3.3 Tandem-manipulator, electronic-equipment
- To 3.4 Device to the production of acoustical lenses, digital equipment for positioning
- To 3.5 Whole equipment

5. Progress to Date

- To 3.1 Some theoretical investigations to the resolution, construction of a two transducer-probe, first experiments.
- To 3.2 Testing of the developed reconstruction- and interface-software for linear scanning in normal-, oblique- and tandem-incidence.
- To 3.3 Software for linear scanning.
- To 3.4 -
- To 3.5 -

6. Results

- To 3.1 The resolution is determined mainly by the divergency of the probes.

To 3.2 The software-development for linear scanning is completed.

To 3.3 The software-development for linear scanning in tandem-
technique is also completed.

To 3.4 -

To 3.5 -

7. Next Steps

To 3.1 Continuation of the theoretical investigations, testing
and further development of the constructed probe.

To 3.2 Theoretical and experimental investigations to get
informations about the distance of the scanning points.
Writing a technical report to the equipment for linear
acoustical holography (hardware and software).

To 3.3 -

To 3.4 Checking of the available construction method by reference
tests.

To 3.5 -

8. Relation with other Projects

Real-time ultrasonic holography (IZfP-Saarbrücken),
reactor security research program RS 2703.

9. References

-

10. Degree of Availability of the Reports

IRS, Köln

PROJECT TITLE : Test of switches for protective functions	CSNI INDEX NUMBER : 12.3
SPONSORING COUNTRY : Sweden	ORGANISATION : AB Atomenergi
DATE INITIATED : November 1975 DATE COMPLETED : March 1977	PROJECT LEADER : Pehr E Blomberg

Description :

The objective is to perform functional tests on different types of switches, used for initiating protective actions in case of tube breaks or leakages in a reactor containment or in a turbine building and to perform check of computing methods, for estimating the pressure build-up and temperature distribution under transient conditions in the containments.

About 30 switches have been installed in the test room, consisting of a concrete hall, size 8 x 13 x 8 metres, into which steam at a temperature of 245°C is blown.

The series of steam blow tests started during fall 1976.

Reference documents are reports to the Swedish Nuclear Power Inspectorate.

PROJECT TITLE : Test with safety valves	CSNI INDEX NUMBER : 12.3
SPONSORING COUNTRY : Sweden	ORGANISATION : AB Atomenergi
DATE INITIATED : 1975-04-01 DATE COMPLETED : 1977-12-31	PROJECT LEADER : Owe Strandberg

Description :

The project includes design and building of a test rig for testing the impulse valves of the safety main valves, as well as tests of those impulse valves. The aim of these tests is to investigate the reasons for malfunction of the valves and to recommend methods for improving their design.

During the tests measurements on actuating pressure, valve-piston stroke, mass flow rate, etc are carried out.

12.4. Human factors
(no entries)

13. SYSTEMS OPTIMIZATION, STANDARDISATION, NEW CONCEPTS
(e.g. integrated primary circuit, new containment concepts)

<u>Classification: 13</u>	
<u>Title 1 (Original Language):</u> Verminderung der Primärkreiskontamination durch Einsatz eines Elektromagnetfilters. Versuche zur prinzipiellen Durchführbarkeit (RS 171 - II.1.6 , Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: KWU, Erlangen
<u>Title 2 (English):</u> Reduction of the contamination of primary system components by electromagnetic filters	<u>Project Leader:</u> Dr. Neeb
<u>Initiated (Date):</u> 1. 7. 75 <u>Status:</u> Completed	<u>Completed (Date):</u> 30. 9. 76 <u>Last Updating (Date):</u> 31. 12. 76

1. General Aim and 2. Particular Objectives

The possibilities for reduction of corrosion product activity and contamination by means of electromagnetic filters, which have been used in fossil hatered plants have been studied. Some tests were conducted in order to find out, whether electromagnetic filters could be used in the primary system of a PWR.

3. Research Program

- a) Tests on the efficiency of electromagnetic filters were conducted, using low concentrations of corrosion products in highly cleaned water at temperatures up to 300 °C
- b) Investigation of the influence of boric acid and LiOH with concentrations as used in the coolant system of a PWR
- c) Metal release rates of the balls of the electromagnetic filter in boric acid and LiOH-solutions, as far as possible in this experimental facility
- d) The possibility of backflushing has been tested with respect to the corrosion product oxides, separated on the balls of the electromagnetic filter

4. Experimental Facilities

The experimental studies have been carried out in the Pressurized Water Chemistry Loop (DCK) of KWU-Erlangen. A high-pressure electromagnetic filter has been fabricated and was used in connection with the DCK. Measurements by normal and radio-tracer analytical techniques were done by the KWU laboratories.

5. Progress to Date

The radio-tracer tests of radioactive corrosion products were continued, using added corrosion products from KWO and KWB-A. Samples were taken at the entrance and the outlet of the electromagnetic filter, the Fe content and the distribution of radio-tracers were investigated.

The backwashing of the corrosion products, which were deposited at high temperatures, was tested several times.

The influence of boric acid on the deposition of particles was tested at a boron content of 2000 ppm.

The activity distribution was measured on the electromagnetic balls and the pipe walls when the electromagnetic filter was switched off.

6. Results

The radio-tracer tests needed a long time, because the activity was rather low. The result was that only a rather low deposition rate was recognized.

The presence of boric acid has no influence on the deposition of particles on the electromagnetic filter.

7. Next Steps

The work has been completed.

8. Relation with Other Projects

9. References

10. Degree of Availability

<u>Classification: 13</u>	
<u>Title 1 (Original Language):</u> Vergleich verschiedener Druckbehälterkonzepte für Leichtwasserreaktoren (RS 219 - II.1.6, Jahresbericht A 76)	COUNTRY: BRD
	SPONSOR: BMFT
	ORGANIZATION: Battelle-Institut e.V., Ffm.
<u>Title 2 (English):</u> Comparison of Different Pressure Vessel Concepts for Light-Water Reactors	Project Leader: G. Langer
<u>Initiated (Date):</u> September 1, 1976 <u>Status:</u> Continuing	<u>Completed (Date):</u> June 30, 1977 <u>Last Updating (Date):</u> December 31, 1976

1. General Aim

The aim of the study is the clear presentation of the characteristic properties of various pressure vessels and a comparison of these pressure vessels on the basis of general comparative criteria. With the aid of a technological assessment, the most suitable pressure vessel concept for a possible back-up solution for the thick-walled steel pressure vessel is to be ascertained.

2. Particular Objectives

3. Research Program

The study will be carried out essentially in two steps.

3.1. Collection of the Necessary Data

The information necessary for comparison will be collected by a literature survey. It will be supplemented by data obtained in interviews and discussions with manufacturers and consultants.

3.2. Technological Assessment of the Individual Concepts

The individual concepts will be assessed by utility value analysis.

4. Experimental Facilities, Computer Codes

5. Progress to Date

The following work was performed in the report period:

- Drawing up of a detailed plan of the report
- Completion of about one fourth of the volume of the research program outlined in Section 3.1
- Performance of the first investigations of Section 3.2

It is planned to include the following concepts in the research program:

- Thick-walled steel pressure vessel
- Multi-layer pressure vessel
- Pressure vessel produced by build-up welding
- Prestressed concrete pressure vessel
- Prestressed cast-iron pressure vessel
- Prestressed cast-steel pressure vessel

6. Results

The comparative criteria used in the study require detailed technical data. Because of the very low level of development, however, these are not available for all the concepts listed above. This applies to the concepts of the pressure vessel produced by build-up welding and the prestressed cast-steel pressure vessel.

7. Next Steps

Continuation of the work according to the research program outlined in Section 3.

8. Relation with Other Projects

-

9. References

-

10. Degree of Availability of the Reports

-

PROJECT TITLE : UNDERGROUND NUCLEAR POWER PLANTS	CSNI INDEX NUMBER : Water reactors 13
SPONSORING COUNTRY : Switzerland	ORGANISATION : EIR
DATE INITIATED : 1975 DATE COMPLETED : ---	PROJECT LEADER : J. C. Mayor

Description :

The safety potential of the underground siting of nuclear power plants is being investigated.

Two alternatives are, at present, under investigation: a rock cavity plant, completely underground, in a hill-side and the cut-and-fill concept for a LWR nuclear island.

Emphasis is given also to civil engineering works and to geotechnical problems.

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31.719

NUCLEAR
SAFETY RESEARCH INDEX

1977

VOLUME II
WATER REACTORS
(continued)

14. PROBABILISTIC METHODS OF SAFETY ANALYSIS

PROJECT TITLE : A Systems Analysis Approach for Establishment of Risk Criteria for Nuclear Power Generation	CSNI INDEX NUMBER : 14.
SPONSORING COUNTRY : Canada (Atomic Energy Control Board)	ORGANISATION : University of Toronto
DATE INITIATED : June 1975 DATE COMPLETED : September 1977	PROJECT LEADER : Prof. C.R. Phillips

Description :

1. General Aim

- (i) To determine and place the safety and environmental risks associated with the nuclear industry in Canada in perspective with those associated with other energy alternatives, with industrial occupational exposures, and with other societal and individual risks.
- (ii) To recommend appropriate risk criteria for regulatory purposes.

3. Experimental Facilities and Programme

June 1975 - June 1976

- (a) study of environmental effects of nuclear and non-nuclear forms of energy generation
- (b) development of methodology for determining and assessing socio-technical risk parameters

June 1976 - June 1977

- (a) determination and assessment of risk parameters
- (b) assessment of energy demand and supply in Canada and review of energy resource options
- (c) completion of environmental and safety factors in various industries.

4. Project Status

To date the bulk of the effort has been focussed on the gathering of large amounts of accident and fatality data from the nuclear, fossil fuel and metal mining industries and on gathering information on environmental and health effects. Very detailed reports on this data have been received over the life of the project but synthesis of this information into a comprehensible picture of comparative risk is less complete. A report entitled "Risks of Nuclear and Non-Nuclear Sources", which comprises Section 3 of the general report, was submitted. This section compares the industrial and public health (but not environmental) effects of the energy alternatives. This section concludes that hazards from coal generated electricity (in terms of fatalities, injuries and impaired health) are substantially higher than those of nuclear based energy.

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PROJECT TITLE : A Systems Analysis Approach for Establishment of Risk Criteria for Nuclear Power Generation	CSNI INDEX NUMBER : 14.
SPONSORING COUNTRY : Canada (Atomic Energy Control Board)	ORGANISATION : University of Toronto
DATE INITIATED : June 1975 DATE COMPLETED : September 1977	PROJECT LEADER : Prof. C.R. Phillips

Description :

Continuation

5. Reference Documents
 Final report (University of Toronto) in draft.
9. Budget
 \$50,000

OECD NUCLEAR ENERGY AGENCY - 38, Boulevard Suchet, 75016 PARIS
Telex : 63668 AEN NEA - Tel : 524 82-00

14.560

PROJECT TITLE : The Process of Risk Assessment in National, Provincial and Local Policy Decisions on Nuclear Power	CSNI INDEX NUMBER : 14.
SPONSORING COUNTRY : Canada (Atomic Energy Control Board)	ORGANISATION : University of Toronto
DATE INITIATED : June 1975 DATE COMPLETED : December 1977	PROJECT LEADER : Prof. I. Burton

Description :

1. General Aim

To study the process of risk assessment in the field of nuclear energy production, in Canada, the United States, and the United Kingdom. The study is to be descriptive rather than prescriptive and will attempt to describe the processes whereby risks are assessed and the use of such assessment information. It will not make any actual assessment of the risks or propose how these risks should be assessed.

3. Experimental Facilities and Programme

Preparation of questionnaires to determine what has been the process of risk assessment in policy decisions at the national, provincial and municipal levels.

Random sample interviews with members of the public and interviews with officials at the national, provincial, and municipal levels.

International comparison of information and preparation of consolidated report by the US researcher, Clark University, "Risk Assessment in Rare Events: A Three Nation Study of Disaster Prevention in Nuclear Energy Programmes." This report has since been dropped in favour of a jointly authored book.

4. Project Status

This research project is part of a larger international study (U.K., U.S. and Canada) in which the socio-political perception and the methods of technical assessment of the risks associated with nuclear power are being compared among the participating countries. Although the total project is primarily funded by a Ford Foundation grant, the AEBC has supplemented the funding of the Canadian component with a contract. The Canadian component of the study has consisted of "key actor" interviews with industrial, government and political leaders as well as "man-in-the-street" opinion sampling. A first round of these interviews is complete but is being repeated to take advantage of lessons learned in the earlier round and to

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14.560

PROJECT TITLE : The Process of Risk Assessment in National, Provincial and Local Policy Decisions on Nuclear Power	CSNI INDEX NUMBER : 14.
SPONSORING COUNTRY : Canada (Atomic Energy Control Board)	ORGANISATION : University of Toronto
DATE INITIATED : June 1975 DATE COMPLETED : December 1977	PROJECT LEADER : Prof. I. Burton

Description :

Continuation

add a time base to the results. An extensive survey of the methods used to estimate risk is also being carried out. The final report will take the form of a book on which researchers from each of the three countries will collaborate. We have in this reporting period received copies of two papers which have been published as part of this study.

5. Reference Documents

Kasperson et al. "Confronting Nuclear Hazards: British, Canadian and American Approaches", Occasional Paper, 1976 #2. Institute of International Studies, U. of S. Carolina.

Whyte, A. "Public Perception of Nuclear Risks in Canada, U.K. and U.S.A.", Paper presented at Conference on Socio-Psychological Aspects of Nuclear Energy; Paris, 13-15 January, 1977.

9. Budget

~ \$50,000

OECD NUCLEAR ENERGY AGENCY - 38, Boulevard Suchet, 75016 PARIS
Telex : 63668 AEN NEA - Tel : 524 82-00

14.560

PROJECT TITLE : Reliability project	CSNI INDEX NUMBER : 14, (6)	
SPONSORING COUNTRY : Finland	ORGANISATION : Technical Research Centre Reliability Project Group	of Finland (Electr. Eng. Lab.
DATE INITIATED : 1970 DATE COMPLETED : Continues	PROJECT LEADER : Ph.D Pentti Uuspää	

Description :

The objectives of the reliability project are to study and develop theoretical and practical means of reliability analysis, to acquire necessary knowledge and relevant statistical data and to apply the methods and skill developed especially to the analysis of the nuclear power plants being built in Finland now and in the future, taking also into consideration the needs for personnel training and education and the regulations laid down for nuclear safety.

The methods of reliability analysis as well as the numerical calculation methods of system reliability have been developed. The most important program is a Monte Carlo simulation program SAFTE-LR, but also analytical methods for hand-calculation have been developed.

The Reliability Group has made very many safety related reliability analyses for Loviisa and TVO I nuclear plant. The Group has also taken part in some nuclear plant safety common studies with Nuclear Engineering Laboratory. A project for developing the nuclear plant reliability data system in Finland has just begun.

Titre Probabilités d'accident : Evénements initiateurs d'accidents sur les PWR et séquences d'accidents.	Pays : FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Accident probabilities : Accident initiating events on PWR and accidents sequences.	Organisme exécuteur : CEA/DSN -SETSSR
	Responsable : MME A. CARNINO (SETSSR)
Date de démarrage : 01/09/76 Date prévue d'achèvement : 31/12/81 Etat actuel : démarrage Dernière mise à jour : 02/11/76	Scientifiques : JP. SIGNORET J. DUBAU

Objectif général :

Détermination des événements initiateurs d'accidents par perte des différentes barrières dues à des agressions. Construction des arbres d'événements à partir de ces initiateurs et détermination de séquences accidentelles.

Etat de l'étude :

Avancement à ce jour :

Principe de la méthode.

Prochaines étapes :

Détermination des initiateurs.

Documents de référence :

"Séquences accidentelles importantes pour la sûreté d'une centrale à eau", A. CARNINO, B. GACHOT, JP. SIGNORET - Rapport DSN 62, Avril 1975.

Titre Banque de données de fiabilité.	Pays : FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Reliability data bank.	Organisme exécuteur : CEA/DSN-SETSSR
	Responsable : MME. A. CARNINO (DSN-SETSSR)
Date de démarrage : 01/01/73 Etat actuel : en cours	Date prévue d'achèvement 31/12/81 Dernière mise à jour : 27/01/77
Scientifiques :	

Objectif général :

L'action a comme objectif général de fournir des données de fiabilité qui représentent la base indispensable de toutes les analyses de sûreté par les méthodes probabilistes. Il s'agit en permanence de collecter des données sur les matériels des centrales où cela est possible et de rechercher les données jugées "crédibles" dans la littérature ou dans des enquêtes.

Objectifs particuliers :

Une action de collecte de pannes et fichier informatisé correspondant marche depuis 1973 sur le réacteur Phénix. Compte tenu du temps cumulé de fonctionnement, des données intéressantes pour la filière pourront être extraites par un travail d'étroite collaboration avec EDF-Production Thermique. Il a été possible de tirer des données d'exploitation de centrales nucléaires françaises (Saint-Laurent 1 + 2), ainsi que d'extraire, de l'expérience de centrales conventionnelles, certaines enquêtes sur des matériels électriques.

Titre Données de fiabilité tirées de l'expérience de fonctionnement d'un réacteur nucléaire.	Pays : FRANCE
	Organisme directeur : CEA/DSN
Titre (anglais) Reliability data from the operating experience of a nuclear reactor.	Organisme exécuteur : CEA/DSN - SETSSR
	Responsable : MME A. CARNINO (DSN - SETSSR)
Date de démarrage : 01/01/73 Etat actuel : en cours terminaison	Date prévue d'achèvement : 01/03/77 Dernière mise à jour : 02/11/76
Scientifiques : MME C. VILLEROUX A. BLIN	

Objectif général :

Détermination de données opérationnelles de fiabilité à partir de l'expérience de fonctionnement du réacteur de Saint-Laurent-des-Eaux.

Objectifs particuliers :

Calculer, par les informations contenues dans le fichier de Saint-Laurent, les taux de défaillances et de réparations des matériels. Confronter ces résultats avec les résultats internationaux.

Etat de l'étude :

1) Avancement à ce jour :

Analyse des informations de Saint-Laurent en 1973 et 1974.
Calcul de données concernant les vannes et pompes de Saint-Laurent en 1975 et 1976.

2) Résultats essentiels :

Rapports sur les taux de défaillance de calculateurs, de thermocouples, les turbosoufflantes, les relais, des pompes et des vannes. Erreur humaine.

Prochaines étapes :

Comparaison des résultats avec des organismes étrangers (NRC, SRS, UKAEA).

.../...

Documents de référence :

"Résultats de fiabilité du fichier d'incidents de la centrale de Saint-Laurent des Eaux". A.BLIN- Rapport SETS 27, octobre 1973.
"Données de fiabilité des vannes de la centrale de Saint-Laurent des Eaux", C.VILLEROUX - Rapport SETS 40, novembre 1975.

Titre Fiabilité des systèmes de sécurité des P.W.R.	Pays : FRANCE
Titre (anglais) Reliability of safeguard systems of P.W.R.	Organisme directeur : CEA
Date de démarrage : 01/04/73 Etat actuel : en cours	Organisme exécuteur : CEA/DSN-SETSSR Responsable : MME A. CARNINO Scientifiques : A. ELIN JP. SIGNORET R. QUENEE
Date prévue d'achèvement : 31/12/81 Dernière mise à jour : 02/11/76	

Objectif général :

Etudier la fiabilité/disponibilité de tous les systèmes de sauvegarde et des systèmes vitaux des centrales P.W.R.

Objectifs particuliers :

Répondre aux besoins de l'analyse de sûreté de Fessenheim et Bugey.
Etudier le comportement de certains systèmes dans le cas d'accidents.

Etat de l'étude :

1) Avancement à ce jour :

Un certain nombre de systèmes de sauvegarde et vitaux des centrales PWR ont été étudiés. (Fessenheim et Bugey).

2) Résultats essentiels :

Les résultats sont consignés dans les rapports cités en référence.

Prochaines étapes :

Analyse d'autres systèmes des P.W.R. et du comportement à long terme après accident de certains systèmes (E.A.S., R.C.V., R.P.R.)

.../...

Documents de référence :

- "Centrale de Fessenheim - Analyse qualitative partielle de la fiabilité du circuit d'alimentation de secours des générateurs de vapeur à l'aide de la méthode des arbres de défaillances", JP.SIGNORET, B.GACHOT, P.VUILLEMIN - Rapport DSN 48, septembre 1974.
- "Analyse qualitative et quantitative de la fiabilité du circuit hydraulique d'injection de sécurité basse pression (ISEP)", JP.SIGNORET - Rapport SETSSR 33.
- "Analyse qualitative et quantitative de la fiabilité du circuit hydraulique d'injection de sécurité haute pression (ISHP)", JP.SIGNORET - Rapport SETSSR 32.
- "Fiabilité du circuit RRI de Fessenheim", A.BLIN - Rapport SETSSR 48, juin 1976.

<u>Classification: 14</u>	
<u>Title 1 (Original Language):</u> System- und Zuverlässigkeitsanalyse an Leichtwasser- reaktoren (ATT 085 A - I.1.8., Jahresbericht A 76)	<u>COUNTRY:</u> BRD
	<u>SPONSOR:</u> BMI
	<u>ORGANIZATION:</u> LRA, Garching
<u>Title 2 (English):</u> System and Reliability Analysis on Light Water Reactors	<u>Project Leader:</u> i.V. Dr. D. Wach Dr. Hörtner Dr. Kafka
<u>Initiated (Date):</u>	<u>Completed (Date):</u>
<u>Status:</u> continuing	<u>Last Updating (Date):</u> December 1976

1. General Aim

The detailed investigation of possible accident sequences, especially of their probability and the implementation of the results in a risk concept.

2. Particular Objectives

- Detailed investigation of typical 1200 MWe-PWR-plants
- Set-up of cause-consequence diagrams in order to find out the most important accident sequences
- Detailed analysis of the engineered safety features installed to cope with the associated accidents and estimation of their reliabilities
- Systematic investigation of important electronic equipments with the aim of finding the failure rates of the relevant failure modes.

3. Experimental Facilities and Program

Not relevant.

4. Project Status

4.1 Progress to Date

For the nuclear power plants considered, the reliability investigation /1/ of the engineered safety features needed to control the design basis accident "Large LOCA" were finished. Very detailed sensitivity studies and discussions of the results were done, including common-mode failures. The reliability analysis of the reactor control rod scram systems for different transients was also completed by a discussion of common-mode failures. A reliability analysis was carried out for the reactor coolant system safety and relief valves needed to cope with the ATWS "Loss of Normal Power".

The experiences gained by the investigations of the engineered safety features control system were summarized.

For different electronic modules investigations were performed in order to find out the various failure modes and to evaluate the associated failure rates.

The computer programs to calculate the reliability data for complex systems were further developed and are described in special reports /2, 10, 12/.

A report with critical comments on the Reactor Safety Study WASH-1400 was finished /6/.

4.2 Essential Results

The results of the studies for the engineered safety features were discussed in papers /5,7,12,17,20,21/. From these results it can be seen that very detailed analyses of the systems are necessary. Only

by this means all the significant contributions to the unavailability can be detected, e.g. contributions through common-mode failures caused by system intermeshings. Examinations of the entire system needed to cope with the considered accident are therefore required.

The results of the reliability investigations for electronic devices were described in /4,8,10/. These reports contained the detailed tabulations of the failure mode and effect analyses performed. The intermediate and final results for the different failure modes were shown.

5. Next Steps

The project on System analysis of the engineered safety features of a German PWR plant has been finished. The experiences gained during this work will now be used for the risk study of a German PWR plant (Project RS 215 and RS 217).

6. Relation with Other Projects

Blowdown and refilling calculations for LOCAs as well as investigations concerning the dynamic behaviour of plant in the case of transient events are done within other sections of the LRA. These investigations are taken as the basis for the reliability analysis.

7. Reference Documents

/1/

H. Hörtner, E. Dressler, E. Nieckau, H. Spindler
Kernkraftwerk Biblis, Block A. Zuverlässigkeitsuntersuchung der für die Beherrschung des Auslegungsstörfalls "Bruch einer kalten Hauptkühlmittel-
leitung" erforderlichen Sicherheitssysteme, Teil I. Vertraulicher Be-
richt

MRR-V-9, November 1975

/2/

E. Dressler, H. Lurz
SAFTL und CRESS. Beschreibung zweier Programmsysteme zur Berechnung
der Zuverlässigkeit von komplexen Systemen. Programmbeschreibung
MRR-P-21, Dezember 1975

/3/

K. Singh-Wadwa
Perspectives on the Risks of Nuclear Power Production - A Literature
Review. Internal Report
MRR-I-62, Januar 1976

/4/

S. Goßner, S. Steindl
Experimentelle Ausfalleffektanalyse an dem Binärverteiler ZL 412 (S&F)
Vertraulicher Bericht
MRR-V-13, Februar 1976

/5/

H. Hörtnner, E. Dressler, E. Nieckau, H. Spindler
Kernkraftwerk Biblis, Block A. Zuverlässigkeitsuntersuchung der für die
Beherrschung des Auslegungsstörfalls "Bruch einer kalten Hauptkühlmittel-
leitung" erforderlichen Sicherheitssysteme, Teil II. Vertraulicher Be-
richt
MRR-V-14, April 1976

/6/

H.P. Balfanz, P. Kafka
Kritischer Bericht zur Reaktorsicherheitsstudie (WASH 1400). Interner
Bericht
MRR-I-65, IRS-I-87, April 1976

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H. Hörtnner, E. Nieckau
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MRR-V-15, Mai 1976

/8/

S. Goßner, S. Steindl

Experimentelle Ausfalleffektanalyse an dem Rechengerät TZA 20 (H&B).

Vertraulicher Bericht

MRR-V-16, Mai 1976

/9/

W. Bastl

CSNI Task Force on Problems of Rare Events in the Reliability Analysis of Nuclear Power Plants

Ispra, June 1976

/10/

S. Goßner

Experimentelle Ausfalleffektanalyse an dem CMR-Meßumformer ETU 120 (H&B)

Vertraulicher Bericht

MRR-V-17, Juni 1976

/11/

R. Daugherty, L. Schlösser

CRESSEX - Beschreibung eines Zuverlässigkeitsrechenprogramms zur Ermittlung wichtiger Kenngrößen von komplexen Systemen. Programmbeschreibung

MRR-P-23, September 1976

/12/

W. Ullrich, W. Frisch

Untersuchungen von Betriebsstörungen bei Versagen der Reaktorschnellabschaltung (ATWS) und anderer ausgewählter Sicherheitseinrichtungen

MRR 163, IRS-W-22, September 1976

/13/

E. Dressler

Theoretische Grundlagen zum Programmsystem SAFTL und CRESS zur Berechnung der Zuverlässigkeit von Systemen

MRR 164, September 1976

/14/

J. v. Linden

Kernkraftwerk Biblis, Block A. Untersuchung der Kühlmittelverluste im Nuklearen Zwischenkühlkreis für den Auslegungsstörfall "Bruch einer Hauptkühlmittelleitung". Vertraulicher Bericht
MRR-V-19, Oktober 1976

/15/

P. Kafka

Reactor Safety Systems

Lecture at IAEA Course, Karlsruhe, 11.11.1976

/16/

E. Nieckau

Schlußfolgerungen aus Zuverlässigkeitsanalysen von Reaktorschutzsystemen
MRR 166, November 1976

/17/

W. Dietlmeier

Kernkraftwerk Biblis, Block A. Fehlerbaumanalyse der Notspeise-Pumpen-Turbine. Vertraulicher Bericht
MRR-V-20, November 1976

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KTG-Fachseminar "Störfallverhalten von Leistungsreaktoren", Jülich,
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MRR-I-77, Interner Bericht, Dezember 1976

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P. Kafka, E. Münch

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KTG-Fachseminar "Störfallverhalten von Leistungsreaktoren", Jülich,
23. - 26.11.76
MRR-I-80, Interner Bericht, Dezember 1976

/20/

H. Hörtner, E. Nieckau, H. Spindler

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leitung"

MRR 168, Dezember 1976

/21/

S. Goßner, J. v. Linden

Kernkraftwerk Biblis. Zuverlässigkeitsuntersuchung des Gesamtsystems zur Reaktorschnellabschaltung. Vertraulicher Bericht

MRR-V-21, Dezember 1976

/22/

Quarterly Reports in the Series IRS-Forschungsberichte

8. Degree of Availability

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

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

<u>Classification: 14</u>	
<u>Title 1 (Original Language):</u> Zuverlässigkeitsbeurteilung für den Sicherheits- einschluß (SE) am Beispiel des Druckwasserreaktors (RS 201 - I.1.8, Jahresbericht A 76)	COUNTRY: BRD
	SPONSOR: BMFT
	ORGANIZATION: Tech.Univ.Munich
<u>Title 2 (English):</u> Reliability Assessment of the Secondary Containment of a PWR	<u>Project Leader:</u> Prof.H. Kupfer
<u>Initiated (Date):</u> June 1, 1976 <u>Status:</u> Continuing	<u>Completed (Date):</u> May 31, 1978 <u>Last Updating (Date):</u> Dec. 1976

1. General Aim

The general aim of the project is the assessment of the structural reliability of a PWR which may be subjected to internal and external load conditions, i.e. hazards. The structural reliability, as defined here, is the probability of survival of the structure within its design life.

2. Particular Objectives

Based on the probability of occurrence of various load conditions and the utilization of stochastic processes the reliability for each particular external load case i.e. earthquake, airplane crash, external pressure wave will be determined on one hand and on the other hand the reliability, given an internal load condition, is to be sought.

Particular emphasis will be given the determination of the joint probability of occurrence of the earthquake together with other failure events. Stochastic models for the prediction of these joint probabilities will be developed. The scale parameter for the combinations of the particular load intensities is the reliability which needs not to be higher than that for each separate load case.

Furthermore the problem of the stochastic treatment of the fracture of shell structures, with reference the probabilities of failure of

those structures, is investigated. Load effects are determined by static and dynamic structural analysis. The concept to be developed during course of this research project will be applied to the containment of the powerplant BIBLIS B.

3. Research Program

Essentially the investigation consists of seven problem areas:

- 3.1 Determination of the various failure events or failure paths of the system which are to be taken into account, considering the originating events inside and outside of the plant. This, of course, with reference to the containment.
- 3.2 Determination of the types of loads to be expected acting on the containment and their characteristic values based on the statistically distributed failure events as a result of 3.1.
- 3.3 Calculation of the relevant load spectra which result from the various types of loads with reference to the reliability assessment.
- 3.4 Evaluation of the materials properties under load effects which are generated by loads as described above. Consideration of fracture mechanical methods for impact loads.
- 3.5 Evaluation of the load effects using stochastic theory of plasticity.
- 3.6 Calculation of the structural failure probabilities under single and combined loading.
- 3.7 Global reliability assessment of the containment.

4. Experimental Facilities, Computer Codes

(With reference to item 3.: Research Program)

- 3.1 Given a LOCA the ZOCO-Program is used to evaluate the load conditions due to the resulting pressure- and temperature increase.
- 3.4 Experimental facilities to be developed during the course of the research will simulate impact load conditions which may have to be expected to act on the full scale structure. Test series are to be carried out using plane and reinforced concrete samples.
- 3.5 For the calculation of the load effects under static and dynamic load action in the linear and non-linear range, the SAP and NONSAP-program systems are used respectively.

5. Progress to Date

(With reference to item 3.: Research Program)

- 3.1 According to the working schedule the evaluation of the internal loads resulting from the LOCA "Large Leak" was analysed as the first case. For this purpose it was assumed that a leak with a cross-sectional area of $> 1000 \text{ cm}^2$ occurring between the main pump of the coolant and the pressure vessel is the worst case. According to WASH 1400 this failure event has an estimated probability of occurrence of $10^{-4}/a$.

Following this major failure event all systems, the reactor protection system, etc. are checked during the failure event sequence with respect to their functioning or non functioning. It is therefore intended to determine for each possible failure sequence the resulting survival probability. For calculating the pressure values with the Program ZOCO the following input data have been evaluated:

- the internal structures which are located in the containment as different numbers of pressure rooms, for determining the influence of these partitions on the maximum pressure, the occurrence time of the maximum pressure and the computational time
- the flow from the primary system into the containment within a short interval during different failure sequences using the code BRUCH-D-05.

- 3.4 The results gained at the conference in Boston (Second International Conference on the Mechanical Behaviour of Materials) have been classified and evaluated.

In the course of the computation of the steelhull a sensitivity analysis has been performed. This calculation shows the influence of the different material properties and the influence of some internal loading conditions on the probability of failure of the containment on the basis of a simple analysis. Furthermore data of acceptance tests of special containment steels have been statistically evaluated.

- 3.5 For calculating the load effects of the steel hull with the finite element program SAP, the whole containment was divided up

into 250 thin shell elements. Two of the largest perforations, the material- and the staff lock have been taken into account.

In order to simulate numerically the impact loading process - which is to be determined as described above under item 3.4 - shell model consisting axi-symmetric elements has been performed. For this purpose the load effects as obtained from the dynamic analysis of the global system are used as boundary conditions.

Since for the time being no reliable values and information concerning the type of distributions and their parameters for loads and strength are available, only estimations of the order of magnitude of the failure probability of the steel hull given a LOCA has been calculated so far. This was done using normal distributions.

6. Results

(With reference to item 3.: Research Program)

- 3.1 A complete numerical analysis of the failure event sequence as described above has not been performed as yet. However for the purpose of performing preliminary stress calculations a maximum value of 4.7 bar has been estimated. As a first estimate, the failure of the emergency cooling system has been determined to be 5.10^{-4} /usage. The treatment of the problem in a more differentiated manner is carried out at the present.
- 3.4 The evaluation of the results reported at the conference in Boston indicates in the case of high rates of loading on unreinforced concrete a decrease of the coefficient of variation of the ultimate strength and shows an increase of the meanvalue of the ultimate strength. The sensitivity analysis shows the strong influence of the statistic dispersion of the strength of the material on the probability of failure. Under certain circumstances the safety intervall, chosen at conventional calculation, has not the only decisive importance.
- 3.5 For the linear elastic solution a stress concentration factor at the boundaries of the perforations of 1.7 has been determined as a preliminary information.

7. Next Steps

(With reference to item 3.: Research Program)

- 3.1 The time dependent pressure- and temperature values are calculated using ZOCO. The frequencies of their occurrence are calculated using failure event sequence analysis.
- 3.4 The material data of containmentsteels will be adapted to probability density functions in a physically justifiable way. Furthermore material data of fracture mechanics will be recorded as far as they influence the probability of failure of the containment.
- 3.5 The calculation of the load effects for the given internal pressure- and temperature values will be continued using linear elastic and non linear analysis respectively. Parallel to that, the pipewhipping problem is modeled by a finite element code. For this purpose the force-time relation will be determined using an approximate energy calculation. Furthermore, applying the time history method the dynamic analysis of the steel hull under earthquake loading will be performed.

8. Relation with Other Projects

-

9. References

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10. Degree of Availability of the Reports

-

<u>Classification: 14</u>	
<u>Title 1 (Original Language):</u> Risikostudie zur Sicherheitsbeurteilung von Kernkraftwerken mit Druckwasserreaktoren für einen Standort in der Bundesrepublik Deutschland (RS 215 u. 217 - I.1.8, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: GRS
<u>Title 2 (English):</u> An Assessment of Risk for Nuclear Power Plants PWR on German Siting Conditions	<u>Project Leader:</u> Prof. Birkhofer Dr. Heuser
<u>Initiated (Date):</u> 1.6.1976 <u>Status:</u> continuing	<u>Completed (Date):</u> 30.9.1978 <u>Last Updating (Date):</u> December 1976

1. General Aim

The principal objective of the study is an overall risk analysis for a nuclear power plant with reference to German siting conditions. As has been done in the American Reactor Safety Study WASH 1400, /1/, a quantitative assessment of risks will be made, which are due to potential accidents in nuclear power plants. Especially the KWU-typed PWR Biblis B is taken as the reference plant for the study.

2. Particular Objectives

Work on the study is divided into two temporally subsequent phases:

Phase A is closely related to the methodic approach and modelling that has been outlined in WASH 1400. Referring to German system design and siting conditions it is one of the aims to conclude some results for risk assessment comparable with those of WASH 1400.

Phase B will be concentrated on a refined analysis of special problems, mainly on weak points obtained as results during Phase A, or respectively on those accident paths having been found as main contributors to overall risk.

3. Research Program

The study will be performed by the Gesellschaft für Reaktorsicherheit (GRS) as the main contractor in cooperation with other coworking organizations and institutions, such as the Gesellschaft für Kernforschung (GfK), the Gesellschaft für Strahlenforschung (GSF) and others. The following table gives a survey on the research program and its main subtasks:

3.1 Coordination

3.2 Accident Initiating Events

3.3 Loss of Coolant Accidents

3.4 Transients

3.5 Component Failure Behavior and Failure Data

3.6 Operating Experience from Nuclear Plants

3.7 Containment and Fission Product Release

3.8 Core Meltdown

3.9 External Events

3.10 Radioactive Sources

3.11 Dispersion and Consequences

3.12 Risk Assessment

4. Experimental Facilities, Computer Codes

Reliability analysis calculations required for system analysis will be performed with the GRS reliability computer programs CRESSEX /2/ and FESIVAR. Calculations that are required for description of meltdown processes and fission product release from core and containment will be performed with the codes BOIL and CORRAL used in WASH 1400.

Especially with respect to non-comparable siting conditions and a higher population density within the Federal Republic of Germany there is a need for developing an own consequence model apart from this one applied in WASH 1400.

5. Progress to Date

After having started in June 1976 work has been initiated nearly on all subtasks of the research program. In the following some comments are

given to various topics.

To 3.3 Loss of Coolant Accidents

First of all the spectrum of different sizes of LOCAs and adjoint requirements for emergency cooling and decay heat removal have been specified according to the requirements of licensing procedure. The corresponding event trees describing various accident sequences have been developed. First calculations of adjoint fault trees for the emergency cooling and decay heat removal systems have been performed.

To 3.4 Transients

A systematic approach for the analysis of transients potentially leading to a core meltdown has been started. Furthermore a detailed event and fault tree analysis starting with the event "Loss of Offsite Electric Power" is underway.

To 3.7 Containment

3.8 Core Meltdown

Referring to the results of the event tree analysis obtained for the large LOCA calculations for core meltdown have been performed by means of the US-program BOIL and for comparison of different modelling using the German program system BILANZ. Subsequently first calculations of long-termed containment pressure response have been done.

To 3.11 Dispersion and Consequences

The consequence model foreseen for the study will be developed by the GfK in cooperation with GSF. After having outlined conception and main features of the model, special work has been done to establish two versions of a meteorological dispersion model (short distance and long distance model) for computing dispersion of radioactivity in terms of concentration in the air and on the ground as a function of time and distance from the reactor.

6. Results

-

7. Next Steps

In addition to the comments given in section 5, brief remarks on some work having been started now or being planned for the next future, will be made in what follows.

To 3.3 Loss of Coolant Accidents

For a more refined development of accident sequence analysis special attention is given to an assessment of uncertainties associated with calculational techniques for emergency cooling.

To 3.5 Component Failure Behavior and Failure Data

A first proof of failure data which are summarized in WASH 1400, App. III, and further completion of this list by data from literature and data that have been derived from own field experience in a conventional plant, /3/ is taken up in cooperation of the GRS with the Technical University of Berlin.

Furtheron investigations on the failure behavior of pressurized components (reactor pressure vessel, steam generator et al.) start with an evaluation of failure statistics available for conventional pressure vessels and steam drums.

To 3.6 Operating Experience

In order to check accident and system analysis of the study an evaluation of operating experience of German nuclear plants especially with respect to PWRs, is planned.

To 3.11 Dispersion and Consequences

In context of the consequence model an evacuation model will be developed by the Institut für Unfallforschung of the Technischer

Überwachungsverein Rheinland. For Phase A of the study influence of evacuation will be handled in a similar way as has been considered in WASH 1400.

8. Relations with other Projects

Besides the methods of probabilistic analysis that have been outlined and applied in WASH 1400 in general, the investigations that are required for the study are performed on the basis of results which are derived from the German Research Program on Reactor Safety which is sponsored by the Federal Ministry of Research and Technology. Hereto special reference is given to the results that have been derived so far for the analysis of core melt accidents.

9. References

/1/ Reactor Safety Study

An Assessment of Accidental Risk in US Commercial Nuclear Plants,
US Nuclear Regulatory Commission WASH 1400
(NUREG 75/014), October 1975

/2/ R. Dougherty, L. Schlösser

CRESSEX-Beschreibung eines Zuverlässigkeitsrechenprogramms zur
Ermittlung wichtiger Kenngrößen von komplexen Systemen, Programm-
beschreibung,
MRR-P-23, September 1976

/3/ P. Hömke, H. Krause

Der Modellfall IRS-RWE zur Ermittlung von Zuverlässigkeitskenn-
größen im praktischen Betrieb,
IRS-W-16, November 1975

<u>Classification:</u> 14	
<u>Title 1 (Original Language):</u> Unsicherheiten der Ausfallraten von Komponenten und der Eintrittswahrscheinlichkeit störfallauslösender Ereignisse RS 228, Jahresbericht A 1976	<u>COUNTRY:</u> BRD
	<u>SPONSOR:</u> BMFT
	<u>ORGANIZATION:</u> IKT TU Berlin
<u>Title 2 (English):</u> Uncertainty of the Failure Rates of Components and the Probability of Occurrences of Failure Causing Events	<u>Project Leader:</u> Prof. Dr. Memmert
<u>Initiated (Date):</u> Aug. 1, 1976 <u>Status:</u> continuing	<u>Completed (Date):</u> July 31, 1978 <u>Last Updating (Date):</u> Dec. 31, 1976

1. General Aim

Determination of the Uncertainty of the Failure Rates of Components and the Probability of Occurrences of Failure Causing Events.

2. Particular Objectives

The reliability data for individual components necessary for the system analysis, particularly failure rates respectively lifetimes as well as their probability distributions have hitherto not been available with sufficient exactitude and specification. Therefore it is necessary to compile the generally accessible references and to compare them with one another. In doing this, it is important to give special consideration to data from German nuclear power plants.

Furthermore it seems reasonable to make the final decision on which type of distribution function is to be used on the basis of actual fault-trees.

3. Research Program

- 3.1 Compilation of reliability data for components
- 3.2 Investigation of mean values and distributions of the reliability data
- 3.3 Comparison of the values with those quoted in WASH 1400 /1/
- 3.4 Critical assessment of the data
- 3.5 Comparison of different distribution functions for describing the data

4. Experimental Facilities, Computer Codes

- ad 3.2 Development of a computer code to calculate the necessary statistical parameters of different distributions for the data which are available

5. Progress to date

- ad 3.1 Compilation and examination of already known literature containing reliability data. This has essentially been finished. The addition of more literature to the collection is possible during the whole period of investigation.
- ad 3.3 Characteristic values of the logarithmic normal distribution of failure rates quoted in WASH 1400 /1/ were calculated and compared with the statements made there.

6. Results

- ad 3.1 Up to now there is no information available about reliability data of components in nuclear power plants from the Federal Republic of Germany. The only systematic data collection was carried out for a conventional 300 MWe power plant /3/. However, these data should be considered in any case, because they represent at the moment the only relevant German data material. The further procedure will be based on the data quoted in WASH 1400 /1/ to which values from /2/ and /3/ will be added.
- ad 3.3 The calculated mean values and distributions largely agree with those given in WASH 1400 /1/. Differing results still require a detailed analysis.

7. Next Steps

- ad 3.1 Additional data collections should be considered
- ad 3.2 Determination of mean values and distributions according to certain aspects for a list of defined components.
- ad 3.3 Analysis of the deviation of the mean values and distributions from the data in WASH 1400 /1/.

8. Relation with other Projects

-

9. References

- /1/ Reactor Safety Study,
An Assessment of Accidental Risk in US Commercial Nuclear Power
Plants, US Nuclear Regulatory Commission, WASH 1400 (NUREG 75/014),
October 1975
- /2/ Nuclear Plant Reliability Data System,
1975 Annual Reports of System and Component Reliability,
Southwest Research Institute, San Antonio, Texas 78 284
- /3/ Zusammenstellung von Ausfallraten im Kraftwerk Neurath-RWE,
Interne Mitteilung des IRS vom 15. Oktober 1976

10. Degree of Availability of the Reports

-

<p><u>Title 1 (Original language)</u> Sviluppo di una procedura avanzata per l'analisi di sicurezza di un PWR seguendo l'approccio probabilistico.</p>	<p><u>Classification</u> 14</p>
<p><u>Title 2 (English)</u> Development of an advanced procedure for the PWR safety analysis following the probabilist approach</p>	<p><u>Country</u> ITALY <u>Sponsor</u> FIAT-T.T.G. <u>Organisation</u> Nuclear Energy Division</p>
<p><u>Date initiated</u> June 1976 <u>Date completed</u> 1980 <u>Last updating</u> April 1977</p>	<p><u>Project Leader</u> G.P. Pozzi</p>

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

4. Project status

For each accident to be analyzed, a detailed "event tree" has been developed in order to establish: a) the different path the accident can follow; b) the features of the digital programs necessary to analyze the consequences of the accidents.
An analysis of the various reliability programs available has been performed. With the available programs a probabilistic analysis of the LOCA in the CLEOPATRA loop was made.

5. Next steps

a) The digital programs necessary for the analysis of each accident path will be set up (deterministic programs)

<u>Title 1 (Original language)</u>	<u>Classification</u>
Sviluppo di una procedura avanzata per l'analisi di sicurezza di un PWR seguendo l'approccio probabilistico.	14

- b) The results of the deterministic programs will be coupled with the results of the probabilistic programs.
- c) The complete procedure will be used for the design of the PWR engineered safety features.

6. Relation to other projects

The digital programs for the LOCA analysis described in the other project ("Development of a chain of digital programs for the LOCA analysis of a PWR) will be utilized.

7. Reference documents

Not yet prepared

8. Degree of availability

To a limited extent

9. Budget, personnel involved

8 engineers for 5 years.

<u>Title 1 (Original language)</u> Raccolta dati di affidabilità	<u>Classification</u> 14
<u>Title 2 (English)</u> Reliability data collection	<u>Country</u> ITALY <u>Sponsor</u> CNEN <u>Organisation</u> CNEN
<u>Date initiated</u> 1974 <u>Date completed</u> In progress <u>Last updating</u> April 1977	<u>Project Leader</u> G. Tomassetti

General aim

Studies on reliability data banks through experimental and methodological work on data collection field.

Particular objectives

Analysis of reliability data transferability from one plant to other ones in different general contexts.

Experimental facilities

CNEN experimental and conventional plants

Project status; next steps

Data from a sodium loop were elaborated; data collection from Diesel groups and nuclear facilities is under progress.

Reference documents

"Problems of experience data collection and use"
II Reliability Data Bank Seminar, Stockholm, March 1977.

Degree of availability

Open

<u>Title 1 (Original language)</u>	<u>Classification</u> 14
<u>Title 2 (English)</u> Interactive Synthesis of Fault Trees for Safety Assessment of Nuclear Power Plants	<u>Country</u> ITALY <u>Sponsor</u> CNR-EURATOM <u>Organisation</u> CESNEF-Politecnico Milano
<u>Date initiated</u> 1976 <u>Date completed</u> 1978 <u>Last updating</u> April 1977	<u>Project Leader</u> S. GARRIBBA

DESCRIPTION

Each component from the system is the domain of local functions which express, for all normal and abnormal operating conditions, the flow variables (i.e. the variables characterizing transfers of fluid, load, heat, electrical current, etc.) or which express the constitutive variables (i.e. the variables describing rupture, damage, blockage, etc.). The consideration of a number of additional virtual domains also allows to make explicit external influences (e.g. human operators, maintenance, fires, missiles, etc.) in terms of variables. The local functions are interrelated through transfer matrices. Localized transfer matrices are pieced together to form the Approximate Descriptive Model (ADM) of the system. Then, steps towards the generation of an isomorphic fault tree are : i) Definition of the top event. ii) Numbering process. Numbers are assigned to certain points of flow sheets in a certain canonical manner. iii) Translation process. iv) Free reduction process. These steps will be implemented by computer program I-SYNFT (Interactive Synthesis of Fault Tree).

N.V. KEMA

Arnhem, 25 februari 1976

Classification 14	
<u>Title 1</u> Faalkans-analyse met behulp van gebeurtenissen- en foutenbomen	<u>Country</u> The Netherlands <u>Organization</u> KEMA
<u>Title 2</u> Failure analysis by application of event- and fault trees	<u>Projectleader</u> R.W. van Otterloo

1. General aim

To analyse new systems or changes in existing systems. And by application of this technique to make decisions concerning these new systems or changes in existing systems in the right way.

2. Particular objectives

- Analysis of the unavailability of reactor safety systems and of the unreliability of reactor systems.
- Analysis of the probability of different groups of radioactive releases of a nuclear power reactor.

3. Experimental facilities

Not applicable.

4. Project status

Methods and computer codes have been compared.

5. Next steps

Not applicable.

6. Relation with other projects

This project was started to do the "Risk analysis of the fuel cycle in the Netherlands" (RASIN-study) which was finished in June 1975.

7. Reference documents

See 6.

Several applications of this method are written in the Dutch and the English language.

8. Degree of availability

Through the organization KEMA.

NETHERLANDS ENERGY RESEARCH FOUNDATION (ECN)		CLASSIFICATION: 14
TITLE: Faalweg en faalkansvoorspellingen van reaktorsystemen		COUNTRY: NETHERLANDS. SPONSOR: Ministry of Social Affairs ORGANIZATION: ECN
TITLE: (ENGLISH LANGUAGE): Failure mode and failure rate prediction of reactor systems		PROJECTLEADER: K. Terpstra
INITIATED: June 1974	LAST UPDATING: March 1977	
STATUS: Progressing	COMPLETED: Summer 1978	
SCIENTISTS:		

1. General aim

To predict the probability of failure of systems on basis of possible failure modes and component reliability, taking into account accidental as well as systematic failures due to external causes or inherent faults of components.

2. Particular objectives

The first stage of the program consists of:

- making an inventory of the general available methods of reliability analysis and "data banks" on failure rates
- establishing a standard procedure to collect data on failures in nuclear power stations in The Netherlands
- developing procedures to evaluate the reliability of reactor systems for specific cases applicable to Dutch power reactors.

3. Experimental facilities: None

4. Project status:

Updating theoretical knowledge and evaluation and updating of existing computer codes

5. Next steps

The next step is to create a balanced code to do the following:

- a. Constructing the logical fault tree
- b. To determine the MCS
- c. To calculate the availability of components and system
- d. To rank components and MCS according to their importance with respect to the stop event in a certain measure.

After creating the code, a test run has to be made of a reactor safety device.

6. Relations with other projects: -

7. Reference documents: -

8. Degree of availability

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

9. Budget: -

10. Personnel: 0.9 manyear

PROJECT TITLE : Safety evaluation of a Swedish nuclear power plant and analysis of environmental consequences of an accident	CSNI INDEX NUMBER : 14
SPONSORING COUNTRY : Sweden	ORGANISATION : AB Atomenergi
DATE INITIATED : 1976-12-23 DATE COMPLETED : 1977-12-31	PROJECT LEADER : Olov Norinder

Description :

The objective is to evaluate the accident risks and the environmental consequences of an accident in a Swedish BWR plant in relation to the Rasmussen Reactor safety study WASH 1400 and the Swedish SOU report 1974:56 about urban siting of nuclear power plants.

PROJECT TITLE : Probabilistic Accidental Transients Analysis	CSNI INDEX NUMBER : LWR 14
SPONSORING COUNTRY : Commission of the European Communities	ORGANISATION : JRC Ispra
DATE INITIATED : 1974 DATE COMPLETED : 1977	PROJECT LEADER : G. Volta/J. Amesz

Description :

1. General aim

To assess the probability distribution of loads acting on the reactor primary circuit.

2. Methodology and application

The presented method consists in a combined probabilistic-deterministic approach. This approach is probabilistic in the sense that for each combination of system stated a probability figure is determined by means of a fault-tree analysis. It is deterministic in the sense that the corresponding consequence (e.g. magnitude of overpressure or overtemperature) for each combination of states is computed by deterministic codes.

The approach has already been applied to PWR overpressures in the frame of the collaboration contract no. III-73-PIPGF CCR Euratom-Framatome.

The same approach will be applied to the temperature transients at a defined point of the primary circuit of the fast reactor PEC in the frame of the collaboration contract no. 144-75 PIPG/RN 163 CCR Euratom- CNEN.

3. Project status

3.1 Progress to date

A first conclusion of the work performed in the frame of a collaboration contract with CNEN (n° 164-75 PIPG/RN 163) has been presented at the International Meeting on Fast Reactor Safety at Chicago [1].

The reliability analysis of the electrical supply system has been completed assuming the various accidents as mutually exclusive events [2].

For this the computer code ELISA has been developed. The code accepts a fault-tree with NOT gates [3].

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Telex : 630668 AEN NEA - Tel : 524 82-00

3.2 Essential results

See reference documents.

4. Reference documents

- [1] J. Amesz, G. Volta, F. Cesari, R. Righini,
"Probabilistic Analysis of Accidental Transients in a
LMFBR" presented at FRS Meeting at Chicago (USA)
5-8 Oct. 1976.
- [2] J. Amesz, F. Lomazzi,
"Analisi di affidabilità del sistema di alimentazione
elettrica sul reattore PEC"
Nota tecnica P.E.R. 126/77.
- [3] M. Astolfi,
"ELISA, Codice per il calcolo dell'affidabilità di
sistemi. Descrizione e modo d'uso".
Nota tecnica P.E.R. 098/76.

15. INTERRELATION BETWEEN REACTOR PLANT AND
OPERATION PERSONNEL

Classification 15

<u>Title 1</u> Operatør studier	COUNTRY Denmark
	SPONSOR Research Establishment, Risø
	ORGANIZATION Research Establishment, Risø
<u>Title 2</u> Study of the Process Operator	<u>Project leader:</u> L.P. Goodstein
<u>Initiated:</u> Approx 1966 - in its present form 1973 <u>Status:</u> progressing	<u>Completed:</u> <u>Last updating:</u> 1976
	<u>Scientists:</u> L.P. Goodstein J. Rasmussen

1. General aim

2. Particular objectives

To study the process operator - his work situation and procedures - together with methods for supporting him, especially in abnormal situations.

3. Experimental facilities and programme

- Hybrid computer
- Interactive CRT graphics terminal

4. Description and status

Work to date has been centered on the study of human behaviour in process plants as well as on various trouble-shooting tasks - in order to establish ways and means of providing support - especially in critical situations - but also in order to ultimately be in a position to include the effects of human actions/errors in systematic reliability and safety analyses.

In this work, we utilize tape recordings from power stations and other work situations as an aid in formulating ideas regarding the data, procedures and models utilized by the human in the various

tasks. These can then be used in the design of the man-machine interface to provide improved aid to the operator. In addition, we expect to continue our studies of human error - either from published case stories or from simulator experiments.

5. Next steps

- A program to continue with the testing and validation of these models of the operator is being planned. Improved methods for the evaluation of risks which include human error will be evaluated.

6. Relation with other projects

A cooperative Scandinavian program is planned to the 1977-1980 period.

7. Reference Documents

J. Rasmussen 1969

Man-Machine Communication in the Light of Accident Records
July, 1969, S-1-69

Reprinted from IEEE-GMS, ERS International Symposium on Man-Machine Systems, Cambridge, 1969.

J. Rasmussen 1973

The Role of the Man-Machine Interface in Systems Reliability
November 1973, Risø Report R-10-73

Reprinted from NATO Conference on Generic Techniques in Systems Reliability Assessment. Liverpool, July 1973

J. Rasmussen 1974

The Human Data Processor as a System Component

Bits and Pieces of a Model, June 1974, Risø Report R-8-74

J. Rasmussen 1976

Outlines of a Hybrid Model of the Process Plant Operator
Risø Note N-7-76

Reprinted from NATO Conference on Monitoring Behaviour & Supervisory Control, Berchtesgaden, March 1976.

J. Rasmussen & J.R. Taylor

Notes on Human Factors Problems in Process Plant Reliability
& Safety Prediction - Risø-M-1894 1976.

8. Available from the library at Risø.

Titre Paramètres humains de la sûreté.	Pays : FRANCE
Titre (anglais) Human factors in nuclear safety.	Organisme directeur : CEA/DSN
Date de démarrage : 01/01/76 Date prévue d'achèvement : 31/12/81 Etat actuel : en cours Dernière mise à jour : 23/12/76	Organisme exécuteur : CEA/DSN - SETSSR Responsable : MME A. CARNINO (DSN - SETSSR) Scientifiques :

Objectif général :

Lorsqu'on étudie la causalité élémentaire des accidents ou incidents nucléaires on constate que les composants d'origine humaine sont majoritaires, il était donc capital de s'efforcer d'étudier scientifiquement l'organisation humaine, 3ème composante d'une installation nucléaire.

Objectifs particuliers :

- 3 objectifs ont été déterminés :
- 1) Analyse des organisations humaines autour des installations nucléaires;
 - 2) Etudes ergonomiques particulières influant directement sur la sûreté nucléaire (interface homme/pupitre de commande, entretien en milieu radioactif) ;
 - 3) Essai de mesure de fiabilité humaine dans l'industrie nucléaire.

Etat de l'étude :

- 1) Avancement à ce jour :

Deux études d'organisation ont été lancées en 1976 : près d'un réacteur ; près d'une installation de chimie radioactive.

Prochaines étapes :

Deux autres installations vont être prochainement étudiées.

Relation avec d'autres études :

Ces études sont en relation :

Avec les études entreprises sur la part nucléaire au département de protection;

Avec les travaux de fiabilité entrepris par le groupe fiabilité au DSN.

15.1. Behaviour of personnel (under normal and accident conditions)

15.2. Training of personnel

PROJECT TITLE : Planning of a training simulator for the Loviisa PWR nuclear power plant	CSNI INDEX NUMBER : 15.2
SPONSORING COUNTRY : Finland	ORGANISATION : Technical Research Centre of Finland
DATE INITIATED : 1972 DATE COMPLETED : continues	PROJECT LEADER : Björn Wahlström

Description :

The project aims to the construction of a training simulator for a 440 MW PWR nuclear power plant of the Loviisa type.

<u>Classification: 15.2</u>	
<u>Title 1 (Original Language):</u> Entwicklung und Aufbau eines Ausbildungssystems im Medienverbund zur Intensivierung der Schulung und Er-tüchtigung von Betriebspersonal von Kernkraftwerken (RS 152 - II.5.2, Jahresbericht A 76)	COUNTRY: BRD <hr/> SPONSOR: BMFT <hr/> ORGANIZATION: KWU, Frankfurt
<u>Title 2 (English):</u> Development of a Training System for the Staff of Reactor Plants	<u>Project Leader:</u> H. Martin
<u>Initiated (Date):</u> 1.1.75 <u>Status:</u> Continuing	<u>Completed (Date):</u> 31. 3. 77 <u>Last Updating (Date):</u> 31. 12. 76

1. General Aim and 2. Particular Objectives

Information systems have been investigated which were suitable for training the staff of operating reactors. Criteria for the choice of an optimal schooling program were to be developed, considering several information carriers and media. An example of a training program had to be produced and tested.

3. Research Program

Work was concentrated on the following program:

- a) Investigation of the applicability of various information systems.
- b) Development of a suitable combination of several media.
- c) Production criteria for schooling programs.
- d) Build-up of training programs for the staff.
- e) Production of special training examples.
- f) Documentation.

4. Experimental Facilities

No experimental facilities necessary.

5. Progress to Date

Comparative tests with diapositive-projectors with higher power were performed. Tests were started with super-8 and 16 mm films. The quality of the pictures was compared.

Some production examples were created for group schooling purposes. The time sequences were optimized.

An information program for the staff of power plants was developed.

The special training (detail work on systems and functions of a special power station) was considered during the start-up phase on the base of one day per week.

6. Results

The diapositive projectors were not well suited for special purposes. The tests with super-8 and 16 mm films showed that the super-8-film needed higher effort for illumination and had a lower quality of the pictures compared with the 16 mm film.

A trick-technique for the presentation of components by successive build-up of single parts has shown good results. This methods allowed the arrangement of components in the right series.

The time criteria for single szenes and sequences were checked. Real-szeneries should not be longer than 5 min. Theoretical presentations and trick sequences should follow, but not longer than 50 - 60 sec.

The training program will consist of 4 - 5 phases. The basic course has to procure fundamental knowledge. The second phase gives basic informations of the special power station type and the systems. The third phase gives details of special systems and functions. The fourth phase offers simulator-training of dynamic processes, malfunctions and normal operation. The fifth phase is the so-called "on the job"-training when the staff gets the needed knowledge during active work on the station.

7. Next Steps

Corrective discussions with utilities about details of the suggested training program will be completed.

8. Relation with Other Projects

9. References

10. Degree of Availability

16. ENVIRONMENTAL PROTECTION
(no entries)

16.1. Preparation for emergencies
(no entries)

16.2. Emergency equipment
(no entries)

17. NUCLEAR ACCIDENT RECOVERY AND DECOMMISSIONING
(no entries)

17.1. Decontamination

<u>Classification: 17.1</u>	
<u>Title 1 (Original Language):</u> Entwicklung von Dekontaminationsverfahren (PNS 4411 - I.2.4, Jahresbericht A76)	COUNTRY: BRD
	SPONSOR: BMFT
	ORGANIZATION: GfK, Karlsruhe
<u>Title 2 (English):</u> Development of Decontamination Methods	<u>Project Leader:</u> Dr. Dippel Dr. Hentschel DI Kunze
<u>Initiated (Date):</u> Jan 1973 <u>Status:</u> Continuing	<u>Completed (Date):</u> Dec 1977 <u>Last Updating (Date):</u> December 1976

1. General Aim

Decontamination processes for the decontamination of metal surfaces are optimized and new processes aimed at high decontamination efficiency and a minimum production of decontamination wastes. The processes are intended to be applied in a facility for conditioning nuclear components after incidents and for final disposal. The chemical agents used are compatible with the treatment and solidification processes for the radioactive liquids. In case selected chemicals can undergo special treatment to avoid disturbance in the before mentioned processes.

2. Particular objectives

Particular objectives are the reduction of waste volumes containing big loads of chemicals, compatibility of the decontamination agents with waste treatment and solidification processes, and the application of the decontamination processes, by remote working techniques.

3. Research Program

The essential experiments are the preparation of contaminated standard samples, variation of decontamination paste composition, study of the paste effectiveness in relation to composition and

of the behavior of their components on waste treatment and solidification and estimation of waste production rates in comparison to liquid decontamination chemicals followed by the development of a application technique in thin layers for the paste.

Basically the same experiments are carried out with molten salts. Optimization experiments are to be started for better cleansing effectiveness with a liquid cleanser and decontamination chemicals.

4. Experimental Facilities

No special experimental facilities are necessary. Standard laboratory equipment is used for the experiments.

5. Progress to Date

5.1 Development of cleansing pastes and application of molten salts for the decontamination of metal surfaces

Cleansing pastes have been prepared by mixing combinations of HF, HNO₃ and HCl with a HF resistant, highly dispersed baryte type material. The effectiveness of these pastes has been proved with samples taken from power stations.

KHPO₄ and sodium peroxide powder have been applied as decontaminants by coating metal surfaces to be decontaminated. Both melts are very effective, as could be shown by decontaminating power station samples.

The development of technical equipment for the application of pastes and molten salt to surfaces in thin layers is finished and the components in question are ordered.

5.2 Optimization of a liquid cleanser

Base material for a liquid cleanser have been selected by DT analysis with respect to their thermal stability. With the most stable types cleansers have been mixed with foam regulating agents. To this basic mixtures complexing agents have been added

and the decontamination effectiveness was tested.

5.3 Optimization of the Decontamination of Components

A selection of chemicals currently used for decontamination and applicated in dipping techniques have been tested for their effectiveness with respect to working temperature and concentration. Further experiments are on the way. Out of others HNO_3 , HNO_3/HF , alkaline KMO_4 solution, citric acid and oxalic acid are tested.

6. Results

6.1 Development of cleansing pastes and application of molten salts for decontamination of metal surfaces

Baryte material based pastes show in all experiments and application tests the better effectiveness than pastes on polyethelene-titanium-oxide-mixtures. Pastes without hydrochloric acid, have a similar effectiveness than those containing hydrochloric acid, but the decontamination time is longer, about twice as much. Because of the absence of the polyethelene powder the thermal stability of the paste is higher and difficulties in the transportation of the waste waters caused by this powder are eliminated.

It could be demonstrated, that KHPO_4 as decontamination agent is very efficient in decontaminating primary circuit samples from PWRs. Activities of 0.5 to 1.5 $\mu\text{Ci}/\text{cm}^2$ prior to decontamination are removed up to between 1% and 5% residual activity.

Regarding the technical equipment for coating contaminated surfaces it has been demonstrated that pastes can be applied in thin layers. Only current type technical components are necessary. Compatibility problems are solved by selection of special surface protection coatings for all those parts which come in contact with the chemical agents.

6.2 Optimization of a Liquid Cleanser

The above mentioned mixtures show a definitely better thermal stability in comparison to currently used liquid decontamination

cleansers. Up to now the decontamination results compared with those of the Papan-Dekopan cleanser are better with respect to Ru-106, practically the same with respect to Cs-137, and poorer with respect to Co-60.

6.3 Optimization of the Decontamination of Components

Experimental results show that decontamination results are better at a working temperature of 80°C. With respect to effectiveness and amount of waste generated 4 m nitric acid is the best choice. HNO₃/HF 4 concentration can be reduced by a factor of 2, leaving the effectiveness unchanged. The same result is found for the concentration of alkaline KMnO₄ solution; the removal of Ru-106 with it is excellent. Regarding organic acids, the 1% oxalic acid and the 0.2 m citric acid/0.3 m oxalic acid mixture are the most effective.

7. Next Steps

7.1 Development of cleansing pastes and application of molten salts for decontamination of metal surfaces

After a technical demonstration of the coating equipment these developments are finished.

7.2 Optimization of a Liquid Cleanser

In case a composition of the cleanser can be found the effectiveness of which is comparable with that of Papan-Dekopan for the isotopes selected, the range of application of the cleanser will be extended to glass and plastic surfaces.

7.3 Optimization of the Decontamination of Components

The experiments mentioned are continued aimed at optimal concentration with respect to decontamination effectiveness. In this the use of other application techniques is included.

8. Relation with Other Projects

There is a cooperation with KAH company in the development of decontamination agent application equipment.

9. References

Th. Dippel, D. Hentschel, S. Kunze
Decontamination and Decontamination Wastes
Kerntechnik, 12, 256-531, 1976

10. Degree of Availability of the Reports

The report is available at the authors.

17.2. Removal of accident consequences
(no entries)

17.3. Decommissioning

<u>Classification: 17.3</u>	
<u>Title 1 (Original Language):</u> Entwicklung von Methoden und Verfahren zur Stilllegung und Endbeseitigung nuklearer Anlagen (PNS 4421 - I.2,4, Jahresbericht A 76)	<u>COUNTRY:</u> BRD
	<u>SPONSOR:</u>
	<u>ORGANIZATION:</u>
<u>Title 2 (English):</u> Development of Methods and Techniques for the Decommissioning and Ultimate Disposal of Nuclear Facilities	<u>Project Leader:</u> G.W. Köhler
<u>Initiated (Date):</u> Jan. 1974	<u>Completed (Date):</u> 1978
<u>Status:</u> Continuing	<u>Last Updating (Date):</u> December 31, 1976

1. General Aims

Development of methods and techniques for the decommissioning and ultimate disposal of nuclear facilities.

2. Individual Tasks

2.1 Studies on the conditioning of large reactor components.

2.2 Establishment of a general analysis for the safe decommissioning and ultimate disposal of nuclear power stations by evaluation of results obtained at home and abroad.

3. Program of Investigations

Conditioning:

3.1 Means of dismounting large reactor components from the reactor building.

3.2 Supplementing and completing drawings for conditioning facilities.

Possible sites for a conditioning facility on the reactor premises.

3.3 Investigation of thermal separation methods for application in nuclear facilities. Selection of mechanical separation tools.

General Analysis:

- 3.4 Dismounting, crushing - steel/concrete
 - Conditioning of large components
 - Auxiliary systems
- 3.5 Means of reactor component transportation from nuclear power station
- 3.6 Waste conditioning
- 3.7 Setup of time schedule for decommissioning
- 3.8 Special arrangement of reactor components
- 3.9 New reactor lines

4. Test Facilities, Computer Programs

5. Work Performed

Conditioning:

The investigations relating to 3.1 through 3.3 were performed and terminated within the period of reporting.

General Analysis:

The studies relating to 3.4 and 3.5 were performed and terminated.

Points 3.6 through 3.9 were treated although not terminated within the period of reporting.

6. Results

Conditioning Facility:

Ad 3.1 Dismounting large reactor components is feasible technically, even after an extended period of operation.

Except for the pressure vessel of new boiling water reactors, the components can be removed as integral units.

Dismounting unshielded components emitting a high dose rate is very expensive since it has to be done by remote handling and remote inspection.

Ad 3.2 A conditioning facility can be installed on the power station site of new light water reactors. Existing siding tracks can generally be used for transports from and to the facility.

Ad 3.3 Considering the sizes of components and the material thickness, thermal separation techniques should be preferably used in dismantling, above all plasma fusion cutting and autogeneous flame cutting (also combined).

For component dismounting and fragmentation of segments mechanical tools and machine tools, respectively, such as saws and hammer shears, are suitable means.

General Analysis:

Ad 3.4 Dismounting of components in the reactor building is feasible technically, but only reasonable in case of decommissioning.

If a component has to be replaced, external dismantling, e.g. in a conditioning facility, should be preferred so that the installation of the replaced components is not delayed.

Auxiliary systems such as a protective tent, hood, filter, etc. are necessary in both cases in order to avoid propagation of contamination.

Ad 3.5 Due to the high dose rate to be expected after longterm operation (20 - 30 years) the primary components must be shielded during transport. This increases the weight and sizes so that only medium-sized components (e.G. primary pump) can be transported as integral units.

7. Plans for Future Work

Completion of points 3.6 through 3.9.

8. Relations to Other Tasks

9. Literature

G.W. Köhler, J. Weppner, M. Salaske, W. Hennhöfer, Studienentwürfe für eine Anlage zur Konditionierung schwerer aktiver Reaktorkomponenten, Trenn- und Zerlegemethoden sowie ferngesteuerte Arbeitsgeräte.
July 1976 (not published)

J. Weppner, G.W. Köhler,
Die Konditionierung von Reaktor-Großkomponenten
- Ein Überblick -
December 1976 (not published)

10. Availability of Reports

<u>Classification:</u> 17.3	
<u>Title 1 (Original Language):</u> Quantitative Mengenstromanalyse für radioaktive Abfälle (RS 220 - II.6.1, Jahresbericht A 76)	COUNTRY: BRD SPONSOR: BMFT ORGANIZATION: NIS - Hanau
<u>Title 2 (English):</u> Quantitative Analysis of Radioactive Waste	<u>Project Leader:</u> A. Gasch
<u>Initiated (Date):</u> 1. 10. 1976 <u>Status:</u> Continuing	<u>Completed (Date):</u> 30. 6. 1977 <u>Last Updating (Date):</u> 31.12.1976

1. General Aim

The radioactive waste from dismantling of nuclear power plants causes several problems. In order to solve these problems, it is necessary to analyse the radioactive waste. The results will allow to specify the requirements for handling, transport or burrial of the waste.

2. Particular Objectives

Within the framework of an overall project in the view of conditioning heavy reactor-components from the dismantling of nuclear power plants, this study should deliver the necessary data about the quantity and quality of the waste. A longterm summary analysis will be carried out for the BRD. The waste shall be classified considering different points of views. Waste quantities as function of time are to be determined.

3. Research Program

- 3.1 Establishment of a system of classifying criterias for the waste from dismantling nuclear power plants
- 3.2 Determination of quantity of bulky activated waste
- 3.3 Interpretation of operating experience in nuclear power plants
- 3.4 Determination of quantity of bulky contaminated waste
- 3.5 Decontamination of bulky waste from 3.4
- 3.6 Secondary waste from the dismantling of nuclear power plants
- 3.7 Rad waste quantity after LOCA

4. Experimental Facilities, Computer Codes

- re 3.2 The NIS Computer Code AKAT II calculates the activation of reactor components resulting from thermal, epithermal, and fast neutron flux. Radioisotopes produced by various reactions are added by the machine.

Only the following input data are necessary: the activatable reactor components with specific neutron flux, the materials of reactor components with their composition and their exposition time. The nuclidespecific data of the activatable materials and produced radioisotopes are stored in a library (file).

5. Progress to Date

- re 3.1 A system of classifying criterias has been created. Several classifying systems had been tested with respect to applicability for classification of dismantling waste. The classification of dismantling rad waste follows the recommendations of the IAEA.

- re 3.2 The base for calculation of activities has been formed
- re 3.4 Computations of the contaminated bulky waste quantities have been started

6. Results

- re 3.1 The dismantling waste must be handled, transported and buried. Conforming with the resulting necessity the waste will be classified corresponding to their physical conditions. Categories within this classification follow specific activities which is opposite to the recommendations of the IAEA. Furthermore chemical aggression and toxicity are considered.
- re 3.2 Definit results are not available
- re 3.4 Definit results are not available

7. Next Steps

- re 3.2 The work will be finished by April
- re 3.3 The interpretation of operating experiences in nuclear power plants is planned for first quarter of 1977
- re 3.4 The work will be finished by April
- re 3.5 The work will start later on
- re 3.6 The work will start later on
- re 3.7 The work will start later on

8. Relation to other Projects

9. References

IAEA "Standardization of Radioactive Waste Categories"
Technical Reports Series No. 101, 1970

BMI-Entwurf vom 16. 11. 1976

"Verordnung über den Schutz von Schäden durch ionisierende
Strahlen" (Strahlenschutzverordnung)

Staatsanzeiger für das Land Hessen

"Bedingungen über die Lagerung von schwachaktiven Abfällen
im Salzbergwerk Asse"

Nr. 18, Seite 202

H. Ramdohr, Kerntechnik 11. Jahrgang 1969, Nr. 5

10. Availability of Reports

PROJECT TITLE : NUCLEAR POWER PLANT DECOMMISSIONING	CSNI INDEX NUMBER : 17.3 - Water reactors
SPONSORING COUNTRY : Switzerland	ORGANISATION : Nordostschweizerische Kraftwerke AG, Baden + EIR
DATE INITIATED : April 1976 DATE COMPLETED : Summer 1978	PROJECT LEADER : Nordostschweizerische Kraftwerke AG, Baden

Description :

Project study for the decommissioning of an existing nuclear power plant (PWR, 2 x 350 MWe net) at the end of its useful life.

Major points of this study are:

- Determination of the degree of activation of the reactor pressure vessel and its internals
- Disposal and storage of activated/contaminated primary circuit components
- Disposal of concrete structures
- Cost estimates

18. FUEL CYCLE (e.g. fuel production, fuel and waste transport, reprocessing)

PROJECT TITLE : COATING AND EMBEDDING OF HIGH LEVEL RADIOACTIVE WASTE PARTICLES	CSNI INDEX NUMBER : 18
SPONSORING COUNTRY : Austria	ORGANISATION : ÖSGAE
DATE INITIATED : January 1976 DATE COMPLETED : Continuing	PROJECT LEADER : W. Neumann

Description :

The aim of this research program is to contribute to further developments in the field of solidification and conditioning of high level waste from fuel reprocessing plants.

Granules of simulated, calcined HLW are coated by chemical vapour deposition with acetylene derived pyrocarbon in a lab-scale fluidized bed at temperatures below 1000°C. The product properties (mechanical and chemical stability, coating thickness and density, etc.) are continuously determined. The obtained results show a significant increase in mechanical strength and leach resistance.

To avoid the dispersion of the coated granules and to improve the emission of the radiation induced heat embedding studies are performed. A special pressure casting technic makes it possible to fill the spaces between the granules and to cast simultaneously a protective shield. The embedding material is a corrosion-resistant, high thermal conductive Al-Si alloy. The product qualities are examined.

Further studies of product properties and improvements of process performance, especially with regard to the application of coating "hot" waste, are planned.

1977-04-26

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31.719

Titre Traitement des effluents réacteurs.	Pays : FRANCE Organisme directeur : CEA/DSN
Titre (anglais) Processing effluents in nuclear reactors.	Organisme exécuteur : CEA-DSN-SETSSR Responsable : J.GUIRLET
Date de démarrage : 1/1/76 Date prévue d'achèvement : 31/12/81 Etat actuel : en cours Dernière mise à jour : 1/4/77	Scientifiques : J.ROUX

Objectif général :

Etude générale des traitements et installations concernant les effluents liquides gazeux et le conditionnement des déchets solides du point de vue :

- de la nature des procédés mis en oeuvre
- des modes de fonctionnement
- des résultats attendus

Objectifs particuliers :

Etude particulière du fonctionnement des installations mises en place dans le réacteur en service, en cours de mise en exploitation ou en projet dans les domaines précités : nature des installations et résultats d'exploitation, particulièrement du point de vue des performances des installations de traitement des effluents liquides et du conditionnement des déchets.

Etat de l'étude :

1) Avancement à ce jour :

Actuellement analyse des informations contenues dans les documents d'analyse de sûreté des réacteurs en cours de démarrage et en projet.

Prochaines étapes :

Enseignement tiré des installations de traitement des réacteurs de Tihange et de Fessenheim.

Relation avec d'autres études :

Application aux analyses de sûreté
Rejets des installations.

.../...

Documents de référence :

Rapports de sûreté
Notices d'exploitation.

<u>Classification: 18</u>	
<u>Title 1 (Original Language):</u> Kritikalitäts-Studien (ATT 085 A - I.4.1 , Jahresbericht A 76)	<u>COUNTRY:</u> BRD
	<u>SPONSOR:</u> BMI
	<u>ORGANIZATION:</u> LRA, Garching
<u>Title 2 (English):</u> Criticality Studies	<u>Project Leader:</u> Prof. Dr. Birk- hofer W. Thomas
<u>Initiated (Date):</u>	<u>Completed (Date):</u>
<u>Status:</u> continuing	<u>Last Updating (Date):</u> December 1976

1. General Aim

Problems of criticality safety for fissile material outside reactors are investigated and solved. Safety criteria, critical and safe parameters will be established for handling of fissile material in enrichment procedures, fuel manufacturing, reprocessing, storage and transportation. Risks and consequences of criticality accidents for the environment and personnel will be evaluated.

2. Particular Objectives

Special problems of neutron interaction and isolation will be treated. The influence of concrete structures of hot cells on neutron reflection will be considered. Problems of homogeneous and heterogeneous neutron poisoning are investigated especially for reprocessing facilities. Assessments of radiological consequences of criticality accidents or other severe accidents in plutonium and uranium fabrication facilities are included.

3. Experimental Facilities and Program

No experimental work.

4. Project Status

4.1 Progress to Date

During last year criticality data for homogeneous mixtures of ^{233}U and thorium including homogeneous poisons have been computed. Further calculations have been performed for plutoniumtributyl phosphate mixtures and plutonium-polyethylene systems. The separation and reflection effects of various concrete reflectors with and without nuclear absorbers have been investigated in a series of calculations by Monte-Carlo technique. For all cases slab geometry has been assumed. Similar calculations for storage assemblies of plutonium nitrate cylinders in concrete vaults are under way. A study for use of boron or hafnium plates as nuclear poison has been done for an extraction column in the first cycle of an oxide fuel reprocessing plant. Overview calculations have been performed for the storage of spent PWR fuel bundles in poisoned casks.

In the field of risk analysis of the nuclear fuel cycle a theoretical analysis to predict the power and time dependence of criticality accidents has been issued. Further implementations in the computer code RADCA for calculation of the consequences of criticality accidents have been incorporated.

4.2 Essential Results

A new supplement has been issued to the Criticality Handbook.

5. Next Steps

Completion of our Criticality Handbook. A safety study is under way for plutonium recycle in LWR, especially for fuel manufacturing of uranium elements and U-Pu-mixed oxide fuel.

6. Relation with Other Projects

Relations exist with design studies for a new reprocessing plant performed at INR, Karlsruhe. Similar work is under way in the UK, France and USA, as documented in:

- Handbook of Criticality Data, J.H. Chalmers et al., AHSB 1965/7
- Guide de Criticité, CEA, CEA-R-3114, 1967
- Criticality Handbook Vol. I-III, R.D. Carter et al., ARH 600, 1968/72.

7. Reference Documents

/1/ .

W. Thomas

Plattenvergiftung als Kritikalitätskontrolle im ersten Extraktionszyklus einer Wiederaufarbeitungsanlage

MRR 158, Mai 1976

/2/

W. Heinicke, W. Thomas

Kritikalitätskontrolle durch heterogene Vergiftung bei der Lagerung von BE-Bündeln in Wasserbecken

atw 8, August 1976, S. 411-412

/3/

W. Heinicke, Ch. Müller, W. Thomas, R. Warnemünde, W. Weber

Handbuch zur Kritikalität, 7. Teillieferung

LRA Garching, November 1976

8. Degree of Availability

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

<u>Title 1 (Original language)</u> Valutazione delle proprietà chimiche e fisiche di rifiuti radioattivi solidificati.	<u>Classification</u> 18
<u>Title 2 (English)</u> Evaluation of chemical and physical properties of radioactive solidified wastes	<u>Country</u> ITALY <u>Sponsor</u> CNEN <u>Organisation</u> CNEN
<u>Date initiated</u> 1975 <u>Date completed</u> 1978 <u>Last updating</u> 1977	<u>Project Leader</u> A. DONATO

General aim

The technical and experimental control of the main chemical and physical properties of solidified radioactive wastes, for obtaining basic technical data to evaluate possible long-term environmental impacts. Liquid radioactive wastes (evaporation concentrates, exhausted ions exchangers etc.) and solids are incorporated in cement, concrete, bitumen or plastic. The following characteristics are taken into account: leachability, thermal stability, radiation damage, compressive strength and weathering resistance.

Particular objectives

Set-up of methods for testing

Project status

Urea-Formaldheyde and Polymer Impregnate Cement have been evaluated

Degree of availability

Cooperation may be envisaged for standardization of testing methods. Exchange of information can be arranged.

<u>Title 1 (Original language)</u> Ricerche e sviluppo su un contenitore di trasporto.	<u>Classification</u> 18
<u>Title 2 (English)</u> Research and development of a fuel cask.	<u>Country</u> ITALY <u>Sponsor:</u> C.A.M.E.N. <u>Organisation</u> C.A.M.E.N. - UNIVERSITY OF PISA
<u>Date initiated</u> : 1975 <u>Date completed:</u> In progress <u>Last updating</u> : April 1977	<u>Project Leader</u> G. SARNO (CAMEN) G. FORASASSI (UNIVERSITY)

Description :

Preliminary design, experimental researches, calculations (mechanics, heat transfer, etc.), model elaboration, tests and optimization studies on the necessary modification of an obsolete cask for spent fuel.

<u>Title 1 (Original language)</u> Ricerche su contenitori di trasporto per materiale fissile	<u>Classification</u> I8
<u>Title 2 (English)</u> Researches on shipping casks for fissile material	<u>Country</u> ITALY <u>Sponsor</u> CNEN <u>Organisation</u> (≠)
<u>Date initiated</u> 1974 (present phase) <u>Date completed</u> In progress <u>Last updating</u> April 1977	<u>Project Leader</u> CNEN - Divisione Ricerche Sicurezza

(≠) CNEN-ENEL-AGIP Nucleare-FIAT Nucleare - Nuovo Pignone-Università di Pisa.

General aim Research and development on shipping casks for fissile material.

Particular objectives Development of spent fuel shipping casks. Design, experimental researches, calculations (shielding, mechanics, heat transfer) model elaboration, tests, optimization studies on type of casks and means of transport related to power plants and reprocessing facilities.

Project status A first set of research contracts (CNEN-AGIP Nucleare, CNEN-FIAT Nucleare, CNEN-Pisa University) is completed; other additional studies have been carried out.

Next steps Researches related to some aspects of casks development and licensing.

Relation to other projects I8 (other programs)

<u>Title 1 (Original language)</u> Sicurezza del trasporto del materiale radioattivo.	<u>Classification</u> 18
<u>Title 2 (English)</u> Safety Problems in Design of Packaging and Transportation of Radioactive Materials.	<u>Country</u> ITALY <u>Sponsor</u> CNEN-CNR <u>Organisation</u> University of Pisa
<u>Date initiated</u> 1968 <u>Date completed</u> 1980 <u>Last updating</u> May 1977	<u>Project Leader</u> FORASASSI Giuseppe

- 1) General aim
 Study of safety problems in design, construction and use of transport casks for radioactive materials;
- 2) Particular objectives
 Present research activity is referred to determination of energy absorption characteristics of steel structures and to test models and components to support the design of spent fuel shipping casks.
- 3) Experimental facilities and program
 Facilities in Scalbatraio Center of University of Pisa can be utilized for experimental studies as well as to carry out IAEA standard test series on packagings for radioactive materials.
 The following equipments are operable:
 - a) Drop test tower for casks and models up to 2000 Kg of weight;
 - b) Guided impact hammer (weighting up to 2300 Kg) for dynamic tests;
 - c) Thermal test station (open fire);
 - d) Vessel for hydraulic and tightness tests;
 - e) Instrumentation for measures of acceleration, displacements, forces and temperatures during tests.
- 4) Project status
 - Research and testing series, completed two years ago, have brought to finalize the design of a low cost type B, fissile class II container;
 - Studies and test program have been carried to determine mechanical energy absorption properties of several types of metallic structures related to the design of casks, as:
 - Steel honeycomb
 - Straight and circular fins.
- 5) Next steps
 Up to date study and test programs include:
 - Several tests on finned surfaces
 - study of containment problems as well as performances of gaskets
 - Studies and dynamic tests on bolts and tie-downs
 - Study of problems related to the use of models in standard tests.

<u>Title 1 (Original language)</u>	<u>Classification</u>
Sicurezza del trasporto del materiale radioattivo.	18

6) Relation to other projects

18 (Other programs)

7) Reference documents:

1. G. ELETTI, R. ZAMBRINI, G. FORASASSI, P. SACCOMANNO
Sicurezza del trasporto del materiale radioattivo.
Atti del convegno sulla sicurezza negli Impianti Nucleari.
Pisa 21-26 Settembre 1970.
2. P. CITTI, G. FORASASSI, B. GUERRINI
Rilievi su prototipi di un contenitore per il trasporto di materiale radioattivo.
Atti del 1° Convegno AIAS - Palermo 1972.
3. G. ELETTI, G. FORASASSI
Italian testing station following IAEA Regulation; CF6 a fissile Class II Type B packaging.
4° International Symposium on packaging and transportation of Radioactive materials.
22-27 Sept. 1974.
4. P. CITTI, G. FORASASSI, S. REALE
Problemi di similitudine connessi con due prove standard per contenitori per trasporto di materiale radioattivo.
Ist. Imp. Nucl. Università di Pisa. RL 171(74).
5. G. FORASASSI, B. GUERRINI
Comportamento di una struttura metallica a celle esagonali in prove di simulazione di "fuel cask drop".
Ist. Imp. Nucl. Università di Pisa. RL 204(75).
6. G. FORASASSI
Assorbimento di energia meccanica mediante superfici alettate.
Ist. Imp. Nucl. Università di Pisa. RL 209(75).

PROJECT TITLE: A Test study for Safety Evaluation in case that a Spent-Fuel Shipping Cask Drops into the Deep Sea.	CSNI INDEX NUMBER: 18
SPONSORING COUNTRY: Japan	ORGANIZATION: Mitsui Engineering & Shipbuilding Co., Ltd.
DATE INITIATED : May 1976 DATE COMPLETED : March 1977	PROJECT LEADER: M. Sakata

Description :

1) Objectives

The objective of this study is to obtain the data for safety evaluation in case that a spent-fuel shipping cask drops into the deep sea by accident.

2) Project Status

- (1) A test study for water tightness at closure head and closure head seals of a cask.

This test under the high external pressure (Max. 500 kg/cm², water) was carried out on a 1/4 scale model of the cask of 80 tons in weight. Deformation and sealing performance of the specimen and water tightness were observed during the test.

- (2) A test study for collapse strength and water tightness of the drain valve and pressure relief valve.

This test under the high external pressure (Max. 500 kg/cm², water) was carried out on the valves. Water tightness of the specimens was observed during the test.

- (3) Research and development of pressure equalizer

The pressure equalizer (valve) has such structural features as to open valve seat over the pre-set differential pressure, allow inleakage of sea water and equalize the pressures between the exterior and interior of a cask within the pre-set limit.

This test under the high external pressure (Max. 500 kg/cm², water) was carried out on this valve. Sealing performance was observed during the test.

This study was supported by the ATomic Energy Bureau, Science and Technology Agency of Japan with the Research Fund for the Peaceful Uses of Atomic Energy.

3) Essential Results

1. Water would not leak in form the closure head seal to ballance the pressures inside and outside of the specimen at higher pressure.
2. Pressure equalizer was found to function satisfactorily.

4) Projection

None

5) Reports

IAEA-SR-10/31 Collapse test of a model shipping cask under high external pressure, O. Terada et al. August 1976.

PROJECT TITLE : Study on Fire Resistance of Spent Fuel Shipping Casks		CSNI INDEX NUMBER : 18
SPONSORING COUNTRY : Japan	ORGANISATION : Special Fire Section, 2nd Research Division Fire Research Institute, Fire Defence Agency	
DATE INITIATED : April, 1968	PROJECT LEADER :	
DATE COMPLETED : Progressing (March 1977)	H. Shimada	

Description :

1. Objective

The purpose of this study is to obtain fundamental data for assessing fire safety of spent fuel shipping casks, through thermal test of cask models.

2. Project Status

Thermal tests were conducted to obtain the transient temperature distribution in a cask model, 1/8 scale of 80 ton cylindrical prototype cask, in a fire-simulated furnace and under actual open fire conditions.

3. Essential Results

The temperatures in the cask model went up less in the open fire tests than in the furnace tests, because of low emissivity of flame.

4. Projection

A comparison and evaluation of thermal test methods will be made to establish a thermal test equivalent to actual fires. The effect of a minute gap between a outer shell and a lead layer of the cask on its fire resistance will be investigated.

5. Reports

H. Shimada; Fire Resistivity of Irradiated Nuclear Fuel Shipping Cask, Report of Fire Research Institute of Japan, Serial No. 39, March 1975

PROJECT TITLE : Study on the Safety of Spent Nuclear Fuel Shipping Casks	CSNI INDEX NUMBER : 18
SPONSORING COUNTRY : Japan	ORGANISATION : Ship Research Institute
DATE INITIATED : March, 1976 DATE COMPLETED : Progressing (Mar. 77)	PROJECT LEADER : Hisao Yamakoshi

Description :

1 Objectives:

The purpose of this study is to make a reliable calculation code which is able to estimate the intensities of the radiation sources contained in the spent fuels and the radiation level on and around a cask.

2 Project Status:

The ORIGEN code developed in ORNL is equipped for the computer TOSBAC-5600/120 of Ship Research Institute so that we are able to calculate the amount of radioactive materials contained in spent fuels and intensities of radiation sources.

The shielding effects of cooling fins equipped on a cask has been estimated by detecting gamma-rays transmitted through a cylindrical iron shell with cooling fins. The gamma-ray source is a ⁶⁰Co.

3 Essential Results:

The amount of radioactive materials contained in spent fuels, and intensities and energy spectra of photons and neutrons in it are tabulated with the ORIGEN code as a function of reactor type, enrichment, specific power, and cooling times.

The cooling fins of casks are expected to reduce the gamma-ray dose level on and around the cask. The shielding effects of the fins depend upon the radiation source, configuration and material of the fins, and distance from the cask to the detector.

4 Projection:

A reliable calculation method will be established for evaluation of radiation dose level on and around a cask by taking account of the experimental results on the shielding effects of the fins and the results of the ORIGEN calculations.

5 Report:

"An Examination on Neutron Shielding for Spent Fuel Shipping Casks" K. Ueki, H. Yamakoshi, K. Takeuchi. 5 th International Conference on Reactor Shielding, April 1977.

OECD NUCLEAR ENERGY AGENCY - 38, Boulevard Suchet, 75016 PARIS
Telex : 630668 AEN NEA - Tel : 524 82-00

PROJECT TITLE : Management of radioactive waste slurry from nuclear power plants	CSNI INDEX NUMBER : 18
SPONSORING COUNTRY : Sweden	ORGANISATION : AB Atomenergi
DATE INITIATED : January 1974 DATE COMPLETED :	PROJECT LEADER : Arne Lindskog

Description :

The origin and volumes of radioactive waste slurry from nuclear power plants are investigated. The amounts of radioactive nuclides in slurry from primary coolant clean-up systems and condensate clean-up systems are calculated. Appropriate segregation and treatment methods, such as drying, burning, solidification and storage, has been studied.

As a result of the study work is now concentrated in the design and construction of silos for storing of radioactive waste slurry after adequate preparation.

PROJECT TITLE : Management of spent non-fuel bearing core components	CSNI INDEX NUMBER : 18
SPONSORING COUNTRY : Sweden	ORGANISATION : AB Atomenergi, Wasta Management Section
DATE INITIATED : 1975-07-01 DATE COMPLETED : 1976-06-30	PROJECT LEADER : Claes Thegerström

Description :

The project comprises

- Compilation of data on types and numbers of spent core-components and on facilities for the handling of such highly active components.
- Calculation of neutron induced activity.
- Study of appropriate treatment (cutting, compaction and packaging).
- Investigation of transportation and storage systems.
- Safety and economic aspects.

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Telex : 637668 AEN NEA - Tel : 524 82-00

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19. ECONOMICS OF SAFETY
(no entries)

20. LARGE SCALE TEST FACILITIES - INTEGRAL STUDIES

<u>Classification:</u> 20	
<u>Title 1 (Original Language):</u> HDR-Sicherheitsprogramm (RS 0123 A + B - III, Jahresbericht A 76)	COUNTRY: BRD
	SPONSOR: BMFT
	ORGANIZATION: GfK, Karlsruhe
<u>Title 2 (English):</u> HDR Safety Program	<u>Project Leader:</u> Müller-Dietsche
<u>Initiated (Date):</u> April 1, 1974 <u>Status:</u> continuing	<u>Completed (Date):</u> 1981 <u>Last Updating (Date):</u> December 1976

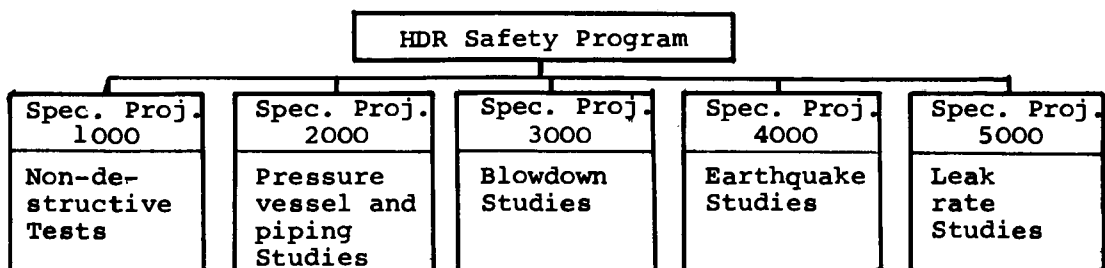
1. General Aim

The safety program which will be performed on the decommissioned HDR-plant near Frankfurt/Germany, serves for clarification of important problems within the framework of the light water reactor safety research program of the Federal Minister of Research and Technology (BMFT).

The unrestricted availability of this reactor for investigations concerning safety technology offers the rare possibility to check on a complete plant, the validity of engineering theories, applied design methods and the results achieved in current research projects up to the actual limits of stress bearing capacity of specific reactor components.

2. Particular Objectives

The HDR safety program is divided into five Specific Projects (SP):



These five Specific Projects were determined due to the main objectives presently pursued under the Reactor Safety Research Program of the Ministry after harmonization with the involved authorities.

3. Experimental Facilities and program

3.1 Experimental Facilities

The HDR with 100 MWth power was built to demonstrate the possibility of nuclear superheating. Typewise, it largely corresponds to the German water boiling reactor designed around 1967/1968. The main data can be seen from Table 1. Particular importance is attached to a fully pressure containment and the reactor pressure vessel. No nuclear operation is envisaged for the experiments. The experimental conditions will be obtained with an electrically heated boiler.

3.2 Program

3.2.1 Specific Project 1000: Non-destructive Testings

Non-destructive testings serve for controlling the reactor pressure vessel and the piping under pressure tests and blowdown-tests and for testing in-service inspection systems under development. The following objectives will be investigated:

- Assessment of record of the initial conditions
- Detection of failure generation and failure propagation
- Testing of inspection methods; proof of the suitability of new test methods
- Comparative evaluation of non-destructive testing systems, such as ultrasonic impulse testing, magnetic particle testing, penetration testing, eddy current method, ultrasonic scattering, acoustic emission, acoustic holography, radiography.

3.2 Specific Project 2000: Pressure Vessel and Piping Studies

Studies of pressure vessel and piping should be a further step of intensifying and securing the basic knowledge of their safe design. The most important aspect is a study of the effective safety margin above the design ratings of components. The following objectives are pursued:

- Material properties at the beginning of the experiments.
- Component behavior under special loading conditions, e.g., operating conditions, blowdown, thermoshock, earthquake.

- Component behavior under specific weakening - fabrication and operational defects and crack formations - to determine boundary strengths or critical defects.
- Reliability of the employed methods of calculation.

3.2.3 Specific Project 3000: Blowdown Studies

In the blowdown studies it is possible to measure radial stresses of reactor pressure vessel internals, containment structures and full scale valves by means of reactor typical impulse and mass-flow excitations in their realistic dimensions. The studies comprise the following objectives:

- Stress behavior of the reactor pressure vessel internals and containment structures
- Behavior of full scale valves under mass-flow excitation.
- Testing and further development of the various fluid and structural dynamic computer codes.

3.2.4 Specific Project 4000: Earthquake Studies

Vibration-studies serve to improve and correct the basic knowledge about the safe design of building structures and pipe-systems with respect to the effects of earthquake. This implies the following objectives:

- Verification and optimization of available analytic methods
- Influence of material and soil characteristics and design conditions
- Testing of experimental techniques
- Advancement of analytical procedures for the non-linear range.

3.2.5 Specific Project 5000: Leak Rate Studies

These studies serve to determine the previously unsettled parameters with the aim to standardize the leak rate procedure and the in-service studies of nuclear power stations.

This involves the following objectives:

- Leak behavior of a cold plant.
- Leak behavior of a plant at operating temperature.
- Verification and optimization of existing analytical procedures.

4. Project Status

4.1 Progress to Date

Central Preparation of the HDR Experiments

The last few highly radioactively contaminated components were removed from the plant and most of the work associated with the modification for experimental operation was finished. The test loop with an electric boiler to simulate conditions close to the operating conditions (310°C, 110 bar) was installed. In addition, a central measured data acquisition system for 450 fast and 150 slow measuring stations has been ordered.

Specific Project 1000: Non-destructive Tests

To assess the actual state of the components under study in an accurate way, a first fully volumetric automatic ultrasonic test of the RPV was carried out with a special type of manipulator built for this purpose. Areas specially selected of the RPV pipe connections, top lid, welds, defect areas, transitions and parts of the piping system were additionally tested manually (manual ultrasonic test, manual tandem test, radiography).

In a pressure test (143 bar, 50°C) sound emission measurement was used on the RPV and the piping system. It furnished various noise readings whose evaluation, however, indicates that leakages and other defect noises had been detected and that there had not been any changes in the RPV.

Specific Project 2000: RPV and Piping Studies

The first materials data and the structural conditions were determined in material studies and metallographic examinations of drill cores taken from the pressure vessel and of specimens of the piping system. It turned out that the strength and ductility properties of the materials in the base material and the welds satisfied the requirements as specified and did not show any weakening defects.

An experimental stress analysis carried out on the pressure vessel in the "cold" pressure test (143 bar, 50°C) revealed peak stresses and stress distributions on the RPV and certain regions of the piping system. A first comparison between measurement and calculation of the stresses determined on a large RPV-nozzle showed good agreement.

Specific Project 3000: Blowdown Experiments

Preliminary calculations performed as a basis for the experiments were carried out on the RPV with respect to the following criteria:

- Development of pressure relief waves under various boundary conditions.
- Load acting upon the internals as a function of the length and diameter of the pipe connections.
- Influence of the initial pressure and temperature distributions upon the load forces.

For the containment, the factors calculated were

- the differential pressure loads acting upon the structures,
- the pressure and temperature development to be expected.

Fabrication of the test components, especially of the RPV internals and the valves, was continued. The extensive measuring gear is being purchased.

Specific Project 4000: Earthquake Studies

The experiments carried out with low excitation and various types of excitation, i.e.,

- shaker, snapback, explosion

were evaluated. A first comparison with vibration calculations showed differences between the measurement and calculation, both in the eigenfrequencies determined and in the amplitudes.

Extensive calculations of the vibration behavior under the impact of low excitation compared with measurements have been initiated by various institutions.

Specific Project 5000: Leak Rate Studies

In the period April/May/June 1976 leak rate measurements were carried out on the cold plant. Especially the influence of the sequence of pressure steps, the action of the air trapped in the concrete and the critical swelling pressure were investigated.

4.2 Main Results

The first automatic ultrasonic test of the RPV, the additional manual ultrasonic tests and the material studies indicated that the pressure vessel and the piping system of the HDR did not show any sizable defects or cracks before the start of the blowdown experiments. A de-

tailed stress analysis at 146 bar, 50^o C revealed no strains; in the elastic region the measurement and the finite element calculation showed good agreement.

Here are some of the results of the leak rate studies:

- Checking the leak rate at 0.5 bar is sufficient for the recurrent tests. Extrapolation of the leak rate from the rest pressure to the design pressure is possible. This, e.g., does not apply for a test pressure of 0.03 bar or 1.5 bar.
- The leak rate values measured during the pressure surge phase are on the conservative side.
- A period of assessment of 4 - 6 hours is not sufficient to allow the in- and outgassing effects in the concrete to be neglected. Also after the end of the transient time there may well be in- and outgassing effects accompanying major changes in the absolute pressure.

5. Next Steps

The test loop will be tried out. Afterwards the whole measuring setup will be installed for the first blowdowns, which are planned for mid-1977. They will include measurements of pressure, temperature, strain, displacement, density, mass flow, water entrainment, water separation, and heat transfer.

Important points of further investigations include

- experimental stress analysis (Spec. Proj. 2000) in the "hot" pressure test (310^o C, 110 bar) for RPV and pipings.
- Leak rate measurements on the containment in the "hot" state (SP 5000).
- First Blowdown experiments with steam isolation valves of NW 350 dia. (SP 3000).
- Theoretical studies of the vibration behavior of the plant under the impact of low excitations compared with measurements.

6. Relation with Other Projects

1. Investigation of the Phenomena Involved in the Depressurization of Water Cooled Reactors RS 16/2
2. Investigation of the Phenomena Occurring within a Multicompartiment 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 (in English).

8. Degree of Availability

Unrestricted distribution.

Table 1

Main Plant Data

<u>Safety Containment:</u>	Dimen- sions	
Inside height	m	60
Inside diameter	m	20
Test pressure	atm.g.	7,12
Design pressure: Overpressure	atm.g.	5,60
Underpressure	mm water	350
Design temperature	°C	155
Wall thickness of cylindrical part	mm	30
Material specification of steel cladding	-	F8 50 S fine grained steel
Free containment volume	m ³	10,853

<u>Pressure Vessel:</u>	Dimen- sions	RDB	HDU	SDU	UX
Number	-	1	1	1	2
Inside diameter	mm	2960	1765	1978/1508	700
Overall height	mm	12 700	14 110	10 920	4365
Wall thickness (average)	mm	112	45	61/46	23
Design pressure	atm.g.	110	110	110	110
Design temperature	°C	360	550	400	320
Material	-	23 NiMoCr 36	8Hw 38	8Hw 38	3H 39 S
Flow: primary	t/h	-	170	170	170
secondary	t/h	-	130	130	130

<u>Water Circuits:</u>	Dimen- sions	Primary circuit between			Secondary circuit up to reducer station	Recirculation loop
		RDB and HDU	HDU and SDU	SDU and RDB		
Design pressure	atm.g.	110	110	112	110	112
Design temperature	°C	550	400	320	400	320
Material	-	No. 4961	No. 4550	No. 4550	No. 7335	No. 4550
Inside diameter	mm	250 - 300	250	200	300 - 350	350 - 450

Notes:

Reactor pressure vessel = RDB
 Superheated steam converter = HDU
 Saturated steam converter = SDU
 Subcooler = UX

