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NRC LIGHT-WATER REACTOR SAFETY RESEARCH
FOREIGN TECHNICAL EXCHANGE PROGRAM
VOL. IV
(JULY-DECEMBER 1977)

Debbie S. Sharp

Wm. B. Costrell

Prepared for the U.S. Nuclear Regulatory Commission
Office of Technical Research and Development
Under Interagency Agreement with the U.S. Atomic Energy Commission

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	Reactor Operating Experiences, 1972-1974, U.S. Nuclear Regulatory Commission, Dec. 1975	\$ 8.00

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Debbie S. Sharp and Wm. B. Cottrell
Engineering Technology Division

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FOREWORD

The Nuclear Safety Information Center (NSIC), which was established in March 1963 at Oak Ridge National Laboratory, is principally supported by the U.S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research. Support is also provided by the Division of Reactor Development and Demonstration of the Department of Energy. NSIC is a focal point for the collection, storage, evaluation, and dissemination of safety information to aid those concerned with the analysis, design, and operation of nuclear facilities. Although the most widely known product of NSIC is the technical progress review *Nuclear Safety*, the Center prepares reports and bibliographies as listed on the inside covers of this document. The Center has also developed a system of key words to index the information which it catalogs. The title, author, installation, abstract, and key words for each document reviewed are recorded at the central computing facility in Oak Ridge. The references are cataloged according to the following categories:

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19. Internal Exposure Effects on Humans Due to Radioactivity
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ABSTRACT

Lists of documents exchanged during the second half of 1977 under agreements between the U.S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research and the governments of France, Federal Republic of Germany, Japan, and the United Kingdom are presented. During this period, the NRC received 1 report from France, 40 from the Federal Republic of Germany, and 11 from Japan, and in return sent 112 U.S. reports to each of these three countries and 23 reports to the United Kingdom.

INTRODUCTION

This report lists those documents exchanged during the second half of 1977 under agreements between the U.S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research and the governments of France, Federal Republic of Germany, Japan, and the United Kingdom. This is the fourth report in this series. The first had a similar title, but covered the period November 1974 through December 1975 and was published in May 1976 as TID-3362. The second report covered 1976 and was published in April 1977 as ORNL/NUREG/NSIC-134. The third report covered the first half of 1977 and was published in October 1977 as ORNL/NUREG/NSIC-142. This report differs from its predecessors in that the latter half of 1977 witnessed the extension of the light-water reactor (LWR) exchange agreements to include the United Kingdom and the implementation of exchange agreements on both high-temperature gas-cooled reactor (HTGR) and fast reactor safety research. The HTGR safety research exchange agreement is between the U.S. and Japan. The fast reactor safety research exchange agreements are between the U.S. and the Federal Republic of Germany, Japan, and the United Kingdom, respectively. However, as of the end of

1977, the document flow under these new exchange agreements had begun only with the United Kingdom for LWR documents and with Japan for HTGR documents.

The total number of reports received by the NRC from July through December 1977 from France, Federal Republic of Germany, Japan, and the United Kingdom, as well as the reports sent by NRC to each of these four countries during this period, are tabulated below for both the LWR and HTGR exchanges. Also tabulated are the documents exchanged in 1974 and 1975 as listed in TID-3362, in 1976 as reported in ORNL/NUREG/NSIC-134, and in the first half of 1977 as reported in ORNL/NUREG/NSIC-142. The number of proprietary documents received are listed in parentheses.

Reports received by NRC^a

	France		F.R. Germany		Japan		UK	
	LWR	HTGR	LWR	HTGR	LWR	HTGR	LWR	HTGR
July-December 1977	1	0	40	0	11	0	0	0
January-June 1977	41(12)	0	29(1)	0	24(1)	0	0	0
1976	25(6)	0	75(9)	0	38(5)	0	0	0
1974-1975	10	0	6	0	24	0	0	0

^aThe number of proprietary documents received are listed in parentheses.

Reports sent by NRC

	France		F.R. Germany		Japan		UK	
	LWR	HTGR	LWR	HTGR	LWR	HTGR	LWR	HTGR
July-December 1977	112	0	112	0	112	37	23	0
January-June 1977	107	0	107	0	107	0	0	0
1976	119	0	154	0	155	0	0	0
1974-1975	100	0	181	0	115	0	0	0

For convenience in processing, each of the foreign reports received under the exchange agreements is assigned a unique number identifying it as part of the exchange. The documents from France, Federal Republic of Germany, and Japan are listed in the computerized bibliography first by country, then alphabetically by installation, and finally chronologically by report date. Additional bibliographic information, including the number assigned by the issuing installation, abstract, and keywords, is also presented.

As was done in previous reports in this series, the U.S. reports transmitted to foreign exchange agreement countries are listed alphabetically; the LWR and HTGR reports are listed separately.

All reports, both foreign and domestic, with the exception of proprietary documents, are available from the National Technical Information Service, U.S. Department of Commerce, Springfield, VA 22161, in either microfiche or hard copy. (In most cases, however, the hard copies of foreign reports will be reproduced from microfiche.)

NRC LIGHT-WATER REACTOR SAFETY RESEARCH REPORTS
TRANSMITTED BY THE NRC TO
FOREIGN EXCHANGE AGREEMENT COUNTRIES

The following is a cumulative listing of the U.S. reports transmitted by the NRC to France, Federal Republic of Germany, Japan, and the United Kingdom during the period July-December 1977.

ANL-77-4. *Occurrence of Critical Heat Flux During Blowdown With Flow Reversal.* January 1977. Transmitted October 12, 1977.

ANL-77-10. *Light-Water-Reactor Safety Research Program: Quarterly Progress Report.* March 1977. Transmitted July 8, 1977.

ANL-77-34. *Light-Water-Reactor Safety Research Program: Quarterly Progress Report.* June 1977. Transmitted September 28, 1977.

BMI-NUREG-1971. *Evaluating Strength and Ductility of Irradiated Zircaloy.* April 1977. Transmitted August 10, 1977.

BMI-NUREG-1972. *Steam-Water Mixing and System Hydrodynamics Program.* May 1977. Transmitted July 22, 1977.

BMI-NUREG-1973. *Penetration Behavior in a 1/15-Scale Model of a Four-Loop Pressurized Water Reactor.* June 1977. Transmitted August 25, 1977.

BMI-NUREG-1976. *Evaluating Strength and Ductility of Irradiated Zircaloy.* July 1977. Transmitted November 28, 1977.

BNWL-1956. *Release of Indigenous Gases from LWR Fuel and the Reaction Kinetics with Zircaloy Cladding.* April 1977. Transmitted July 8, 1977.

BNWL-2214. *COBRA-IV: The Model and the Method.* July 1977. Transmitted August 25, 1977.

BNWL-2279. *Quarterly Progress Report to Nuclear Regulatory Commission Division of Reactor Safety Research Fuel Behavior Research Branch Programs for Period Ending March 1977.* April 1977. Transmitted October 12, 1977.

BNL-NUREG-50661. *Reactor Safety Research Programs Quarterly Progress Report, January 1-March 31, 1977.* May 1977. Transmitted September 28, 1977.

CREARE TN-258. *Lower Plenum Voiding With Scaled PWR Plenum Geometries.* May 1977. Transmitted August 25, 1977.

CREARE TN-261. *Preliminary Ramped Transient Analysis with Superheated Walls.* August 1977. Transmitted October 31, 1977.

GEAP-NUREG-21638. *BWR Blowdown/Emergency Core Cooling Program - 64-Rod Bundle Core Spray Interaction (BD/ECC1A) Test Plan.* June 1977. Transmitted November 28, 1977.

LA-NUREG-6797-SR. *Numerical Study of Downcomer Flow Dynamics.* May 1977. Transmitted July 8, 1977.

LA-NUREG-6818-MS. *FPDCYS and FPSPEC: Computer Programs for Calculating Fission-Product Beta and Gamma Multigroup Spectra from ENDF/B-IV Data.* May 1977. Transmitted August 10, 1977.

LA-NUREG-6842-PR. *Nuclear Reactor Safety Quarterly Progress Report, January 1-March 31, 1977.* June 1977. Transmitted September 28, 1977.

LA-NUREG-6896-MS. *Comparisons of Calculated and Experimental Delayed Fission-Product Beta and Gamma Spectra from ^{235}U Thermal Fission.* July 1977. Transmitted October 12, 1977.

LA-NUREG-6917-MS. *Fission-Product Source Terms for Reactor Applications.* August 1977. Transmitted October 31, 1977.

LA-NUREG-6934-PR. *Nuclear Reactor Safety, Quarterly Progress Report, April 1-June 30, 1977.* August 1977. Transmitted October 1977.

NUREG-0007-2. *Improved Ultrasonics Nondestructive Testing of Pressure Vessels, Annual Progress Report.* 1977. Transmitted October 12, 1977.

NUREG-0018-6. *Evaluation of Fission Product Afterheat.* June 1977. Transmitted July 22, 1977.

NUREG-0035-3. *Inspection of Nuclear Reactor Welding by Acoustic Emission.* June 1977. Transmitted July 22, 1977.

NUREG-0047-8. *Evaluation of Pressure Drop Across Area Changes During Blowdown, Quarterly Progress Report January 1, 1977-March 31, 1977.* July 1977. Transmitted September 28, 1977.

NUREG-0090-7. *Report to Congress on Abnormal Occurrences.* June 1977. Transmitted July 22, 1977.

NUREG-0090-8. *Report to Congress on Abnormal Occurrences.* September 1977. Transmitted September 28, 1977.

NUREG-0234. *Summary of LWR Safety Research in the USA.* May 1977. Transmitted July 8, 1977.

NUREG-0237. *A Phenomenological Correlation for Post-CHF Heat Transfer.* June 1977. Transmitted August 10, 1977.

NUREG-0239. *Residual Stresses at Girth-Butt Welds in Pipes and Pressure Vessels.* May 1977. Transmitted July 8, 1977.

NUREG-0280. *Some Aspects of Interpreting Two-Phase Flow Measurements in Instrumented Piping Spool Pieces.* June 1977. Transmitted July 8, 1977.

NUREG-0288. *Modeling of Pool Swell Dynamics.* June 1977. Transmitted July 22, 1977.

NUREG-0289. *Critical Experiments, Measurements, and Analyses to Establish a Crack Arrest Methodology for Nuclear Pressure Vessel Steels.* July 1977. Transmitted August 10, 1977.

NUREG-0306 (VOL. 1). *Experimental Data on Two-Phase Pressure Drops Across Area Changes During Flow Transients.* July 1977. Transmitted August 25, 1977.

NUREG-0306 (VOL. 2). *Experimental Data on Two-Phase Pressure Drops Across Area Changes During Flow Transients.* July 1977. Transmitted August 25, 1977.

NUREG-0307. *Review and Assessment of Research Relevant to Design Aspects of Nuclear Power Plant Piping Systems.* July 1977. Transmitted August 25, 1977.

NUREG-0311. *A Treatment of the Subject of Tearing Instability.* August 1977. Transmitted August 25, 1977.

NUREG-0342. *Photoelastic Studies of Crack Propagation and Crack Arrest.* October 1977. Transmitted November 28, 1977.

ORNL/NUREG-7. *Test of 6-in.-Thick Pressure Vessels. Series 4: Intermediate Test Vessels V-5 and V-9 with Inside Nozzle Corner Cracks.* August 1977. Transmitted October 31, 1977.

ORNL/NUREG/TM-95. *Multirod Burst Test Program Quarterly Progress Report for October-December 1976.* April 1977. Transmitted 1977.

ORNL/NUREG/TM-112. *Noise Diagnostics for Safety Assessment, Quarterly Progress Report for January-March 1977.* June 1977. Transmitted July 8, 1977.

ORNL/NUREG/TM-114. *PWF Blowdown Heat Transfer Separate-Effects Program - Thermal-Hydraulic Test Facility Experimental Data Report for Test 100.* August 1977. Transmitted September 28, 1977.

ORNL/NUREG/TM-120. *Heavy-Section Steel Technology Program Quarterly Progress Report for January-March 1977.* September 1977. Transmitted October 31, 1977.

ORNL/NUREG/TM-121. *Fission Product Beta and Gamma Energy Release, Quarterly Progress Report for January-March 1977.* July 1977. Transmitted August 10, 1977.

ORNL/NUREG/TM-122. *Quarterly Progress Report on Fission Product Behavior in LWRs for the Period January-March 1977.* 1977. Transmitted September 28, 1977.

ORNL/NUREG/TM-125. *Quarterly Progress Report on the Creepdown and Collapse of Zircaloy Fuel Cladding for July 1976-March 1977.* July 1977. Transmitted August 10, 1977.

ORNL/NUREG/TM-128. *PWR Blowdown Heat Transfer Separate-Effects Program - Thermal-Hydraulic Test Facility Experimental Data Report for Test 101.* September 1977. Transmitted October 31, 1977.

ORNL/NUREG/TM-139. *Quarterly Progress Report on Fission Product Behavior in LWRs for the Period April-June 1977.* September 1977. Transmitted October 31, 1977.

ORNL/NUREG/NSIC-135. *Bibliography of Reports on Research Sponsored by the IIRC Office of Nuclear Regulatory Research July-December 1976.* March 1977. Transmitted 1977.

ORNL/NUREG/NSIC-136. *Design Data and Safety Features of Commercial Nuclear Power Plants Including Cumulative Index for Volumes I through VI.* June 1977. Transmitted July 22, 1977.

ORNL/NUREG/NSIC-137. *Annotated Bibliography of Safety-Related Occurrences in Boiling-Water Nuclear Power Plants as Reported in 1976.* August 2, 1977. Transmitted October 31, 1977.

ORNL/NUREG/NSIC-138. *Annotated Bibliography of Safety-Related Occurrences in Pressurized-Water Nuclear Power Plants as Reported in 1976.* August 1977. Transmitted October 31, 1977.

ORNL/NUREG/NSIC-140. *Structural Integrity of Materials in Nuclear Service - A Bibliography.* July 1977. Transmitted August 25, 1977.

ORNL/Sub-7015/1. *Stress Intensities for Nozzle Cracks in Reactor Vessels.* November 1976. Transmitted July 8, 1977.

SAND76-0715 (NUREG-0237). *Proposed Research on Class 1 Components to Test a General Approach to Accelerated Aging Under Combined Stress Environments.* April 1977. Transmitted July 8, 1977.

SAND77-0370. *A Preliminary Model for Core/Concrete Interactions.* August 1977. Transmitted September 28, 1977.

SAND77-0952. *2-MW Plasmajet Facility Thermal Tests of Concrete.* July 1977. Transmitted October 12, 1977.

TREE-NUREG-1055. *Experiment Data Report for Semiscale MOD-1 Tests S-05-6 and S-05-7 (Alternate ECC Injection Tests).* June 1977. Transmitted August 25, 1977.

TREE-NUREG-1056. *Analysis of Standard Problem Six (Semiscale Test S-06-6) Data.* August 1977. Transmitted October 12, 1977.

TREE-NUREG-1058. *Semiscale MOD-1 System Response During the ECC Baseline Test Series (Design of the MOD-1 Baseline ECC Test).* June 1977. Transmitted July 22, 1977.

TREE-NUREG-1072. *Embedded Cladding Surface Thermocouples on Zircaloy-Sheathed Heater Rods.* June 1977. Transmitted August 25, 1977.

TREE-NUREG-1074. *Irradiation Effects Test Series, Test IE-2, Test Results Report.* August 1977. Transmitted November 28, 1977.

TREE-NUREG-1082 (ANCR-NUREG-1350). *Factors that Affect the Calibration of Turbines in Single-Phase Flow.* May 1977. Transmitted July 8, 1977.

TREE-NUREG-1083. *LOFT Facility PSS Experiments: Analysis of Wet Well Vertical Loads Resulting from Transient Initiation.* May 1977. Transmitted July 22, 1977.

TREE-NUREG-1084. *Experiment Data Report for LOFT Nonnuclear Test LI-4.* July 1977. Transmitted October 12, 1977.

TREE-NUREG-1087. *Design Data for 1/4-in.-thick Alloy 718 In-Pile Tube.* May 1977. Transmitted July 8, 1977.

TREE-NUREG-1107. *FRAP-S2: A Computer Code for the Steady State Analysis of Oxide Fuel Rods.* July 1977. Transmitted October 31, 1977.

TREE-NUREG-1118. *Evaluation of the Effects of Break Nozzle Configuration in the Semiscale MOD-1 System.* August 1977. Transmitted October 31, 1977.

TREE-NUREG-1121. *Experiment Data Report for Semiscale MOD-1 Test S-06-1 (LOFT Counterpart Test).* July 1977. Transmitted September 28, 1977.

TREE-NUREG-1122. *Experimental Data Report for Semiscale MOD-1 Test S-06-2 (LOFT Counterpart Test).* August 1977. Transmitted October 12, 1977.

TREE-NUREG-1125. *Experiment Data Report for Semiscale MOD-1 Test S-06-4. (LOFT Counterpart Test).* June 1977. Transmitted August 25, 1977.

TREE-NUREG-1128. *Quarterly Technical Progress Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, January-March 1977.* June 1977. Transmitted September 28, 1977.

TREE-NUREG-1131. *LOFT Fuel Modules Design, Characterization, and Fabrication Program.* June 1977. Transmitted 1977.

TREE-NUREG-1147. *Quarterly Technical Progress Report on Water Reactor
 Safety Programs Sponsored by the Nuclear Regulatory Commission's Division
 of Reactor Safety Research April-June 1977. August 1977. Transmitted
 October 31, 1977.*

WCAP-9108. *FLECHT Low Flooding Rate Skewed Test Series Data Report.
 May 1977. Transmitted August 25, 1977.*

*Computer Listings of Licensee Event Reports Sorted by Facility. August
 31, 1977. Transmitted September 28, 1977.*

**NRC HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY RESEARCH REPORTS
TRANSMITTED BY THE NRC FOREIGN EXCHANGE AGREEMENT COUNTRIES**

The following is a cumulative listing of the U.S. reports transmitted by the NRC to Japan only during the period July-December 1977.

BNL-NUREG-50574. *Correlation of High- and Low-Cycle Fatigue Data for Incoloy-800H.* October 1976. Transmitted August 23, 1977.

BNL-NUREG-50599. *HTGR Safety Evaluation Division Quarterly Report April-June 1976.* December 1976. Transmitted August 25, 1977.

BNL-NUREG-50601. *A Correlation of High- and Low-Cycle Fatigue Data for Solution-Treated Type 304 Stainless Steel.* December 1976. Transmitted August 25, 1977.

BNL-NUREG-50610. *Some Correlations of High- and Low-Cycle Fatigue Data for 2 1/4 Cr-1 Mo Steel in the Annealed, and Normalized and Tempered Conditions.* January 1977. Transmitted August 25, 1977.

BNL-NUREG-50624. *Reactor Safety Research Programs, Quarterly Progress Report, October 1-December 31, 1976.* February 1977. Transmitted August 25, 1977.

BNL-NUREG-50645. *HTGR Safety Evaluation Division Quarterly Report, July-September 1976.* March 1977. Transmitted August 25, 1977.

BNL-NUREG-50654. *The Effects of Mean Tensile Stresses on High-Cycle Fatigue Life and Strain Accumulation in Some Reactor Materials.* May 1977. Transmitted August 25, 1977.

BNL-NUREG-50661. *HTGR Safety Evaluation Division Quarterly Report, January-March 1977.* 1977. Transmitted August 25, 1977.

LA-6179-MS. *Beginning-of-Life Neutronic Analysis of a 300-MW(L) HTGR.* January 1976. Transmitted August 17, 1977.

LA-6224-MS. *Consequence Analysis of Depressurization Accidents.* February 1976. Transmitted August 17, 1977.

LAPR-NUREG-6317 *Reactor Safety and Technology Quarterly Progress Report January 1-March 31, 1976.* May 1976. Transmitted August 17, 1977.

LA-NUREG-6329-SR. *Status of CHAP: Composite HTGR Analysis Program.* May 1976. Transmitted August 17, 1977.

LA-NUREG-6377-MS. *Seismic Response of a Block-Type Nuclear Reactor Core.* July 1976. Transmitted August 17, 1977.

LA-MUREG-6388-P. *A Proposal for a Seismic Facility for Reactor Safety Research.* July 1976. Transmitted August 17, 1977.

LA-MUREG-6447-PR. *Reactor Safety and Technology Quarterly Progress Report April 1-June 30, 1976.* August 1976. Transmitted August 17, 1977.

LA-MUREG-6473-MS. *A Computer Method for Analyzing HTGR Core Block Response to Seismic Excitation.* September 1976. Transmitted August 17, 1977.

LA-MUREG-6499-MS. *Core Heatup and Fission Product Release from an HTGR Core in an LOFC Accident.* September 1976. Transmitted August 17, 1977.

LA-MUREG-6500. *QUIL: A Chemical Equilibrium Code.* February 1977. Transmitted August 17, 1977.

LA-MUREG-6532-M. *MOVIE.LASL Version 1.0 User's Manual.* December 1976. Transmitted August 17, 1977.

LA-MUREG-6563-MS. *LARC-1: A Los Alamos Release Calculation Program for Fission Product Transport in HTGRs During the LOFC Accident.* November 1976. Transmitted August 17, 1977.

LA-MUREG-6570-MS. *LEAF: A Computer Program to Calculate Fission Product Release from a Reactor Containment Building for Arbitrary Radioactive Decay Chains.* November 1976. Transmitted August 17, 1977.

LA-MUREG-6576-M. *User's Manual for the Composite HTGR Analysis Program (CHAP-1).* March 1977. Transmitted August 17, 1977.

LA-MUREG-6579-PR. *Reactor Safety and Technology Quarterly Progress Report July 1-September 30, 1976.* November 1976. Transmitted August 17, 1977.

LA-MUREG-6639-MS. *An Investigation of Failure Mechanisms for HTGR Core Supports.* December 1976. Transmitted August 17, 1977.

LA-MUREG-6685-MS. *Methods for Calculating Group Cross Sections for Doubly Heterogeneous Thermal Reactor Systems.* February 1977. Transmitted August 17, 1977.

LA-MUREG-6698-PR. *Reactor Safety and Technology Quarterly Progress Report October 1-December 31, 1976.* February 1977. Transmitted August 17, 1977.

LA-MUREG-6768-MS. *A Thermocouple Evaluation Model and Evaluation of Chromel-Alumel Thermocouples for High-Temperature Gas-Cooled Reactor Applications.* April 1977. Transmitted August 17, 1977.

LA-MUREG-6842-PR. *Nuclear Reactor Safety Quarterly Progress Report January 1-March 31, 1977.* June 1977. Transmitted August 17, 1977.

ORNL/NUREG/TM-13. *High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research.* May 1976. Transmitted August 15, 1977.

ORNL/NUREG/TM-16. *HTGR Steam Generator Modeling.* July 1976. Transmitted August 15, 1977.

ORNL/NUREG/TM-38. *BLAST: A Digital Computer Program for the Dynamic Simulation of the High-Temperature Gas-Cooled Reactor Reheater-Steam Generator Module.* June 1976. Transmitted August 15, 1977.

ORNL/NUREG/TM-39. *CORTAP: A Coupled Neutron Kinetics-Heat Transfer Digital Computer Program for the Dynamic Simulation of the High-Temperature Gas-Cooled Reactor Core.* January 1977. Transmitted August 15, 1977.

ORNL/NUREG/TM-43. *High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research Quarterly Progress Report April 1, 1976-June 30, 1976.* September 1976. Transmitted August 15, 1977.

ORNL/NUREG/TM-66. *High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research, Quarterly Progress Report July 1, 1976-September 30, 1976.* December 1976. Transmitted August 15, 1977.

ORNL/NUREG/TM-96. *High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research, Quarterly Progress Report October 1, 1976-December 31, 1976.* March 1977. Transmitted August 15, 1977.

ORNL/NUREG/TM-115. *High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research, Quarterly Progress Report January 1, 1977-March 31, 1977.* June 1977. Transmitted August 15, 1977.

ORNL/TM-5159. *ORECA-1: A Digital Computer Code for Simulating the Dynamics of HTGR Cores for Emergency Cooling Analyses.* April 1976. Transmitted August 15, 1977.

ORGANIZATION OF BIBLIOGRAPHY

The bibliography which follows contains all the LWR safety research reports received by NRC under the foreign exchange agreement program from July through December 1977. The reports received from France and Japan are listed chronologically by report date, and the German reports are listed alphabetically by installation name and then chronologically by report date under each installation. The bibliography is sorted into the following categories:

1. French light-water reactor safety reports received by NRC.
2. German (FRG) light-water reactor safety reports received by NRC.
3. Japanese light-water reactor safety reports received by NRC.

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326-350	12.00
351-375	12.50
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576-600	16.50
601-up	^c

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^b These prices are subject to change by NTIS.

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PARTS AND METHOD OF INDEXING ABSTRACTS

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BIBLIOGRAPHY

1. FRENCH LIGHT-WATER REACTOR SAFETY REPORTS RECEIVED BY NRC

THE FOLLOWING IS A LISTING OF REPORTS RECEIVED FROM FRANCE DURING THE SECOND HALF OF 1977 UNDER THE TECHNICAL EXCHANGE AGREEMENT.

132525

POST AC + SAGLE D

SURVEY OF THE FRENCH DEVELOPMENTS IN THE FIELD OF IN-SERVICE INSPECTION

COMMISSARIAT A L'ENERGIE ATOMIQUE, FRANCE

FRANSA-77 + DSN 132E +. 06 PPS. 20 FIGS. APRIL 25, 1977

IN-SERVICE INSPECTIONS OF REACTOR PRESSURE COMPONENTS ARE MAINLY BASED ON NONDESTRUCTIVE TESTING TECHNIQUES. THE SPECIFIC REQUIREMENTS OF SUCH INSPECTIONS AS WELL AS THE PECULIAR NEED FOR PREVENTING CATASTROPHIC FAILURES ASSOCIATED WITH STRONG RADIOACTIVE RELEASES HAVE LED TO NEW DEVELOPMENTS. IN FRANCE AS IN THE OTHER COUNTRIES, THE EFFORTS WERE GOING ON IN VARIOUS DIRECTIONS. THE FIRST ACTION WAS MADE IN THE REGULATION FIELD, AND IS STILL UNDERWAY. SIMULTANEOUSLY A LARGE R AND D PROGRAM WAS SET-UP TO SOLVE THE PROBLEMS ASSOCIATED WITH THE SPECIFIC TYPE OF WATER REACTORS DEVELOPED IN FRANCE. THE PAPER GIVES AN IDEA OF THE VARIOUS ASPECTS OF THESE EFFORTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, PWR + FRANCE + INSERVICE INSPECTION + EQUIPMENT DEVELOPMENT + R AND D PROGRAM + ACOUSTICS + ULTRASONICS + TEST, NONDESTRUCTIVE + OPERATING EXPERIENCE + INSTRUMENT, COMPONENT + SURVEY + WELDS + FOREIGN EXCHANGE

2. GERMAN (FRG) LIGHT-WATER REACTOR REPORTS RECEIVED BY NRC

THE FOLLOWING IS A LISTING OF REPORTS RECEIVED FROM THE FEDERAL REPUBLIC OF GERMANY DURING THE SECOND HALF OF 1977 UNDER THE TECHNICAL EXCHANGE AGREEMENT.

127052

MODEL CALCULATIONS TO VERIFY THE SHORT-TERM BEHAVIOR OF ABSOLUTE AND DIFFERENTIAL PRESSURES MEASURED IN THE RS 50 EXPERIMENTS, SERIES C
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R.G. GERMANY
GERMAN-121 + OF-MS 50-24-1 +. 112 PPS, FIGS. MARCH 1977

THE ABSOLUTE AND DIFFERENTIAL PRESSURES OCCURRING IN THE INDIVIDUAL COMPARTMENTS UPON RUPTURE OF THE COOLANT PIPE ARE CALCULATED AND COMPARED WITH THE MEASURED DATA. ALL CALCULATIONS WERE CARRIED OUT BY MEANS OF THE DEIFF COMPUTER PROGRAM. THE CALCULATIONS HAVE BEEN FOUND TO YIELD CONSERVATIVE VALUES OF ABSOLUTE AND DIFFERENTIAL PRESSURES IN PARTICULAR FOR THE RUPTURE COMPARTMENT AND THE IMMEDIATELY FOLLOWING COMPARTMENTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104

COMPUTER PROGRAM + ACCIDENT, LOSS OF COOLANT + COMPARISON, THEORY AND EXPERIENCE + FOREIGN EXCHANGE + CONTAINMENT, HIGH PRESSURE

127015

QUICK LOOK REPORT EXPERIMENT D6 - INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS (IN GERMANY)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R.G. GERMANY
GERMAN-112 + OF-MS 50-30-06 +. APPROX. 120 PPS, TABS, FIGS. MAY 1977

A SERIES OF SIMPLIFIED IDEALIZED EXPERIMENTS WERE MADE TO PROVIDE EXPERIMENTAL DATA FOR THE DEVELOPMENT OF MORE ACCURATE COMPUTER CODES. THE PRESSURE BUILD-UP IN THE COMPARTMENT ZONES AND THUS THE PRESSURE DIFFERENCES BETWEEN COMPARTMENTS, WAS REMARKABLY LOWER THAN PREDICTED BY CALCULATIONS PERFORMED WITH STANDARD SETS OF CALCULATIONAL PARAMETERS. THE SHORT TERM BEHAVIOR WAS CHARACTERIZED BY INHOMOGENITIES IN THE COMPOSITION OF THE AIR-VAPOR MIXTURE. EFFORTS TO CLARIFY EXISTING DISCREPANCIES BETWEEN CALCULATIONAL AND EXPERIMENTAL RESULTS MUST CONTINUE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

REACTOR, LWR + CONTAINMENT + COMPARTMENT + COMPUTER PROGRAM + WATER VAPOR + THERMAL TRANSIENT + HEAT TRANSFER + ACCIDENT, LOSS OF COOLANT + COMPARISON, THEORY AND EXPERIENCE + FOREIGN EXCHANGE + PRESSURE TRANSIENT

120450

INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER COOLED REACTORS - COMPLETE EXPERIMENT DOCUMENTATION D4 (IN GERMANY)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R.G. GERMANY
GERMAN-130 + OF-MS 50-32-06 +. APPROX. 120 PPS, FIGS. MAY 1977

EXPERIMENTS OF SERIES D (SIMPLIFIED IDEALIZED EXPERIMENTS) WERE RUN TO PROVIDE DATA FOR THE DEVELOPMENT OF MORE ACCURATE COMPUTER CODES. A COMPARISON OF EXPERIMENTAL RESULTS WITH CALCULATIONS PERFORMED WITH THE CONTAINMENT CODE ZOC6 (IGRS/RUNUCH) SHOWED THE PRESSURE BUILD-UP IN THE COMPARTMENT ZONES AND THE PRESSURE DIFFERENCES BETWEEN COMPARTMENTS WAS--AS IN THE EARLIER EXPERIMENTS--REMARKABLY LOWER THAN PREDICTED BY CALCULATIONS PERFORMED WITH STANDARD SETS OF CALCULATIONAL PARAMETERS. EFFORTS TO CLARIFY THE DISCREPANCIES BETWEEN CALCULATIONAL AND EXPERIMENTAL RESULTS MUST THEREFORE CONTINUE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

CONTAINMENT + COMPARTMENT + REACTOR, LWR + COMPUTER PROGRAM + ACCIDENT, STEAM LINE RUPTURE + COMPARISON, THEORY AND EXPERIENCE + HEAT TRANSFER + PRESSURE, INTERNAL + PRESSURE DROP + PRESSURE TRANSIENT + FOREIGN EXCHANGE

129977

MODEL CALCULATIONS TO VERIFY THE SHORT-TERM BEHAVIOR OF ABSOLUTE AND DIFFERENTIAL PRESSURES MEASURED IN THE RS 50 EXPERIMENTS C13 AND C15 (IN GERMANY)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R.G. GERMANY
GERMAN-120 + OF-MS 50-24-2 +. 67 PPS, TABS, FIGS. MAY 1977

DISCUSSES THE INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS. THE PRESSURE HISTORIES FOR THE FIRST 2 SECONDS AND THE RESULTANT DIFFERENTIAL PRESSURES ARE THEORETICALLY DETERMINED AND COMPARED WITH THE MEASURED VALUES. THE INDIVIDUAL VALUES OF THE CALCULATED ABSOLUTE PRESSURE ARE ALWAYS CONSERVATIVE - IN PARTICULAR FOR THE RUPTURE COMPARTMENT AND THE IMMEDIATELY FOLLOWING COMPARTMENTS. THE MOST CONSERVATIVE VALUES WERE OBTAINED IN THOSE CALCULATIONS WHICH CORRESPOND TO THE DESIGN CALCULATIONS FOR REAL POWER PLANTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

ACCIDENT ANALYSIS + ACCIDENT, LOSS OF COOLANT + CONTAINMENT + PRESSURE TRANSIENT + COMPARISON, THEORY AND EXPERIENCE + ANALYTICAL TECHNIQUE + GERMANY + REACTOR, LWR + A AND B PROGRAM + FOREIGN EXCHANGE

129705

129745 UNCLASSIFIED

TAJAKI, H. • NIEDER, J. • SCHULTZ, K. • WINTER, H.
 STUDY OF THE SIGNIFICANCE AND OF THE FEASIBILITY OF MEASURING THE HEATS OF CHEMICAL REACTIONS DURING CORE
 MELTING, AND OF THE TOTAL INTEGRATED HEAT FROM CORE MELTING (IN GERMAN)
 BATTELLE-INSTITUT F.V., FRANKFURT AM MAIN, F.R.G. GERMANY
 GER-56-133 • UFFY-85-187 • 85 PPS, TABS, FIGS, JUNE 1977

THE FIRST PART OF THIS REPORT IS A DESCRIPTION OF HOW IT WAS DETERMINED THAT UNDER CERTAIN
 CONDITIONS THE CHEMICAL HEAT OF REACTION DURING A CORE MELTDOWN CAN BE SUBSTANTIAL COMPARED TO
 THE DECAY HEAT. THE SECOND PART IDENTIFIES SEVEN REACTIONS REQUIRING EXPERIMENTAL INVESTIGATION.
 THE STEEL MELT - STEAM REACTION, THE METAL MELT - CONTAINMENT ATMOSPHERE REACTION, AND THE
 INTEGRAL MELT-CONCRETE INTERACTION WERE SELECTED FOR EMPHASIS IN THE RESEARCH PROGRAM. MEASURING
 METHODS AND DETAILED SET-UPS FOR EXPERIMENTAL INVESTIGATIONS WERE CONCEIVED FOR THE FIRST TWO
 REACTIONS. THE APPARATUS WERE DESIGNED SUCH THAT THEY CAN ALSO BE USED FOR OTHER INVESTIGATIONS
 ON CHEMICAL REACTIONS DURING CORE MELTDOWN.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, LWR • CORE MELTDOWN • CHEMICAL REACTION • THERMAL EXPERIMENT • ACCIDENT, CONSEQUENCES • OUT OF PILE
 EXPERIMENT • GERMANY • FOREIGN EXCHANGE

129374

INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY
 COOLING CIRCUIT IN WATER-COOLED REACTORS - QUICK LOOK REPORT EXPERIMENT 07 (IN GERMAN)
 BATTELLE-INSTITUT F.V., FRANKFURT AM MAIN, F.R.G. GERMANY
 GER-56-136 • UFFY-85-50-07 • APPROX. 135 PPS, TABS, FIGS, JUNE 1977

THE MAIN RESULTS OF EXPERIMENT 07 WERE CONFIRMED. IN ADDITION THE CALCULATIONS OF 07 AND THE
 ANALYTICAL RESEARCH SHOWN THAT USING THE EXISTING VERSION OF ZOCO 3 IT IS NOT POSSIBLE TO
 ACHIEVE FULL AGREEMENT BETWEEN MEASURED AND CALCULATED PRESSURES WITH PHYSICALLY REALISTIC INPUT
 VALUES FOR THE HEAT TRANSFER COEFFICIENT. A MORE PROFOUND ANALYSIS OF THE INTERCOMPARTMENT FLOW
 IS NECESSARY FOR IMPROVING THE FLOW MODELS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CONTAINMENT • COMPARTMENT • REACTOR, LWR • COMPUTER PROGRAM • HEAT TRANSFER COEFFICIENT • COMPARISON, THEORY
 AND EXPERIENCE • PRESSURE TRANSIENT • ACCIDENT, STEAM LINE RUPTURE • FOREIGN EXCHANGE

131327

INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY
 COOLING CIRCUIT IN WATER-COOLED REACTORS - COMPLETE EXPERIMENT DOCUMENTATION 07 (IN GERMAN)
 BATTELLE-INSTITUT F.V., FRANKFURT AM MAIN, F.R.G. GERMANY
 GER-56-136 • UFFY-85-50-07 • APPROX. 125 PPS, FIGS, JUNE 1977

FOLLOWING THE EXPERIMENT CALCULATIONS USING THE CONTAINMENT CODE ZOCO 3 WERE MADE, AND ALSO
 ANALYTICAL RESEARCH WAS DONE USING A NEW ESTABLISHED PROGRAM VERSION OF ZOCO 6. COMPARISON OF
 CALCULATIONAL AND EXPERIMENTAL RESULTS SHOWED THAT IT IS NOT POSSIBLE TO ACHIEVE FULL AGREEMENT
 BETWEEN MEASURED AND CALCULATED PRESSURES WITH PHYSICALLY REALISTIC INPUT VALUES FOR THE HEAT
 TRANSFER COEFFICIENT. THE COMPARISON ALSO SHOWED THE NECESSITY OF A MORE PROFOUND ANALYSIS OF
 INTERCOMPARTMENT FLOW IN VIEW OF IMPROVEMENT OF FLOW MODELS IN CONTAINMENT PROGRAMS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CONTAINMENT • COMPARTMENT • REACTOR, LWR • COMPUTER PROGRAM • HEAT TRANSFER COEFFICIENT • ACCIDENT, STEAM LINE
 RUPTURE • COMPARISON, THEORY AND EXPERIENCE • PRESSURE, INTERNAL • PRESSURE DROP • PRESSURE TRANSIENT •
 FOREIGN EXCHANGE

131560

INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY
 COOLING CIRCUIT IN WATER-COOLED REACTORS - QUICK LOOK REPORT EXPERIMENT 03 (IN GERMAN)
 BATTELLE-INSTITUT F.V., FRANKFURT AM MAIN, F.R.G. GERMANY
 GER-56-136 • UFFY-85-50-03 • 175 PPS, TABS, FIGS, SEPT. 1977

THE CONTAINMENT EXPERIMENTS OF SERIES D ARE TO BE CARRIED OUT UNDER SIMPLIFIED IDEALIZED
 CONDITIONS TO PROVIDE EXPERIMENTAL DATA FOR IMPROVING AVAILABLE CONTAINMENT CODES. IMPORTANT
 PARAMETERS OF EXPERIMENT 03 ARE DESCRIBED. PRECALCULATIONS AND POST-EXPERIMENT CALCULATIONS HAVE
 BEEN CARRIED OUT FOR EXPERIMENT 03. THE RESULTS OF A COMPARISON WITH EXPERIMENTAL DATA ARE GIVEN.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CONTAINMENT • PRESSURE TRANSIENT • ACCIDENT, LOSS OF COOLANT • COMPUTER PROGRAM • REACTOR, LWR • FOREIGN
 EXCHANGE

129825

LANGHEIM, H.
 IMPACT TESTS WITH REINFORCED CONCRETE SLABS, PART I (IN GERMAN)

127825 (CONTINUED)

ERNST-RACH INSTITUT, GERMANY
GERMAN-184 + ERN-2 1670 + RS-102-87-3 +. 40 PPS, 5 TABS, 26 FIGS, 3 REFS, JUNE 1975

DESCRIBES THE DEVELOPMENT AND EVALUATION OF IMPACT STUDIES ON 93 REINFORCED CONCRETE SLABS. CYLINDRICAL, SOLID C 110 W 1 - STEEL PROJECTILES WITH CALIBERS 15, 20 AND 40 MM WERE FIRED WITH VELOCITIES RANGING FROM 100 UP TO 2000 M/S ON 700 X 700 MM SIZED REINFORCED CONCRETE SLABS OF 200 UP TO 400 MM THICKNESS. THE MEASURED PENETRATION DEPTHS ARE COMPARED WITH THE PENETRATION DEPTHS OF THE CLASSIC PENETRATION FORMULA (PPTN, MLC, COE AND M-HADAC). DEPENDENT UPON PROJECTILE CALIBER AND VELOCITY RANGE EITHER THE PPTN OR THE M-HADAC-FORMULA SUPPLY SATISFACTORY CONSISTENCY WITH NUM TEST RESULTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CONCRETE, REINFORCED + IMPACT PROPERTY + MISSILE GENERATION AND PROTECTION + GERMANY + COMPARISON, THEORY AND EXPERIENCE + FOREIGN EXCHANGE

127837

LANGHEIM H
IMPACT TESTS WITH REINFORCED CONCRETE SLABS. PART II (IN GERMAN)
ERNST-RACH INSTITUT, GERMANY
GERMAN-185 + ERN-2 1675 + RS-102-87-4 +. 56 PPS, 4 TABS, 23 FIGS, 3 REFS, OCT. 1975

CYLINDRICAL, SOLID STEEL PROJECTILES (MATERIAL C 110 W 1) WITH CALIBERS 15, 20 AND 30 MM WERE FIRED WITH VELOCITIES FROM 100 UP TO 1700 M/S ON 700 X 700 MM SIZED REINFORCED CONCRETE SLABS OF 75 UP TO 200 MM THICKNESS. THE EVALUATION OF THE NORMALIZED PENETRATION DEPTHS IN DEPENDENCE OF THE PROJECTILE VELOCITIES AND ALSO OF THE CRATER VOLUMES AS A FUNCTION OF THE KINETIC ENERGY OF THE PROJECTILES CONTAIN THE RESULTS OF PREVIOUS STUDIES. FOR THE SAME PROJECTILE VELOCITIES L70 + 1 AND L70 + 2 RESP., THE RESULTING VALUES WITHIN EACH RANGE OF CONCRETE STRENGTHS ARE SITUATED EACH IN A CURVE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CONCRETE, REINFORCED + MISSILE GENERATION AND PROTECTION + TESTING + FOREIGN EXCHANGE

131557

ANNUAL REPORT ON REACTOR SAFETY RESEARCH PROJECTS SPONSORED BY THE MINISTRY FOR RESEARCH AND TECHNOLOGY OF THE FEDERAL REPUBLIC OF GERMANY
GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) GMBH, F.R.G, GERMANY
GRS-F-36 + GERMAN-183 +. 676 PPS, TABS, FIGS, 1976

OBJECTIVE OF THIS PROGRAM IS TO INVESTIGATE IN GREATER DETAIL THE SAFETY MARGINS OF NUCLEAR ENERGY PLANTS AND THEIR SYSTEMS AND THE FURTHER DEVELOPMENT OF SAFETY TECHNOLOGY. THIS COMPILATION OF PROGRESS REPORTS IS THE FIFTH ONE TO BE PUBLISHED ONCE A YEAR IN THE FRAME OF A COMPREHENSIVE REPORTING ABOUT THE CURRENT INVESTIGATIONS WITH REGARD TO REACTOR SAFETY. THIS COMPILATION, IN ADDITION TO THE LOW-INVESTIGATIONS, ALSO CONTAINS FIRST CONTRIBUTIONS ON THE SAFETY OF ADVANCED REACTORS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

GERMANY + OR AND O PLUGMAN + SAFETY ANALYSIS + BLUDDOWN + EMERGENCY COOLING SYSTEM + THERMAL HYDRAULIC ANALYSIS + REACTIVITY EFFECT + CORE MELTDOWN + FUEL COOLANT INTERACTION + EXPLOSION + FISSION PRODUCT TRANSPORT + FISSION PRODUCT RELEASE + STRUCTURAL INTEGRITY + ACCIDENT, PROBABILITY OF + FOREIGN EXCHANGE + MISSILE GENERATION AND PROTECTION

125904

ULLRICH R + ROMKE J + MULLER H + NICKEN SF
ANALYTICAL RESULTS OF THE NON-NUCLEAR LOFT TESTS L1 AND L2. PROGRESS REPORT (IN GERMAN)
GESELLSCHAFT FÜR REAKTORSICHERHEIT GRS, F.R.G, GERMANY
GERMAN-182 + GRS-A-7 +. 181 PPS, TABS, FIGS, 15 REFS, FEB. 1977

EXPERIMENTAL RESULTS FROM LOFT TESTS L1 AND L2 ARE COMPARED WITH ANALYTICAL PREDICTIONS OF THE CODES WHARMOD AND RELAP-4/GRS. THE AIM IS TO INTERPRET THE TEST SEQUENCE OF EVENTS, AND TO CHECK THE CAPABILITIES OF THE TWO CODES. VARIOUS PROBLEMS WITH BOTH EXPERIMENTAL DATA AND THE CODES ARE DISCUSSED, ESPECIALLY THE LOCALIZATION USED IN THE CODES. IN SPITE OF THESE PROBLEMS, IT IS DETERMINED THAT RELAP-4 IS CAPABLE OF DESCRIBING DECOMPRESSION AFTER SEVERE PIPE BREAKS FOR THIS CONFIGURATION.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ACCIDENT, LOSS OF COOLANT + REACTOR, PWR + BLUDDOWN + OUT OF PILE EXPERIMENT + COMPARISON, THEORY AND EXPERIENCE + COMPUTER PROGRAM, DIGITAL + GERMANY + LOFT (S-RR) + FOREIGN EXCHANGE

132893

LIST OF REPORTS FROM THE REACTOR SAFETY RESEARCH PROGRAMS OF BMTF AND USNRG FOR THE PERIOD JANUARY 1-MARCH 31, 1977 (IN GERMAN & ENGLISH)

132933 - CONTINUED
 GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R.G. GERMANY
 GRS-R-33 + GRS-R-114 + 57 PPS, MAY 1977

PRESENTS A LIST OF REPORTS FROM THE FEDERAL REPUBLIC OF GERMANY AND FROM THE UNITED STATES CONCERNING SPECIAL PROBLEMS IN THE FIELD OF REACTOR SAFETY RESEARCH WHICH HAVE BEEN COLLECTED IN THE GESELLSCHAFT FÜR REAKTORSICHERHEIT.

AVAILABILITY - NASHUAARD, INC., P.O. BOX 101, YAN RIDGE, TENN. 37030

GERMANY + UNITED STATES + R AND D PROGRAM + SAFETY PROGRAM + AGENCY, NRC + FOREIGN EXCHANGE

132913
 REPORT ON RESEARCH SPONSORED BY THE FEDERAL MINISTRY OF RESEARCH AND TECHNOLOGY ON THE SUBJECT OF REACTOR SAFETY: 4 UNIT PERIOD JANUARY 1-MARCH 31, 1977 (IN GERMAN)
 GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R.G. GERMANY
 GRS-R-101 + GRS-R-40 + 304 PPS, TABS, FIGS, JULY 1977

THE OBJECTIVE OF THIS PROGRAM IS TO INVESTIGATE IN GREATER DETAIL THE SAFETY MARGINS OF NUCLEAR ENERGY PLANTS AND THEIR SYSTEMS AND THE FURTHER DEVELOPMENT OF SAFETY TECHNOLOGY. EACH PROGRESS REPORT REPRESENTS A COMPILATION OF INDIVIDUAL REPORTS ABOUT THE DIFFERENT PROJECTS OF THE RESEARCH PROGRAM. THE INDIVIDUAL REPORTS ARE PREPARED BY THE CONTRACTORS AS A DOCUMENTATION OF THEIR PROGRESS. EACH REPORT DESCRIBES THE WORK PERFORMED, THE RESULTS AND THE NEXT STEP OF THE WORK.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

GERMANY + R AND D PROGRAM + REACTOR, LBR + SAFETY ANALYSIS + SAFETY MARGIN

132995
 DESCRIPTION OF THE CURRENT PRACTICE IN THE DEVELOPMENT OF SAFETY CRITERIA FOR NUCLEAR POWER PLANTS. SAFETY CRITERIA PART 2: IMPLICATION OF BASIC SAFETY REQUIREMENT (IN GERMAN)
 GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R.G. GERMANY
 GRS-R-103 + 19 PPS, 17 REFS, JULY 1977

DESCRIBES THE BASIC SAFETY REQUIREMENTS IN GERMANY FOR NUCLEAR POWER PLANTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

GERMANY + POWER PLANT, NUCLEAR + DESIGN CRITERIA + SAFETY PRINCIPLES AND PHILOSOPHY + FOREIGN EXCHANGE

131561
 UNUSUAL OCCURRENCES IN NUCLEAR POWER PLANTS IN THE FEDERAL REPUBLIC OF GERMANY IN THE YEARS 1965-1977 (IN GERMAN)
 GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R.G. GERMANY
 GRS-R-107 + 17 PPS, JULY 1977

SUMMARIZES THE UNUSUAL OCCURRENCES THAT HAVE OCCURRED IN NUCLEAR POWER PLANTS IN THE FEDERAL REPUBLIC OF GERMANY DURING THE PERIOD 1/1/65-12/31/76. INCLUDES THE NAME OF THE REACTOR AND A BRIEF DESCRIPTION OF THE OCCURRENCE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

PENDING COMPILATION + GERMANY + REACTOR, POWER + REACTOR, BUR + REACTOR, PUR + FAILURE, EQUIPMENT + FAILURE, INSTRUMENT + FOREIGN EXCHANGE

131558
 FRANZEN L. + GUTSCHMIDT W.
 PRIORITY LISTING OF TECHNICAL SAFETY ASPECTS OF NUCLEAR POWER PLANTS (IN GERMAN)
 GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R.G. GERMANY
 GRS-S-37 + GRS-R-109 + 62 PPS, 5 TABS, 21 FIGS, 17 REFS, JULY 1977

DISCUSSES THE CONSTRUCTION, DESIGN, AND OPERATION OF NUCLEAR POWER PLANTS INCLUDING STANDARD REVIEWS AND PLANTS AND COMPARISONS WITH THE UNITED STATES POWER PLANTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

GERMANY + REVIEW + POWER PLANT, NUCLEAR + UNITED STATES + CONSTRUCTION + OPERATION + LICENSING PROCESS + FOREIGN EXCHANGE

131559
 MELLER M. + MAY M.
 NUCLEAR POWER PLANT RISKS, UNIT A: DESCRIPTION OF EXPERIENCES IN 1975 & 1976 (IN GERMAN)
 GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R.G. GERMANY

131554 CONTINUED
GAS-3-10 • GERAS-150 • 17 PPS, 9 FIGS, 2 REFS, JULY 1977

REVIEWS THE MAJOR OCCURRENCES EXPERIENCED WITH THE PRIMARY AND SECONDARY COOLING CIRCUIT FOR THE 2 YEAR PERIOD.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104

REACTOR, PWR • GERMANY • POWER PLANT, NUCLEAR • OPERATING EXPERIENCE • OPERATING EXPERIENCE SUMMARY • FOREIGN EXCHANGE

130007
CHECKLIST WITH LAYOUT OF A STANDARD SAFETY ANALYSIS REPORT FOR NUCLEAR POWER PLANTS WITH PRESSURIZED WATER REACTOR (PWR) BOILING WATER REACTOR (BWR)
GESELLSCHAFT FÜR REAKTORSICHERHEIT, F.R.G. GERMANY
GERAS-100 • 74 PPS, TRANSLATIONS-SAFETY CODES & GUIDES, EDITION 10/77, 1977

THE CHECKLIST IS TO BE USED FOR THE PREPARATION AND REVIEW OF SAFETY ANALYSIS REPORTS; FOLLOWING USAGE AND IMPROVEMENTS, IT WILL BECOME A REGULATION. WHILE THE CHECKLIST APPLIES TO STANDARD BWR AND PWR TYPE REACTORS IT CAN BE USED FOR OTHER TYPES. (THE PRELIMINARY NATURE OF THE CHECKLIST IS REFLECTED BY THE INTRODUCTION WHICH ENCOURAGES INTERESTED PARTIES TO SUBMIT SUGGESTIONS FOR IMPROVEMENTS.)

AVAILABILITY - GESELLSCHAFT FÜR REAKTORSICHERHEIT, GLUCKENGASSE 2, D-5000 KÖLN 1, F.R.G. GERMANY

GERMANY • PWR/BWR • REPORT, SAR • REACTOR, PWR • REACTOR, BWR • PROCEDURES AND MANUALS • REPORT, PSAR • FOREIGN EXCHANGE

130008
RECOMMENDATIONS FOR THE PLANNING OF EMERGENCY PROTECTIVE MEASURES BY THE LICENSEES OF NUCLEAR POWER PLANTS
GESELLSCHAFT FÜR REAKTORSICHERHEIT, F.R.G. GERMANY
GERAS-100 • 74 PPS, TRANSLATIONS-SAFETY CODES & GUIDES, EDITION 10/77, 1977

PROVIDES GUIDANCE FOR THE PREPARATION OF EMERGENCY PROTECTIVE MEASURES APPLICABLE TO THE PERSONNEL, ADMINISTRATIVE, AND ENGINEERING LEVELS. THE ALERT PLAN CONSIDERS RESPONSIBILITIES, THE REPORTING OF ALERTS, ALERTS AND THEIR MEANINGS, AND CRITERIA FOR SEVERAL ALERTS. PERSONNEL BEHAVIOR, TRAINING, AND RECORDS ARE ALSO CONSIDERED.

AVAILABILITY - GESELLSCHAFT FÜR REAKTORSICHERHEIT, GLUCKENGASSE 2, D-5000 KÖLN 1, F.R.G. GERMANY

GUIDE • PROTECTIVE ACTION GUIDE • EMERGENCY PROCEDURE • POWER PLANT, NUCLEAR • TRAINING • RECORDS • ACCIDENT • GERMANY • FOREIGN EXCHANGE

130009
COMPILATION OF INFORMATION ON THE REACTOR PROTECTION SYSTEM REQUIRED FOR EXAMINATION PURPOSES IN THE LICENSING PROCEDURES OF NUCLEAR POWER PLANTS
GESELLSCHAFT FÜR REAKTORSICHERHEIT, F.R.G. GERMANY
GERAS-100 • 7 PPS, TRANSLATIONS-SAFETY CODES & GUIDES, EDITION 12/77, 1977

PRESENTS A 7 PAGE LISTING OF THE TYPE OF INFORMATION REQUIRED FOR THE LICENSING PROCEDURES UNDER THE GERMAN ATOMIC ENERGY ACT FOR SAFETY EVALUATION OF NUCLEAR POWER PLANTS.

AVAILABILITY - GESELLSCHAFT FÜR REAKTORSICHERHEIT, GLUCKENGASSE 2, D-5000 KÖLN 1, F.R.G. GERMANY

GERMANY • LICENSING PROCESS • SAFETY EVALUATION • REACTOR PROTECTION SYSTEM • POWER PLANT, NUCLEAR • GUIDE • FOREIGN EXCHANGE

131755
RECENT WORLDWIDE DEVELOPMENTS IN RADIOLOGICAL PROTECTION FROM THE UTILIZATION OF NUCLEAR TECHNOLOGY
INSTITUT FÜR REAKTORSICHERHEIT DER TU-BERLIN, F.R.G. GERMANY
GERAS-100 • 1P-7-29 • 163 PPS, TABS, FIGS, DEC. 1976

PRESENTS THE TEXT OF 7 PAPERS (AND 5 DISCUSSION PERIODS) PRESENTED AT THE SYMPOSIUM ON THE SUBJECT OF RADIOLOGICAL PROTECTION.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104

RADIATION SAFETY AND CONTROL • POPULATION EXPOSURE • GERMANY • RADIOLOGICAL IMPACT • EFFLUENT • FOOD CHAIN • FOREIGN EXCHANGE

129003
THEORETICAL AND EXPERIMENTAL EXAMINATION REGARDING THE CARRYING CAPACITY OF HYDRODYNAMIC CYLINDRICAL RADIAL BEARINGS
KLEIN, SCHANZLIN & BECKER, GERMANY

127051 CONTINUOUS
GERNSD-120 + DMT 70/4 02 00 100 PPS, 7 TABS, FIGS, NO DATE

PRESENTS THE RESULTS OF A THEORETICAL AND EXPERIMENTAL EXAMINATION REGARDING THE CARRYING CAPACITY OF HYDRODYNAMIC CYLINDRICAL BEARINGS WITH NON-LAMINAR CLEARANCE FLOW OF THE LUBRICANT. ARE SUMMARIZED: THE BEARING TEST STAND, THE TEST METHOD, THE MATERIAL OF THE SLIDING FACES, THE LIMITING OPERATION CONDITIONS AND THE ALGORITHM REQUIRED FOR THE HANDLING OF EXPERIMENTAL DATA ARE DESCRIBED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

HYDRODYNAMIC ANALYSIS + BEARING + TESTING + NUMERICAL METHOD + DATA COLLECTION + WEAR + FOREIGN EXCHANGE + COMPARISON, THEORY AND EXPERIENCE

127054
FRIEDL W + MITCHELL W + SCHULZE R
FINAL REPORT ON RESEARCH PROJECT RS-77: RESEARCH ON THE THERMODYNAMICS OF TWO PHASE FLOW (IN GERMAN)
COMMISSION OF THE EUROPEAN COMMUNITIES
RS-77 + GERNSD-121 00 130 PPS, FIGS, 1976

DETAILED DESCRIPTION OF THE TEST APPARATUS WITH THE REPORT CONSISTING PRIMARILY OF DATA IN THE FORM OF GRAPHS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

GERMANY + A AND J PROGRAM + THERMODYNAMICS + FLOW, TWO PHASE + FOREIGN EXCHANGE + FLOW THEORY AND EXPERIMENTS

127057
WILHELM W + KUTNER W + FISCHER F
EXPERIMENTAL INVESTIGATIONS OF VAPOR EXPLOSIONS (IN GERMAN)
COMMISSION OF THE EUROPEAN COMMUNITIES
WILHELM W + GERNSD-127 00 50 PPS, 27 FIGS, 4 APPS, MARCH 1977

DESCRIBES THE EXPERIMENT AND APPARATUS USED TO STUDY THE ENERGY RELEASE OF FUEL COOLANT INTERACTION.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

GERMANY + A AND J PROGRAM + FUEL COOLANT INTERACTION + EXPLOSION + WATER VAPOR + FOREIGN EXCHANGE

127065
WILHELM W + FISCHER F
RESEARCH ON THE SELECTIVE CORROSION BEHAVIOR OF LIGHT WATER REACTOR MATERIALS PART 1: LITERATURE STUDY OF CRACKING AND PITTING CORROSION AND CORROSION FATIGUE. (IN GERMAN)
KRAFTWERK UNION, ERLANGEN, F.R.G. GERMANY
GERNSD-122 + DMT RS 150 00 100 PPS, 4 TABS, 27 FIGS, 179 REFS, FEB. 1977

IN ADDITION TO LOW-ALLOY, PRESSURE-VESSEL STEELS, HIGHLY ALLOYED, AUSTENITIC, FERRITIC AND MARTENSITIC STAINLESS STEELS AND NICKEL-BASED ALLOYS ARE USED TO A CONSIDERABLE EXTENT IN LIGHT-WATER REACTORS. THESE MATERIALS CAN BE ENDANGERED UNDER CERTAIN CONDITIONS THROUGH SELECTIVE CORROSION. PRIOR TO PLANNING AN EXPERIMENTAL PROGRAM, THE PRESENT STATE OF KNOWLEDGE IN THIS AREA HAS BEEN REVIEWED. THE STUDY CONCERNS THE INFLUENCE OF INDIVIDUAL PARAMETERS ON SAFETY MARGINS, WITH RESPECT TO THE CONDITIONS UNDER WHICH CRACKING CORROSION, PITTING CORROSION AND CORROSION FATIGUE OF REACTOR MATERIALS CAN BE EXPECTED TO OCCUR.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

SAFETY MARGIN + STEEL + STEEL, STAINLESS + CORROSION + FATIGUE + REACTOR, LW + FOREIGN EXCHANGE + STEEL, FERRITIC

127069
ROSLER W
RESEARCH ON THE SELECTIVE CORROSION BEHAVIOR OF LIGHT WATER REACTOR MATERIALS. PART 2 - LITERATURE STUDY OF STRESS-INDUCED CORROSION, INTERGRANULAR STRESS CORROSION AND HYDROGEN INDUCED STRESS-CORROSION CRACKING (IN GERMAN)
KRAFTWERK UNION, ERLANGEN, F.R.G. GERMANY
GERNSD-123 + DMT RS 159 00 90 PPS, 7 TABS, 30 FIGS, 156 REFS, MARCH 1977

PRIOR TO PLANNING AN EXPERIMENTAL PROGRAM, THE PRESENT STATE OF KNOWLEDGE HAS BEEN REVIEWED. THE STUDY CONCERNS THE INFLUENCE OF INDIVIDUAL PARAMETERS ON SAFETY MARGINS, WITH RESPECT TO THE CONDITIONS UNDER WHICH STRESS-INDUCED CORROSION, INTERGRANULAR STRESS CORROSION CRACKING AND HYDROGEN-INDUCED STRESS CORROSION CRACKING OF REACTOR MATERIALS CAN BE EXPECTED TO OCCUR.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101

CORROSION + REACTOR, LW + MATERIAL + CRACK + STRESS CORROSION + HYDROGEN + FOREIGN EXCHANGE

127057
 ROSSLER U
 RESEARCH ON THE SELECTIVE CORROSION BEHAVIOR OF LIGHT-WATER REACTOR MATERIALS. PART 3 - RESEARCH PROGRAM (IN GERMAN)
 KRAFTWERK UNION, ERLANGEN, F.R.G. GERMANY
 GERMSP-124 + DWT PS 159 +. 27 PPS. 10 FIGS. MARCH 1977

AN EXPERIMENTAL PROGRAM IS DESCRIBED, BY MEANS OF WHICH QUESTIONS CONCERNING THE DETERMINATION OF SAFETY MARGINS WITH REGARD TO CRACK CORROSION, PITTING CORROSION, CORROSION FATIGUE AND STRESS CORROSION CRACKING WILL BE RESOLVED. THE EXTENT OF KNOWLEDGE PRESENTLY AVAILABLE FOR ASSESSMENT OF SAFETY MARGINS, WITH RESPECT TO AN UNACCEPTABLE DEGREE OF SELECTIVE CORROSION, WAS REVIEWED IN PARTS 1 AND 2 OF THIS REPORT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101
 CORROSION + REACTOR, LWR + MATERIAL + CRACK + STRESS CORROSION + SAFETY MARGIN + FOREIGN EXCHANGE

127058
 ROSSLER U
 RESIDUAL STRESSES IN AUSTENITIC STRIPCLAD REACTOR PRESSURE VESSEL STEEL (IN GERMAN)
 KRAFTWERK UNION, ERLANGEN, F.R.G. GERMANY
 GERMSP-125 + DWT PS 94 +. 179 PPS. 50 FIGS. 70 REFS. MARCH 1977

USING AN EXPERIMENTAL-MATHEMATICAL SECTIONING TECHNIQUE, RESIDUAL STRESSES WERE DETERMINED IN THE AUSTENITICALLY STRIPCLAD PRESSURE VESSEL STEEL 22 NUMBER 37 (ASTM A 508 CL. 2) AS A FUNCTION OF THE FOLLOWING PARAMETERS: THICKNESS OF BASE- AND CLADDING MATERIAL, HEAT INPUT DURING CLADDING, PARENT AND CLADDING MATERIAL, AND HEAT TREATMENT OF THE CLADDED SAMPLE PLATES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101
 PRESSURE VESSELS + STEEL, STAINLESS + STRESS + WELDING + CLADDING + FOREIGN EXCHANGE

127060
 ROSSLER U + MOFFEN G + MEISCH M
 RESIDUAL STRESSES IN AUSTENITIC STRIPCLAD REACTOR PRESSURE VESSEL STEEL. PART 2 - RESIDUAL STRESSES IN THE VICINITY OF BONDED STRIP CLADDING (IN GERMAN)
 KRAFTWERK UNION, ERLANGEN, F.R.G. GERMANY
 GERMSP-126 + DWT PS 91 +. 108 PPS. 9 TABS. 49 FIGS. 16 REFS. APRIL 1977

ACTIVITIES IN PART 1 WERE CONCENTRATED IN MEASURING RESIDUAL STRESSES IN AUSTENITICALLY STRIP-CLADDED MATERIAL WITH THE AID OF THE EXPERIMENTAL-MATHEMATICAL SECTIONING TECHNIQUE. PART 2 WAS CONCERNED WITH: COMPUTATION OF RESIDUAL STRESSES IN A LARGE RPV MULTIPASS WELD WITH THE FINITE ELEMENT METHOD (FEM), MEASUREMENT OF RESIDUAL STRESSES IN A LARGE RPV MULTIPASS WELD WITH THE CORE DRILLING METHOD, AND CALIBRATION EXPERIMENTS FOR X-RAY STRESS ANALYSIS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101
 PRESSURE VESSELS + STEEL, STAINLESS + STRESS + WELDING + CLADDING + STRESS ANALYSIS + X-RAY + FOREIGN EXCHANGE

127071
 REED M
 REDUCTION OF PRIMARY SYSTEM CONTAMINATION THROUGH THE INSTALLATION OF ELECTROMAGNETIC FILTERS (IN GERMAN)
 KRAFTWERK UNION, ERLANGEN, F.R.G. GERMANY
 GERMSP-127 + DWT PS 171 +. 91 PPS. 14 TABS. 19 FIGS. 13 REFS. APRIL 1977

WITH RESPECT TO THE REDUCTION OF PWR PRIMARY SYSTEM CONTAMINATION THE EFFECTIVENESS OF A HIGH-TEMPERATURE ELECTROMAGNETIC FILTER DEVICE FOR REMOVAL OF INACTIVE AND RADIOACTIVE CORROSION PRODUCTS FROM PRIMARY COOLANT HAS BEEN INVESTIGATED. THE TEST HAVE BEEN PERFORMED IN A PRESSURIZED WATER LOOP AT PWR NORMAL OPERATION CONDITIONS. THE RESULTS OBTAINED HAVE SHOWN NO SIGNIFICANT PURIFICATION EFFECTIVENESS WITH LOW FE CONCENTRATIONS SIMILAR TO THOSE PRESENT IN A PWR MAIN COOLANT AT CONSTANT LOAD OPERATION; RADIOACTIVE TRACERS ARE PARTLY REMOVED UNDER THESE CONDITIONS. WITH HIGHER FE CONCENTRATIONS (> 20 PPM) DECONTAMINATION FACTORS IN THE ORDER OF 4 FOR BOTH INACTIVE AND RADIOACTIVE CORROSION PRODUCTS HAVE BEEN OBSERVED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101
 REACTOR, PWR + FILTER, HIGH TEMPERATURE + MAIN COOLING SYSTEM + CORROSION + TEST, FILTER + CONTAMINATION + IRON + FOREIGN EXCHANGE

127023
 HARTNER M + NIECKAU E + SPINDLER M
 RESULTS OF A RELIABILITY ANALYSIS OF THE ENGINEERED SAFETY FEATURES FOR THE DESIGN BASIS ACCIDENT (IN GERMAN)
 TECHNISCHE UNIVERSITÄT MÜNCHEN, GARCHING, F.R.G. GERMANY
 GERMSP-118 + MM-100 +. 100 PPS. 8 TABS. 25 FIGS. 61 REFS. DEC. 1976

127623 *CONTINUED*

GIVES A COMPREHENSIVE PRESENTATION OF THE DETAILED RELIABILITY INVESTIGATION CARRIED OUT FOR THE ENGINEERED SAFETY FEATURES INSTALLED TO Cope WITH THE DESIGN BASIS ACCIDENT "LARGE LOCA" OF A GERMAN NUCLEAR POWER PLANT WITH PRESSURIZED WATER REACTOR. THE INVESTIGATION IS BASED ON THE ENGINEERED SAFETY FEATURES OF THE DUEBES NUCLEAR POWER PLANT, UNIT A. THE RELIABILITY INVESTIGATION IS CARRIED OUT BY MEANS OF A FAULT TREE ANALYSIS. THE INFLUENCE OF COMMON-MODE FAILURES IS ASSESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

RELIABILITY ANALYSIS * ENGINEERED SAFETY FEATURE * ACCIDENT, LOSS OF COOLANT * ACCIDENT, DESIGN BASIS * GERMAN * REACTOR, PWR * FAULT TREE ANALYSIS * FAILURE, COMMON MODE * FOREIGN EXCHANGE

128496

MADICAMPST *

SIMULATION EXPERIMENTS OF THE FUNCTIONAL TESTING OF ENGINEERED SAFETY SYSTEMS WITH PROCESS CALCULATIONS. PART 2: RESULTS OF SIMULATION EXPERIMENTS WITH A NEW CONCEPT OF COMPUTERIZED SYSTEM INSPECTION TECHNIQUE (IN GERMAN)

TECHNISCHE UNIVERSITÄT MÜNCHEN, F.R.G. GERMANY

GERNS-142 * NUM-169 *. 173 PPS, 66 FIGS, 66 REFS, DEC. 1976

COMPUTERIZED INSPECTION TECHNIQUES OF ENGINEERED SAFETY SYSTEMS IMPROVE THE DIAGNOSIS CAPABILITY, RELATIVE TO THE PRESENTLY USED TECHNIQUES. EVEN IF ANTICIPATED SYSTEM DISTURBANCES CAN ONLY BE QUALITATIVELY PREDICTED. TO ACHIEVE THIS, THE SYSTEM TO BE INSPECTED MUST BE PARTITIONED INTO SMALL SUBSYSTEMS, WHICH CAN BE TREATED INDEPENDENTLY FROM EACH OTHER. THIS REPORT CONTAINS THE FORMULATION OF A STANDARDIZED INSPECTION CONCEPT BASED ON SYSTEM DECOMPOSITION. ITS PERFORMANCE IS DISCUSSED BY MEANS OF SIMULATION EXPERIMENTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ENGINEERED SAFETY FEATURE * SIMULATION * TESTING * TEST, SYSTEM OPERABILITY * COMPUTER CONTROL * INSERVICE INSPECTION * FOREIGN EXCHANGE

128392

MENZOG J * ZELINSKY G * CHRIST H * SCHNEIDER K

DECONTAMINATION AND TRANSPORTATION PROBLEMS RELATED TO THE DECOMMISSIONING OF NUCLEAR POWER PLANTS AND THE ELIMINATION OF FAILURE CONSEQUENCES. PART FOUR, VOL. 1: DECOMMISSIONING, JAN. 1, 1975-JUNE 30, 1976

TRANSNUKLEAR GMBH, F.R.G. GERMANY

GERNS-128 * NUM-312 *. 127 PPS, 20 TABS, 23 FIGS, 20 REFS, MAY 1977

THIS FIRST VOLUME OF THE MAIN WORK BEGINS WITH A REVIEW OF THE POSSIBLE FACILITIES FOR DECONTAMINATION AND CONDITIONING, AND THE SHIPMENT OPTIONS RELATED TO THE NORMAL DECOMMISSIONING OF LARGE NUCLEAR POWER PLANTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

*TRANSPORTATION AND HANDLING * DECOMMISSIONING * REGULATION, IAEA * DECONTAMINATION * FOREIGN EXCHANGE

128393

MENZOG J * ZELINSKY G * CHRIST H * SCHNEIDER K

DECONTAMINATION AND TRANSPORTATION PROBLEMS RELATED TO THE DECOMMISSIONING OF NUCLEAR POWER PLANTS AND THE ELIMINATION OF FAILURE CONSEQUENCES. (APPENDIX 1) - PRINCIPLES, JAN. 1, 1975-JUNE 30, 1976

TRANSNUKLEAR GMBH, F.R.G. GERMANY

GERNS-129 * NUM-312 *. 61 PPS, 10 TABS, 7 FIGS, MAY 1977

THE TECHNICAL CONDITIONS WHICH MUST BE CLEARED IN ORDER TO OVERCOME THE DECONTAMINATION AND SHIPMENT PROBLEMS ARISING FROM THE DECOMMISSIONING OF NUCLEAR POWER PLANTS AT THE END OF THEIR SERVICE LIFE AND FROM THE ELIMINATION OF SYSTEM FAILURE CONSEQUENCES IN SUCH PLANTS, ARE COMPILED AND SUBJECTED TO A PRELIMINARY ASSESSMENT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

SHIPPING CONTAINER * DECOMMISSIONING * FOREIGN EXCHANGE * *TRANSPORTATION AND HANDLING * REGULATION, IAEA

128394

CHRIST H * SCHNEIDER K * SCHRODER H

DECONTAMINATION AND TRANSPORTATION PROBLEMS RELATED TO THE DECOMMISSIONING OF NUCLEAR POWER PLANTS AND THE ELIMINATION OF FAILURE CONSEQUENCES. APPENDIX II - SHIPMENT OF HEAVY COMPONENTS AND IRRADIATED SODIUM, JAN. 1, 1975-JUNE 30, 1976

TRANSNUKLEAR GMBH, F.R.G. GERMANY

GERNS-130 * NUM-312 *. 62 PPS, TABS, FIGS, MAY 1977

CHAPTER I DEALS WITH THE SHIPMENT OF ACTIVATED AND/OR CONTAMINATED HEAVY COMPONENTS FROM LIGHT WATER REACTORS FROM DECOMMISSIONING OPERATIONS; THE SECOND ONE THE TRANSFER AND OFF-SITE REMOVAL OF IRRADIATED SODIUM FROM SODIUM-COOLED FAST REACTORS.

120304 OCENTINTECH
 AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104
 TRANSPORTATION AND MODELING • DECOMMISSIONING • DECONTAMINATION • REGULATION, IEA • FOREIGN EXCHANGE

120470
 SERIE D • FROELICH G • ULLER H
 LITERATURE STUDIES ON VAPOR EXPLOSIONS (IN GERMAN)
 UNIVERSITAT STUTTGART, GERMANY
 GERMAN-136 • DWT-PS 76 •. APPROX. 210 PPS, 33 TABS, 49 FIGS, 274 REFS, FEB. 1976

THIS REPORT CONTAINS A LITERATURE REVIEW OF REPORTS ON ACCIDENTS RELATING TO THE VAPOR EXPLOSIONS CAUSED BY THERMODYNAMIC INTERACTIONS. EXPERIMENTS AND THEORETICAL INVESTIGATIONS ARE LISTED ALSO. REVIEWS ANALYTICAL MODELS AND COMPUTER PROGRAMS DEVELOPED TO DESCRIBE THE PHENOMENA OF VAPOR EXPLOSIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104
 BIBLIOGRAPHY • INCIDENT COMPILATION • GAS • VAPOR PRESSURE • EXPLOSION • ANALYTICAL MODEL • COMPUTER PROGRAM • GERMAN • FOREIGN EXCHANGE

120493
 SERIE D • FROELICH G • GOLDAMMER H
 THEORETICAL STUDIES OF VAPOR EXPLOSIONS (IN GERMAN)
 UNIVERSITAT STUTTGART, F.R. GERMANY
 GERMAN-137 • DWT-PS 76 •. 100 PPS, 1 TAB, 11 FIGS, 14 REFS, APRIL 1976

FOR FUEL-COOLANT-INTERACTIONS WHICH MAY LEAD TO VAPOR EXPLOSIONS, CALCULATIONS WITH TIME-INDEPENDENT METHODS HAVE BEEN PERFORMED IN ORDER TO MAKE ESTIMATIONS ON THE UPPER LIMITS FOR QUASI-STATIC PRESSURES AND MECHANICAL ENERGIES. THESE TIME INDEPENDENT METHODS ARE NOT SUFFICIENT TO PREDICT THE VAPOR EXPLOSIONS IN ANY DETAIL. THEREFORE, ADDITIONAL STUDIES BASED ON SIMPLE TIME-DEPENDENT MODELS HAVE BEEN PERFORMED. THE MOST ADVANCED MODEL DESCRIBES THE INCREASE OF THE SURFACE OF THE MOLTEN FUEL CAUSED BY THE ENERGY RELEASED DURING THE COLLAPSE OF VAPOR BUBBLES. THE RESULTS OF THIS MODEL ARE SUFFICIENT FOR A QUALITATIVELY CORRECT DESCRIPTION OF THE EXPERIMENTALLY OBSERVED CORRELATIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104
 FUEL COOLANT INTERACTION • CORE MELTDOWN • ANALYTICAL MODEL • GERMAN • FOREIGN EXCHANGE

120291
 SERIE A • SCHWALBE B • ULLER H
 AN ESTIMATE OF ENERGY RELEASE AND PRESSURE OUTFLOW DURING VAPOR EXPLOSIONS IN REACTOR GEOMETRY (IN GERMAN)
 UNIVERSITAT STUTTGART, GERMANY
 GERMAN-135 • DWT-PS 206 •. 130 PPS, 0 TABS, 43 FIGS, 42 REFS, DEC. 1976

THE PRESSURE OUTFLOW DURING VAPOR EXPLOSIONS IN LIGHT WATER REACTORS HAS BEEN ESTIMATED BY MEANS OF A SENSITIVITY STUDY APPLYING SIMPLE ENGINEERING METHODS. THE INITIAL DATA REQUIRED FOR THIS PURPOSE AND THE MATERIAL PROPERTIES OF THE CORE MELT HAVE BEEN OBTAINED FROM THE COURSE OF A POSTULATED, HYPOTHETICAL CORE MELTDOWN ACCIDENT. FOR THE ESTIMATION THE BEHAVIOR OF THE PRESSURE WITH RESPECT TO TIME HAS BEEN DIVIDED INTO TWO PHASES, A VERY SHORT, BUT HIGHLY TRANSIENT ONE FOLLOWED BY A LONGER, QUASI-STATIC PHASE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104
 ACCIDENT ANALYSIS • EXPLOSION • ACCIDENT MODEL • VAPOR PRESSURE • REACTOR, SAFETY RESEARCH • THEORETICAL INVESTIGATION • FOREIGN EXCHANGE

121120
 SERIE D • FROELICH G • GOLDAMMER H • HOFMANN H
 THEORETICAL AND EXPERIMENTAL INVESTIGATIONS OF VAPOR EXPLOSIONS (IN GERMAN)
 UNIVERSITAT STUTTGART, F.R. GERMANY
 FUR/CE-15/116/770 • GERMAN-150 •. 130 PPS, FIGS, REFS, JAN. 1977

A MORE ACCURATE DESCRIPTION OF THE OCCURRENCE AND COURSE OF VAPOR EXPLOSIONS IS ONLY POSSIBLE BY MEANS OF AN IMPROVED KNOWLEDGE OF THE DETAILS OF THE MECHANISM OF FRAGMENTATION. FOR THIS REASON, THE DEVELOPMENT OF SEVERAL FRAGMENTATION MODELS HAS BEEN STARTED. IN ORDER TO CALCULATE THE THERMODYNAMIC BEHAVIOR DURING A FUEL-COOLANT-INTERACTION IN A CHANNEL, AN ADVANCED MODEL APPLICABLE TO CHANNEL GEOMETRY IS IN JOURCES; FOR FUEL-COOLANT-INTERACTIONS IN A TANK GEOMETRY SUCH A MODEL HAS YET TO BE DEVELOPED. ALSO DISCUSSED THE VERIFICATION OF THE THEORETICAL WORK FOR WHICH EXPERIMENTS WERE PERFORMED IN A SHOCKTUBE DEVICE AND A TANK FACILITY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104
 GERMAN • D AND D PROGRAM • FUEL COOLANT INTERACTION • ANALYTICAL MODEL • THERMAL HYDRAULIC ANALYSIS • POSTER VAPOR • GAS • EXPLOSION • SHOCK WAVE • METAL WATER REACTION • FOREIGN EXCHANGE

3. JAPANESE LIGHT-WATER REACTOR SAFETY REPORTS RECEIVED BY NRC

THE FOLLOWING IS A LISTING OF REPORTS RECEIVED FROM JAPAN DURING THE SECOND HALF OF 1977 UNDER THE TECHNICAL EXCHANGE AGREEMENT.

131132

MURAO Y • IGUCHI Y • SUGIYAMA Y • SUGI Y
RE-GRY ON SERIES 3 REFLUOD EXPERIMENT (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-N-0001 • JNPR58-00 • 71 PPS, TABS, FIGS, REFS, FEB. 1, 1977

THE PURPOSE OF THE EXPERIMENT WAS TO CONFIRM TEMPERATURE RESPONSE AND DURABILITY OF THE IMPROVED THERMOCOUPLE INSTALLATION AND TC EXHAUST SYSTEM EFFECT WITH FLOW HOUSING TEMPERATURE AND PRIMARY LOOP FLOW RESISTANCE. THE RESULTS ARE: THE IMPROVED THERMOCOUPLES INSTALLATION STILL HAS SOME PROBLEMS; THE FLOW HOUSING TEMPERATURE HAS A LARGE INFLUENCE ON THE REFLUOD PHENOMENON, ESPECIALLY OSCILLATION; AND THE PRIMARY LOOP RESISTANCE DETERMINES THE FLOODING RATE, AND SO INFLUENCES THE REFLUOD PHENOMENON.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104

JAPAN • A AND B PROGRAM • TESTING • CORE REFLUODING • SAFETY INJECTION • ENGINEERED SAFETY FEATURE • THERMOCOUPLE • HYDRAULIC EFFECT • REACTOR, PWR • FOREIGN EXCHANGE

131133

IGUCHI Y • MURAO Y • SUGIYAMA Y
DATA REPORT ON SERIES 3 REFLUOD EXPERIMENT (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-N-0002 • JNPR58-00 • 200 PPS, TABS, FIGS, FEB. 1, 1977

THE DATA OBTAINED IN THE SERIES 3 REFLUOD EXPERIMENT FROM DECEMBER 1975 TO JANUARY 1976 ARE PRESENTED. IN THIS SERIES OF EXPERIMENTS, THE FLOW RESISTANCE IN A PRIMARY LOOP AND THE TEMPERATURE OF THE FLOW HOUSING WERE INVESTIGATED. PERFORMANCE AND DURABILITY OF THE IMPROVED THERMOCOUPLE ATTACHMENT WERE ALSO INVESTIGATED AND ARE REPORTED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104

JAPAN • THERMOCOUPLE • FASTENER • PERFORMANCE • CORE REFLUODING • FLOW THEORY AND EXPERIMENTS • SAFETY INJECTION • THERMAL HYDRAULIC ANALYSIS • FOREIGN EXCHANGE

131134

MURAO Y • IGUCHI Y • SUGIYAMA Y • SUGI Y
REPORT ON SERIES 4 REFLUOD EXPERIMENT (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-N-0003 • JNPR58-00 • 44 PPS, 8 TABS, 31 FIGS, 10 REFS, FEB. 1, 1977

DISCUSSES SYSTEM-PRESSURE EFFECTS AND SYSTEM EFFECTS WITH HEATER RODS HAVING ABOUT THE SAME HEAT CAPACITY AND FLOW RESISTANCE AS FUEL RODS. THE RESULTS ARE: THE CORE FLOOD VELOCITY INCREASES AND THE QUENCH TIME DECREASES WITH SYSTEM PRESSURE; THE RELATION BETWEEN CORE DIFFERENTIAL PRESSURE AND CORE POWER DENSITY IS OBTAINED; AND THE STEADY-STATE THERMAL CONDITION OCCURS IN THE TEST SECTION WITH CONSTANT POWER DENSITY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104

JAPAN • A AND B PROGRAM • SIMULATION • FUEL ROD • HEATERS • CORE REFLUODING • SAFETY INJECTION • THERMAL HYDRAULIC ANALYSIS • FOREIGN EXCHANGE

131135

MURAO Y • SUGIYAMA Y
A STUDY ON QUENCH PHENOMENA DURING REFLUOD PHASE. 1. QUENCH MODELS FOR BOTTOM FLOODING
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-N-0004 • JNPR58-01 • 39 PPS, 1 TAB, 22 FIGS, 15 REFS, FEB. 1, 1977

USING BLAIR'S CORRELATION FOR QUENCH VELOCITY, THE APPROXIMATE CORRELATION FOR MAXIMUM LIQUID SUPERHEAT, THE ASSUMPTION THAT THE HEAT TRANSFER UPSTREAM OF THE QUENCH FRONT IS A FUNCTION OF THE LOCAL LIQUID SUBCOOLING AND THE DATA OF PUR-FLECHY EXPERIMENTS, THE CORRELATION FOR QUENCH VELOCITY OF THE LIQUID COLUMN TYPE AND OF THE BOUTON TYPE ARE OBTAINED. THE QUENCH TEMPERATURE FOR THE DROPLET-QUENCHING TYPE IS ALSO DERIVED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22104

JAPAN • CORE REFLUODING • THERMAL HYDRAULIC ANALYSIS • HEAT TRANSFER ANALYSIS • SUPERHEAT • ANALYTICAL MODEL • SAFETY INJECTION • COMPUTER PROGRAM • ACCIDENT, LOSS OF COOLANT • QUENCHING • HEAT FLUX, DRYOUT • FOREIGN EXCHANGE

131136

SUGIYAMA Y
STUDY ON THERMOCOUPLE ATTACHMENT IN REFLUOD EXPERIMENTS (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-N-0005 • JNPR58-02 • 36 PPS, FIGS, REFS, FEB. 1, 1977

130130 CONTINUED

DESCRIBES THE METHOD OF INFERREDIBLE ATTACHMENT TO HEATER RODS FOR SURFACE TEMPERATURE MEASUREMENT IN REPLICATED EXPERIMENTS. THE METHOD USES TAPS FOR IN JAPAN'S REPLICATED EXPERIMENTS DO NOT ESTIMATE THE QUENCH TIME EXACTLY. VARIOUS ATTACHMENT METHODS HAVE BEEN TESTED AND PROVE PROVED TO BE EFFECTIVE AND DESCRIBED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

JAPAN - OTHER AVAILABLE - CORE REPLICATING - SAFETY INJECTION - MEASUREMENT - INSTRUMENT, TEMPERATURE - HEATERS - FUEL ROD - FASTENED - FOREIGN EXCHANGE

130137

SHIMIZU M + KAWANO Y + MATSUMOTO I + UENO H
NISA-11 TEST DATA EFFECT NO. 7 - QUENCH BEAR CAPABILITY AND PUMP EFFECT - IMPUS 300, 320, 321, 322, 323 (EN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO
JAPRI-R-7105 + JAPRI-S-010, 100 PAGES, 10 FIGS, MAY 9, 1977

A SIMULATED FAILURE OF LOCA INCLUDING ECCS OPERATION WAS PERFORMED WITH THE USE OF THE NISA-11 TEST FACILITY WHICH IS DESIGNED TO SIMULATE THE PRIMARY REACTOR IN TERMS OF THERMAL-HYDRAULIC RESPONSE. PART OF THE EXPERIMENTAL RESULTS ARE GIVEN WITH AN INTERPRETATION OF THE DATA. THE OBJECTIVE OF EACH RUN WHICH IS DESCRIBED WAS TO FRAME THE QUENCH BEAR CAPABILITY OF A REACTOR PREDICTED BY PRELIMINARY ANALYSIS WITH RELAP-3. TWO KINDS OF HEAR CAPABILITY WERE TAKEN WITH OR WITHOUT PUMP RECOVERY AFTER A QUENCH. ECCS WAS PROVIDED IN HOT LEGS AND COLD LEGS FOR NISA-11, IN HOT LEGS ONLY FOR L-1, AND IN COLD LEGS ONLY FOR THE ACCUMULATORS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

JAPAN - R AND D PROGRAM - TESTING - EXPERIMENTAL CIVIL IN. SYSTEM - PERFORMANCE - ACCIDENT, LOSS OF COOLANT - SIMULATION - REACTOR, PWR - FOREIGN EXCHANGE

130150

SHIMIZU M + KAWANO Y + UENO H + SUGIYAMA Y
DATA REPORT ON TESTS OF REPLICATED EXPERIMENT FOR JAPANESE & FOREIGN
JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO
JAPRI-R-7106 + JAPRI-S-010, 100 PAGES, 5 FIGS, 10 PAGES, JUNE 20, 1977

MEASURED RESULTS AND CALCULATED HEAT TRANSFER CAPABILITY OF THE SERIES A REPLICATED EXPERIMENT ARE PRESENTED. EXPERIMENTS WERE CARRIED OUT IN JUNE AND JULY 1976. THE OBJECTIVE WAS TO INVESTIGATE THE REPLICATED BEHAVIOR UNDER DIFFERENT SYSTEM MODES USING HEATER RODS OF SIMULATED HEAT CAPACITY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

JAPAN - R AND D PROGRAM - CORE REPLICATING - SAFETY INJECTION - SIMULATION - FUEL ROD - HEATERS - HEAT TRANSFER ANALYSIS - FOREIGN EXCHANGE

130151

MAKINAKA K + KIMURA S + KAWANO Y
SIMULATED DECOMPRESSION ANALYSIS OF LOFT L1-3A TEST DATA WITH DIGITAL COMPUTER CODE DOPET-MULTI
JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO
JAPRI-R-7113 + JAPRI-S-010, 100 PAGES, 13 FIGS, 5 PAGES, JULY 1977

AS A PART OF THE ANALYTICAL EFFORT OF THE NISA PROGRAM, THE DIGITAL COMPUTER CODE DOPET-MULTI WHICH CALCULATES SIMULATED DECOMPRESSION OF PRIMARY COOLANT IN THE EVENT OF A POSTULATED PWR LOCA, WAS DEVELOPED. THE CAPACITY OF THE CODE WAS ALREADY BEEN DEMONSTRATED BY COMPARING THE ANALYTICAL RESULTS WITH THE NISA-1, -II TEST DATA AND THE LOFT VERTICAL TEST DATA. IN THIS REPORT THE SIMULATED DECOMPRESSION OF THE LOFT L1-3A TEST IS ANALYZED BY DOPET-MULTI, AND THE RESULTS ARE COMPARED WITH L1-3A TEST DATA AND ALSO WITH THE RESULT OF PRETEST ANALYSIS MADE BY UNION.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

COMPUTER PROGRAM - PERFORMANCE - ACCIDENT, LOSS OF COOLANT - SIMULATING - REACTOR, PWR - JAPAN - FOREIGN EXCHANGE

130150

MAE T + UHIMOTO M
QUASI STEADY STATE MULTI-DIMENSIONAL SPACE-DEPENDENT KINETIC CODE, EURPRA-SPACE (EN JAPANESE & FOREIGN)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO
JAPRI-R-7108 + JAPRI-S-010, 77 PAGES, 3 FIGS, 10 PAGES, JUNE 30, 1977

THE CODE WAS DEVELOPED TO ANALYZE A WATER COOLED POWER REACTOR DURING A REACTIVITY ACCIDENT. THE CODE INCLUDES THE FOLLOWING FEATURES: (1) MULTI-DIMENSIONAL POWER AND FLUX DISTRIBUTIONS ARE CALCULATED FOR SEVERAL FUEL ELEMENTS; (2) THERMAL-HYDRAULIC CALCULATION FOR UP TO A MAXIMUM OF FIVE REGIONS IS PERFORMED; AND (3) FEEDBACK REACTIVITY FROM DOPPLER EFFECT, MODERATION EFFECT, VOID FORMATION AND CLADDING THERMAL EXPANSION EFFECTS IS TAKEN INTO ACCOUNT. THIS REPORT PRESENTS DETAILED DESCRIPTIONS OF THE CODE, INCLUDING BASIC EQUATIONS, AND LISTS OF INPUT AND

131554 -CONTINUED-
OUTPUT AND POST CALCULATION RESULTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

COMPUTED PROGRAM • THERMAL HYDRAULIC ANALYSIS • ACCIDENT, REACTIVITY • REACTOR KINETICS • REACTOR, LWR • JAPAN • FOREIGN EXCHANGE

131555
TAKEDA T
USER'S GUIDE FOR CODE-NO 4: A COMPUTER PROGRAM TO CALCULATE NUCLEON YIELDS IN COMPLEX DECAY AND ACTIVATION CHAIN FOR JAPANESE & ENGLISH
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JARI-8-7733 • JARI-54-97 • 96 PPS, 3 TABS, 20 FIGS, 5 PPS, JULY 29, 1977

THIS REPORT IS PREPARED AS A USER'S INPUT MANUAL AND PROVIDES A GENERAL DESCRIPTION OF THE CODE AND INSTRUCTIONS FOR ITS USE. THE CODE IS DESCRIBED IN DETAIL AND BY THE USE OF VARIOUS EXAMPLES. THE USER APPROPRIATES A VERSION OF THE CODE-NO. 4 CODE. THE CODE DEVELOPED IS CAPABLE OF CALCULATING RADIOACTIVE NUCLEON YIELDS IN AN ANY GIVEN COMPLEX DECAY AND ACTIVATION CHAIN IN ACCORDANCE WITH A GIVEN IRRADIATION HISTORY. LISTS OF FORTRAN STATEMENTS, LISTS OF INPUT PARAMETERS AND DEFINITIONS AND EXAMPLES OF INPUT DATA AND OUTPUT ARE GIVEN IN THIS REPORT. IN ADDITION TO VARIOUS EXAMPLES, DETAILS OF INPUT DATA PREPARED FOR COMPLEX CHAIN INVOLVED IN FISSION AND ACTIVATION ARE ALSO GIVEN.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

JAPAN • GUIDE • COMPUTER PROGRAM • FISSION PRODUCT YIELD • FISSION PRODUCT TRANSPORT • ACTIVATION • ACTIVATION PRODUCT • FOREIGN EXCHANGE

131556
HONDA T • ZINCAIDU CHANDING UNDER A POSTULATED LOSS-OF-COOLANT ACCIDENT CONDITION (IN JAPANESE & ENGLISH)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JARI-8-7707 • JARI-54-94 • 100 PPS, 14-TS, FIGS, PPS, AUG. 19, 1977

PRESENTS A REVIEW OF STUDIES ON THE BEHAVIOR OF ZINCAIDU CHANDING UNDER A POSTULATED LOSS-OF-COOLANT ACCIDENT CONDITION. EXPERIMENTAL DATA OF ZINCAIDU-STEAM REACTION RATE, EQUIVALENT OF A STANDARDIZED ZINCAIDU AND CHANDING INFORMATION ARE SUMMARIZED. SENSITIVITY STUDIES OF VARIOUS PARAMETERS ON THE ANALYSIS BY A COMPUTER CODE ARE PRESENTED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

JAPAN • CHANDING • REACTIVITY • ZINCAIDU • ACCIDENT, LOSS OF COOLANT • EQUIVALENT • FUEL SWELLING • FOREIGN EXCHANGE

A COLLECTION OF REVIEWS IS USED TO IDENTIFY THE MAIN SAFETY RELATED POINTS COVERED IN EACH ARTICLE. THE FOLLOWING INDEX IS AN ALPHABETICAL LISTING OF THE REVIEWS CARRYING REFERENCES TO EACH ARTICLE WHICH WAS REFERRED TO IT.

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NUCLEAR YIELDS IN COMPLEX DECAY AND ACTIVATION CHAIN (IN	127067	0
GERMAN)	127068	0
NUCLEAR YIELDS IN COMPLEX DECAY AND ACTIVATION CHAIN (IN	127069	0
GERMAN)	127070	0
NUCLEAR YIELDS IN COMPLEX DECAY AND ACTIVATION CHAIN (IN	127071	0
GERMAN)	127072	0
NUCLEAR YIELDS IN COMPLEX DECAY AND ACTIVATION CHAIN (IN	127073	0
GERMAN)	127074	0
NUCLEAR YIELDS IN COMPLEX DECAY AND ACTIVATION CHAIN (IN	127075	0
GERMAN)	127076	0
NUCLEAR YIELDS IN COMPLEX DECAY AND ACTIVATION CHAIN (IN	127077	0
GERMAN)	127078	0
NUCLEAR YIELDS IN COMPLEX DECAY AND ACTIVATION CHAIN (IN	127079	0
GERMAN)	127080	0
NUCLEAR YIELDS IN COMPLEX DECAY AND ACTIVATION CHAIN (IN	127081	0
GERMAN)	127082	0
NUCLEAR YIELDS IN COMPLEX DECAY AND ACTIVATION CHAIN (IN	127083	0
GERMAN)	127084	0
NUCLEAR YIELDS IN COMPLEX DECAY AND ACTIVATION CHAIN (IN	127085	0
GERMAN)	127086	0
NUCLEAR YIELDS IN COMPLEX DECAY AND ACTIVATION CHAIN (IN	127087	0
GERMAN)	127088	0
NUCLEAR YIELDS IN COMPLEX DECAY AND ACTIVATION CHAIN (IN	127089	0
GERMAN)	127090	0
NUCLEAR YIELDS IN COMPLEX DECAY AND ACTIVATION CHAIN (IN	127091	0
GERMAN)	127092	0
NUCLEAR YIELDS IN COMPLEX DECAY AND ACTIVATION CHAIN (IN	127093	0
GERMAN)	127094	0
NUCLEAR YIELDS IN COMPLEX DECAY AND ACTIVATION CHAIN (IN	127095	0
GERMAN)	127096	0
NUCLEAR YIELDS IN COMPLEX DECAY AND ACTIVATION CHAIN (IN	127097	0
GERMAN)	127098	0
NUCLEAR YIELDS IN COMPLEX DECAY AND ACTIVATION CHAIN (IN	127099	0
GERMAN)	127100	0

[illegible]

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