

ornl

**OAK
RIDGE
NATIONAL
LABORATORY**

**UNION
CARBIDE**

**OPERATED BY
UNION CARBIDE CORPORATION
FOR THE UNITED STATES
DEPARTMENT OF ENERGY**

NUREG/CR-1101
ORNL/NUREG/NSIC-170
(Vol. VII of TID-3362)

THIS DOCUMENT CONTAINS
POOR QUALITY PAGES

**Reports Distributed Under
the NRC Reactor
Safety Research Foreign
Technical Exchange Program
Vol. VII (January-June 1979)**

Debbie S. Queener

Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Under Interagency Agreements DOE 40-551-75 and 40-552-75

NUCLEAR SAFETY INFORMATION CENTER

NSIC

7912140186

AVAILABILITY OF NSIC DOCUMENTS

Recent NSIC reports that may be ordered from the National Technical Information Service, U.S. Department of Commerce, 5285 Port Royal Road, Springfield, Virginia 22161 are listed below.

ORNL/ NSIC	Title	Price*
53	Radiography Incidents and Overexposures, R. L. Scott and R. B. Gallaher, Dec. 1972	\$10.50
55	Design Data and Safety Features of Commercial Nuclear Power Plants, Vol. I, Docket No. 50-3 through 50-295, F. A. Heddleson, Dec. 1973	\$15.00
Vol. I		
55	Design Data and Safety Features of Commercial Nuclear Power Plants, Vol. II, Docket No. 50-296 through 50-395, F. A. Heddleson, Jan. 1972	\$15.00
Vol. II		
55	Design Data and Safety Features of Commercial Nuclear Power Plants, Vol. III, Docket No. 50-397 through 50-449, F. A. Heddleson, Apr. 1974	\$15.00
Vol. III		
55	Design Data and Safety Features of Commercial Nuclear Power Plants, Vol. IV, Docket No. 50-452 through 50-503, F. A. Heddleson, June 1975	\$15.00
Vol. IV		
	See ORNL/NSIC-96 for Vol. V and ORNL/NUREG/NSIC-136 for Vol. VI.	
74	Calculation of Doses Due to Accidentally Released Plutonium from an LMFBR, B. R. Fish, G. W. Keilholtz, W. S. Snyder, and S. D. Swisher, Nov. 1972	\$15.00
82	Chemical and Physical Properties of Methyl Iodide and its Occurrence Under Reactor Accident Conditions - A Summary and Annotated Bibliography, L. F. Parsly, Dec. 1971	\$12.50
91	Safety-Related Occurrences in Nuclear Facilities as Reported in 1970, R. L. Scott, Dec. 1971	\$10.50
96	Design Data and Safety Features of Commercial Nuclear Power Plants (Fifth Volume of ORNL/NSIC-55), F. A. Heddleson, June 1976	\$15.00
97	Indexed Bibliography of Thermal Effects Literature-2, J. G. Morgan and C. C. Coutant, May 1972	\$10.50
100	Nuclear Power and Radiation in Perspective, Selections from <i>Nuclear Safety</i> , J. K. Buchanan, Mar. 1974	\$15.00
101	Indexed Bibliography on Environmental Monitoring for Radioactivity, B. I. Houser, May 1972	\$10.50
102	Compilation of National and International Nuclear Standards (Excluding U.S. Activities) 8th Ed., 1972, J. P. Blakely, June 1972	\$10.50
103	Abnormal Reactor Operating Experiences, 1969-1971, R. L. Scott and R. B. Gallaher, May 1972	\$ 9.00
105	Indexed Bibliography on Nuclear Facility Siting, H. B. Piper, June 1972	\$10.50
106	Safety-Related Occurrences in Nuclear Facilities as Reported in 1971, R. L. Scott and R. B. Gallaher, Sept. 1972	\$12.50
107	Index to <i>Nuclear Safety</i> , A Technical Progress Review by Chronology, Permuted Title, and Author, Vol. 1, No. 1 through Vol. 13, No. 6, J. Paul Blakely and Ann Klein, May 1973	\$ 9.00
109	Safety-Related Occurrences in Nuclear Facilities as Reported in 1972, R. L. Scott and R. B. Gallaher, Dec. 1973	\$15.00
110	Indexed Bibliography of Thermal Effects Literature-3, J. G. Morgan, July 1973	\$12.50
111	Reactor Protection Systems: Philosophies and Instrumentation, Reviews from <i>Nuclear Safety</i> , E. W. Hagen, July 1973	\$15.00
112	Compilation of Nuclear Standards, 9th Edition, 1972, Part I: United States Activities, J. P. Blakely, Oct. 1973	\$12.50
113	A Selected Bibliography on Emergency Core Cooling Systems (ECCS) for Light-Water-Cooled Power Reactors (LWRs), Wm. B. Cottrell, Jan. 1974	\$12.50
114	Annotated Bibliography of Safety-Related Occurrences in Nuclear Power Plants as Reported in 1973, R. L. Scott and R. B. Gallaher, Nov. 1974	\$15.00
117	Protection of Nuclear Power Plants Against External Disasters, Wm. B. Cottrell, May 1975	\$15.00
119	A Selected Bibliography on Pressure Vessels for Light-Water-Cooled Power Reactors (LWRs), Fred A. Heddleson, Jan. 1975	\$15.00
120	Annotated Bibliography of Hydrogen Considerations in Light-Water Power Reactors, G. W. Keilholtz, Feb. 1976	\$10.00
121	Reactor Operating Experiences, 1972-1974, U.S. Nuclear Regulatory Commission, Dec. 1975	\$ 8.00

(Continued on Inside Back Cover)

NUREG/CR-1101
ORNL/NUREG/NSIC-170
(Vol. VII of TID-3362)
Dist. Category AE

Contract No. W-7405-eng-26

Nuclear Safety Information Center

REPORTS DISTRIBUTED UNDER THE
NRC REACTOR SAFETY RESEARCH
FOREIGN TECHNICAL EXCHANGE PROGRAM
VOL. VII
(JANUARY-JUNE 1979)

Debbie S. Queener
Engineering Technology Division

Manuscript Completed - October 23, 1979
Date Published - November 1979

Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Under Interagency Agreements DOE 40-551-75 and 40-552-75

NRC FIN No. B0126

Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
DEPARTMENT OF ENERGY

Printed in the United States of America. Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States nor any agency thereof, nor any of their employees, makes any warranty, expressed or implied, or assumes any legal liability or responsibility for any third party's use or the results of such use of any information, apparatus, product or process disclosed in this report, or represents that its use by such third party would not infringe privately owned rights.

CONTENTS

	<u>Page</u>
FOREWORD.....	v
PREVIOUS REPORTS IN THIS SERIES.....	vii
ABSTRACT.....	ix
INTRODUCTION.....	ix
AVAILABILITY OF REPORTS.....	x
ORGANIZATION OF BIBLIOGRAPHY.....	xii
PRICES FOR DOCUMENTS ABSTRACTED IN THIS REPORT.....	xiii
PARTS AND METHOD OF INDEXING ABSTRACTS.....	xiv
BIBLIOGRAPHY.....	xv
1. FRENCH LIGHT-WATER REACTOR SAFETY RESEARCH REPORTS RECEIVED BY NRC.....	1
2. GERMAN (FRG) LIGHT-WATER REACTOR SAFETY RESEARCH REPORTS RECEIVED BY NRC.....	10
3. GERMAN (FRG) FAST REACTOR SAFETY RESEARCH REPORTS RECEIVED BY NRC.....	26
4. JAPANESE LIGHT-WATER REACTOR SAFETY RESEARCH REPORTS RECEIVED BY NRC.....	27
5. JAPANESE FAST REACTOR SAFETY RESEARCH REPORTS RECEIVED BY NRC.....	30
6. UNITED KINGDOM LIGHT-WATER REACTOR SAFETY RESEARCH REPORTS RECEIVED BY NRC.....	31
7. UNITED KINGDOM FAST REACTOR SAFETY RESEARCH REPORTS RECEIVED BY NRC.....	35
KEYWORD INDEX.....	37
AUTHOR INDEX.....	43
PERMUTED-TITLE INDEX.....	47

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 Research and Technology 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 keywords to index the information which it catalogs. The title, author, installation, abstract, and keywords for each document reviewed are recorded at the central computing facility in Oak Ridge. The references are cataloged according to the following categories:

1. General Safety Criteria
2. Siting of Nuclear Facilities
3. Transportation and Handling of Radioactive Materials
4. Aerospace Safety (inactive ~1970)
5. Heat Transfer and Thermal Hydraulics
6. Reactor Transients, Kinetics, and Stability
7. Fission Product Release, Transport, and Removal
8. Sources of Energy Release under Accident Conditions
9. Nuclear Instrumentation, Control, and Safety Systems
10. Electrical Power Systems
11. Containment of Nuclear Facilities
12. Plant Safety Features - Reactor
13. Plant Safety Features - Nonreactor
14. Radionuclide Release, Disposal, Treatment, and Management
(inactive September 1973)
15. Environmental Surveys, Monitoring, and Radiation Dose Measurements
(inactive September 1973)
16. Meteorological Considerations

17. Operational Safety and Experience
18. Design, Construction and Licensing
19. Internal Exposure Effects on Humans Due to Radioactivity in the Environment (inactive September 1973)
20. Effects of Thermal Modifications on Ecological Systems (inactive September 1973)
21. Radiation Effects on Ecological Systems (inactive September 1973)
22. Safeguards of Nuclear Materials

Computer programs have been developed that enable NSIC to (1) operate a program of selective dissemination of information (SDI) to individuals according to their particular profile of interest, (2) make retrospective searches of the stored references, and (3) produce topical indexed bibliographies. In addition, the Center Staff is available for consultation, and the document literature at NSIC offices is available for examination. NSIC reports (i.e., those with the ORNL/NSIC and ORNL/NUREG/NSIC numbers) may be purchased from the National Technical Information Service (see inside front cover). All of the above services are free to NRC and DOE personnel as well as their direct contractors. They are available to all others at a nominal cost as determined by the DOE Cost Recovery Policy. Persons interested in any of the services offered by NSIC should address inquiries to:

J. R. Buchanan, Assistant Director
Nuclear Safety Information Center
P.O. Box Y
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830

Telephone 615-574-0391
FTS number is 624-0391

PREVIOUS REPORTS IN THIS SERIES

<u>Report No.</u>	<u>Period Covered</u>	<u>Publication Date</u>
TID-3362	November 1974-December 1975	May 1976
ORNL/NUREG/NSIC-134	January-December 1976	April 1977
ORNL/NUREG/NSIC-142	January-June 1977	October 1977
ORNL/NUREG/NSIC-146	July-December 1977	June 1978
ORNL/NUREG/NSIC-156	January-June 1978	November 1978
ORNL/NUREG/NSIC-159	July-December 1978	May 1979

REPORTS DISTRIBUTED UNDER THE
NRC REACTOR SAFETY RESEARCH
FOREIGN TECHNICAL EXCHANGE PROGRAM
VOL. VII
(JANUARY-JUNE 1979)

Debbie S. Queener
Engineering Technology Division

ABSTRACT

Lists of documents exchanged during the first half of 1979 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. These agreements cover safety research on high-temperature gas-cooled reactors (HTGR), light-water reactors, and fast reactors. During this period, the NRC received 43 reports from France, 81 from the Federal Republic of Germany, 17 from Japan, and 84 from the United Kingdom. In return, the NRC sent 163 United States light-water reactor safety research reports to each of these four countries, 50 fast reactor safety research reports to all except France, and 8 HTGR research reports to Japan.

INTRODUCTION

This report lists the documents exchanged during the first half of 1979 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 seventh report in this series. 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 United States and Japan. The fast reactor safety research exchange agreements are between the United States and the Federal Republic of Germany, Japan, and the United Kingdom, respectively.

The total number of reports received by the NRC from January through June 1979 from France, the Federal Republic of Germany, Japan, and the

United Kingdom, as well as the reports sent by the NRC to each of these four countries during this period, are listed in Table 1 for the LWR, HTGR, and fast reactor exchanges. Also tabulated are the documents exchanged from 1974 through the second half of 1978. The number of proprietary documents received are listed in parentheses.

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, the Federal Republic of Germany, Japan, and the United Kingdom are listed in the computerized bibliography first by country, then alphabetically by installation, and finally chronologically by report date; the LWR and fast reactor reports are listed separately. Any reports concerning general safety research and those of no specific reactor type that were received under the LWR exchange are listed in the LWR category. Additional bibliographic information, including the number assigned by the issuing installation, abstract, and keywords, is also presented.

Beginning with this semiannual report, the United States reports transmitted to foreign exchange agreement countries are no longer listed here. The distribution of these documents, formerly handled by the Nuclear Safety Information Center (NSIC), is now being effected by the NRC Division of Technical Information and Document Control.

AVAILABILITY OF REPORTS

All domestic reports are available from the National Technical Information Service, U.S. Department of Commerce, Springfield, Va. 22161, in either microfiche or hard copy. The foreign reports are no longer available from NTIS, although it is designated in the bibliography as the sales agent for those documents which were processed before October 1, 1979. Persons interested in obtaining a copy of a foreign report should write to Dr. G. L. Bennett, U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Washington, D.C. 20555.

Table 1
Reports received by NRC^a

	France			F.R. Germany			Japan			United Kingdom		
	LWR	HTGR	Fast Reactor	LWR	HTGR	Fast Reactor	LWR	HTGR	Fast Reactor	LWR	HTGR	Fast Reactor
January-June 1979	39(4)	0	0	76(4)	0	1	12(2)	0	3	15(68)	0	1
July-December 1978	46	0	0	72	0	0	40(10)	0	3	87(67)	0	6
January-June 1978	21(3)	0	0	87	0	0	45(22)	0	2	62(55)	0	7
July-December 1977	1	0	0	40	0	0	11	0	0	0	0	0
January-June 1977	41(12)	0	0	29(1)	0	0	24(1)	0	0	0	0	0
1976	25(6)	0	0	75(9)	0	0	38(5)	0	0	0	0	0
1974-1975	10	0	0	6	0	0	24	0	0	0	0	0

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

XI

Reports sent by NRC

	France			F.R. Germany			Japan			United Kingdom		
	LWR	HTGR	Fast Reactor	LWR	HTGR	Fast Reactor	LWR	HTGR	Fast Reactor	LWR	HTGR	Fast Reactor
January-June 1979	163	0	0	163	0	50	163	8	50	163	0	50
July-December 1978	177	0	0	177	0	26	177	7	26	177	0	26
January-June 1978	125 ^a	0	0	125 ^a	0	22	125 ^a	0	22	125 ^a	0	22
July-December 1977	112	0	0	112	0	0	112	37	0	23	0	0
January-June 1977	107	0	0	107	0	0	107	0	0	107	0	0
1976	119	0	0	154	0	0	155	0	0	0	0	0
1974-1975	100	0	0	181	0	0	115	0	0	0	0	0

^aIncludes 4 reports which include LWR, HTGR, and fast reactor safety information.

ORGANIZATION OF BIBLIOGRAPHY

The bibliography which follows contains all the safety research reports received by NRC under the foreign exchange agreement program from January through June 1979. The German reports are listed alphabetically by organization name and then chronologically by report date under each organization. The reports received from France, Japan, and the United Kingdom are listed chronologically by report date, since the reports were issued by one organization in each country. The bibliography is sorted into the following categories:

1. French light-water reactor safety research reports received by NRC.
2. German (FRG) light-water reactor safety research reports received by NRC.
3. German (FRG) fast reactor safety research reports received by NRC.
4. Japanese light-water reactor safety research reports received by NRC.
5. Japanese fast reactor safety research reports received by NRC.
6. United Kingdom light-water reactor safety research reports received by NRC.
7. United Kingdom fast reactor safety research reports received by NRC.

PRICES FOR DOCUMENTS ABSTRACTED IN THIS REPORT

The prices of the documents abstracted in this report depend upon the number of pages in the individual documents. The page count found in the line preceding the abstract determines the price according to the following schedule:

<u>Page Range</u>	<u>Domestic^{a,b} Price</u>
001-025	\$ 4.00
026-050	4.50
051-075	5.25
076-100	6.00
101-125	6.50
126-150	7.25
151-175	8.00
176-200	9.00
201-225	9.25
226-250	9.50
251-275	10.75
276-300	11.00
301-325	11.75
326-350	12.00
351-375	12.50
376-400	13.00
401-425	13.25
426-450	14.00
451-475	14.50
476-500	15.00
501-525	15.25
526-550	15.50
551-575	16.25
576-600	16.50
601-up	^c

^aDouble the cost per copy for foreign price.

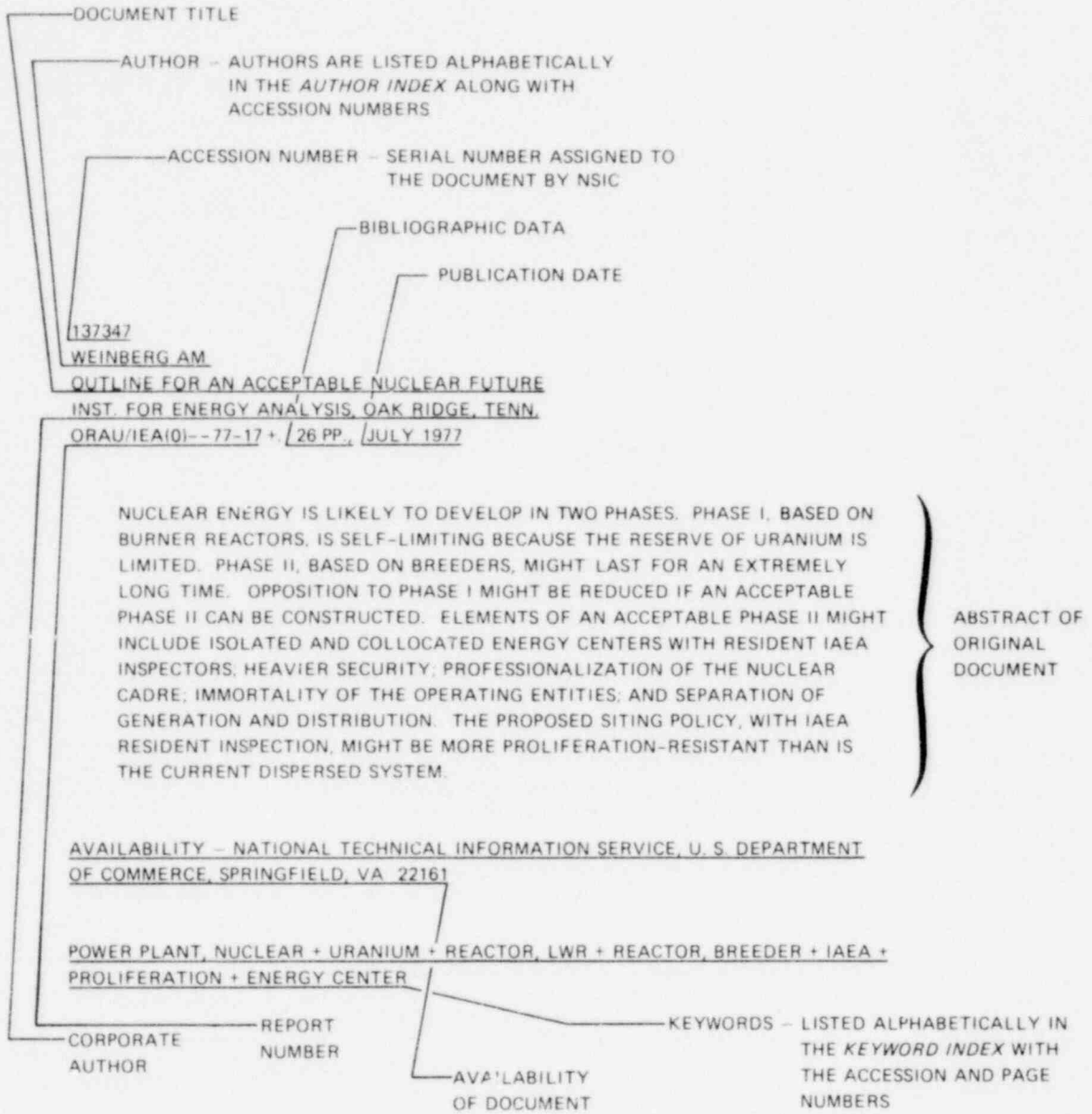
^bThese prices are subject to change by NTIS.

^cAdd \$2.50 for each additional 100-page increment over 600 pages.

Microfiche copies, regardless of document size, sell for \$3.00 per copy (domestic) and \$4.50 per copy (foreign), with requests from Mexico and Canada being counted as domestic. All documents may be ordered from

The National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22161

METHOD OF INDEXING DOCUMENTS



BIBLIOGRAPHY

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

THE FOLLOWING IS A LISTING OF REPORTS RECEIVED FROM FRANCE DURING THE FIRST HALF OF 1979 UNDER THE TECHNICAL EXCHANGE AGREEMENT.

143930
DASCALAKIS J
UNDIMENSIONAL MODELS FOR 2PHASE FLOWS (IN FRENCH)
CEA DEPARTEMENT DES REACTEURS A EAU, FRANCE
DRE/SRE/ZLET/77/112 + FRRSR-153 +, 35 PPS, FIGS, NOV. 8, 1977

THIS REPORT REVIEWS AND INVESTIGATES THE THEORETICAL BASES FOR THE SCALED DOWN REPRESENTATION OF TWO PHASE FLUID MECHANICS. SEVERAL ONE DIMENSIONAL ANALYTICAL AND NUMERICAL MODELS FOR TWO PHASE FLOWS ARE EXAMINED AND THEIR RESTRICTIONS AND LIMITATIONS ARE DISCUSSED.

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

*FLOW, TWO PHASE + ANALYTICAL MODEL + VOID FRACTION + *MASS TRANSFER + *NUMERICAL METHOD

145296
SIGNORET JP
SYSTEMS WHICH ARE UNCONNECTED AND WAITING FOR PERIODIC TESTING-NONNEGLECTIBLE TEST DURATION, TEST EFFICIENCY NOT 100%, AND 1 OUT OF 2 STANDBY SYSTEM (IN FRENCH)
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
DSN 206 + FRRSR-151 +, 98 PPS, JAN, 1978

USING THE BASIC MATHEMATICAL MODEL OF DSN REPORT 113, A SINGLE SYSTEM IS CHARACTERIZED BY THE FOLLOWING PARAMETERS: STAND-BY FAILURE RATE, REPAIR RATE, PROBABILITY NOT TO START ON DEMAND, AND TEST INTERVAL. IN EACH OF THE 3 PARTS ANALYTICAL FORMULAS ARE DEVELOPED TO ASSESS: POINTWISE (INSTANTANEOUS) AVAILABILITY, MEAN AVAILABILITY, THE LIMIT OF POINTWISE AVAILABILITY, AND STEADY STATE AVAILABILITY, WHEN THE NUMBER OF TEST INTERVALS IS HIGH.

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

FRANCE + *SYSTEM ANALYSIS + *ANALYTICAL MODEL + FAILURE MODE ANALYSIS + *AVAILABILITY + PROBABILITY + STATISTICAL ANALYSIS

145871
CHAGNOT M
CUPIDON: A CODE DESCRIBING THE THERMAL AND MECHANICAL BEHAVIOR OF A PWR FUEL ROD DURING A LOCA (IN FRENCH)
INSTITUT DE PROTECTION ET DE SURETE NUCLEAIRE, FRANCE
FRRSR-177 +, 9 PPS, PRESENTED AT IAEA SPECIALISTS' MEETING ON FUEL ELEMENT MODELING, MARCH 13-17, 1978

CUPIDON IS A TWO DIMENSIONAL CODE USING A FINITE DIFFERENCE RESOLVING TECHNIQUE. IT CALCULATES THE RADIAL THERMAL PROFILE ACROSS EACH SECTION OF THE ROD, THE STRESS AND CREEP RATE TO WHICH THE CLADDING IS SUBMITTED AND THE RATE OF FORMATION OF THE OXIDE LAYER ON THE SURFACE OF THE CLADDING UNDER STEADY STATE AND TRANSIENT CONDITIONS. AS CLADDING PLASTIC STRAIN INPUT DATA, IT IS USING THE EDGAR-ZY EXPERIMENTAL RESULTS.

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

*FUEL ROD + THERMAL MECHANICAL EFFECT + REACTOR, PWR + ACCIDENT, LOSS OF COOLANT + CREEP + TRANSIENT + FRANCE + DEFORMATION + COMPUTER PROGRAM

143911
LE COQ G + RAYMOND P
CRITICAL FLOW AND FLOW BLOCKAGE PHENOMENON FOR A TWO PHASE FLOW (IN FRENCH)
CEA CENTRE D'ETUDES NUCLEAIRES DE SACLAY, FRANCE
SERNA/S-341 + FRRSR-161 +, 25 PPS, 9 FIGS, 7 REFS, APRIL 1978

A FLOW IS DEFINED AS A CRITICAL FLOW IN A CROSS SECTION, WHEN ANY DOWNSTREAM PERTURBATION CANNOT BE PROPAGATED IN THE UPSTREAM FLOW. THEN, THE FLUID VELOCITY IS SONIC. FOR THE SIX EQUATIONS MODEL, WITHOUT DIFFERENTIAL TERMS FOR THE TRANSFERS BETWEEN PHASES, THIS DEFINITION LEADS TO A TWO PHASE FLOW MACH NUMBER. HOWEVER, EXPERIMENTS SHOW THAT BEFORE THE FLOW BECOMES CRITICAL, AN IMPORTANT VARIATION OF THE DOWNSTREAM CONDITIONS DOESN'T HAVE ANY SIGNIFICANT EFFECT ON THE UPSTREAM FLOW. WE CALL THIS PHENOMENON: FLOW BLOCKAGE. FROM THE SIX EQUATIONS MODEL, WE DEFINE A FUNCTION WHICH DEPENDS ON LOCAL THERMODYNAMIC PROPERTIES AND ALGEBRAIC TRANSFER TERMS BETWEEN PHASES, AND WHICH PERMITS TO DESCRIBE THE FLOW BLOCKAGE PHENOMENON.

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

FLOW, CRITICAL + FLOW, TWO PHASE + REACTOR, PWR + ACCIDENT, LOSS OF COOLANT + FLOW BLOCKAGE

143928
RAYMOND P
FLOW AND HEAT TRANSFER THERMODYNAMIC MODELISATION DURING THE REFLOODING PHASE OF A PWR'S CORE (IN FRENCH)
CEA CENTRE D'ETUDES NUCLEAIRES DE SACLAY, FRANCE
CEA-N-2025 + FRRSR-170 +, 148 PPS, TABS, FIGS, APRIL 1978

SOME GENERALITIES ABOUT L.O.C.A. ARE FIRST RECALLED. THE FRENCH EXPERIMENTAL STUDIES ABOUT EMERGENCY CORE COOLING SYSTEM ARE BRIEFLY DESCRIBED. THE DIFFERENT HEAT TRANSFER MECHANISMS TO TAKE INTO ACCOUNT, ACCORDING TO THE FLOW PATTERN IN THE DRY ZONE, AND THE CORRELATIONS OR METHODS TO CALCULATE THEM, ARE DEFINED. THEN THE THERMODYNAMIC CODE COMPUTER: FLIRA, WHICH DESCRIBES THE REFLOODING PHASE, AND A MODELISATION TAKING INTO ACCOUNT THE DIFFERENT FLOW PATTERNS ARE

143928 *CONTINUED*

DISCUSSED. A FIRST INTERPRETATION OF ERSEC EXPERIMENTS WITH A TUBULAR TEST SECTION SHOWS THAT IT IS POSSIBLE, WITH THIS MODELLISATION AND SOME CLASSICAL HEAT TRANSFER CORRELATIONS, TO DESCRIBE THE REFLLOODING PHASE.

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

ACCIDENT, LOSS OF COOLANT + EMERGENCY COOLING SYSTEM + CORE REFLLOODING + REACTOR, PWR + FLOW, TWO PHASE + HEAT FLUX, DRYOUT + DROPLET

143756

BROYERE M + LE BERRE F

STUDY OF THE STABILITY OF VARIOUS SYSTEMS AND DESCRIPTION OF EQUATIONS FOR HYDRODYNAMIC ANALYSIS (IN FRENCH)
CEA CENTRE D'ETUDES NUCLEAIRES DE CADARACHE, FRANCE
DRE/STRE/LET/78/138 + FRNSR-156 +, 9 PPS, APRIL 19, 1978

SPECIFIES THE STEADY STATE FIELDS OF THE TWO FIRST HYDRODYNAMICS EQUATIONS WHEN DISCRETIZED WITH A THREE POINT EXPLICIT OR IMPLICIT SCHEME.

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

FRANCE + *ANALYTICAL TECHNIQUE + NUMERICAL METHOD + *HYDRODYNAMIC ANALYSIS + SYSTEM ANALYSIS

143751

ABRAMSON D + MENNESSIER O

MODEL FOR THE CALCULATION OF THE RATE OF VOIDING DURING A RAPID FAILURE (BLOWDOWN) COMPARISON OF SEVERAL MODELS (IN FRENCH)
CEA CENTRE D'ETUDES NUCLEAIRES DE CADARACHE, FRANCE
DRE/STRE/LET/78/006 + FRNSR-157 +, 38 PPS, FIGS, 26 REFS, APRIL 19, 1978

THE FIRST PART IS A DESCRIPTION OF THE DIFFERENT METHODS OF CALCULATING THE VOID FRACTION. THESE ARE CLASSIFIED INTO FIVE CATEGORIES ACCORDING TO THE TYPE OF CORRELATION USED: (1) MARTINELLI-NELSON'S MODEL, (2) SLIP RATIO MODELS, (3) VOLUMETRIC QUALITY MODELS, (4) DRIFT FLUX MODELS, (5) RELATIVE VELOCITY MODELS. THE SECOND PART PRESENTS THE METHOD USED TO FIND A CORRELATION OF RELATIVE VELOCITY THAT AGREES WITH THE VOID FRACTION MEASUREMENTS MADE AT GRENOBLE ON PATRICIA LOOP. THE FORM OF THE CORRELATION IS GIVEN.

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

*VOID + VOID FRACTION + BLOWDOWN + COMPARISON + *CORRELATION + MODEL + FLOW, TWO PHASE + COMPARISON, THEORY AND EXPERIENCE

143914

ROUSSEAU JC + RIEGEL B

SUPER CANON EXPERIMENTS (IN ENGLISH)

COMMISSARIAT A L'ENERGIE ATOMIQUE, FRANCE

FRNS-145 +, 22 PPS, FROM CSNI SPECIALISTS MEETING ON TRANSIENT TWO-PHASE FLOW: PARIS, FRANCE, JUNE 1978

THIS PAPER CONTAINS A DESCRIPTION OF EXPERIMENTS MEASURING THE MEAN VOID FRACTION IN A TOTAL CROSS SECTION OF PIPE USING THE NEUTRON SCATTERING METHOD. CALIBRATION TESTS WERE PERFORMED IN STEADY STATE AT VARIOUS VOID FRACTIONS AND DIFFERENT VOID DISTRIBUTIONS. IT IS DEMONSTRATED THAT EVEN FOR LARGE PIPE WALL THICKNESSES SUPPORTING HIGH PRESSURES, THE NEUTRON SCATTERING METHOD ALLOWS GOOD MEAN VOID FRACTION MEASUREMENTS WITH HIGH CONTRASTS. THIS METHOD IS USED FOR VOID FRACTION EVALUATION DURING A FAST BLOWDOWN EXPERIMENT.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D.C. 20551 (8 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

*FLOW, TWO PHASE + *BLOWDOWN + *MEASUREMENT + NEUTRON + *VOID FRACTION

143753

GRANDOTTO M

CALCULATIONAL METHOD FOR TWO DIMENSIONAL VISCOUS FLOW USING FINITE ELEMENT METHOD (IN FRENCH)

CEA CENTRE D'ETUDES NUCLEAIRES DE CADARACHE, FRANCE

DRE/STRE/LET/78/153 + FRNSR-150 +, 27 PPS, FIGS, 16 REFS, JUNE 1978

A COMPUTATIONAL METHOD TO STUDY TWO DIMENSIONAL VISCOUS FLOW IS PRESENTED. NAVIER-STOKES EQUATIONS ARE SOLVED USING A FINITE ELEMENT METHOD. THE FOLLOWING POINTS ARE GIVEN IN DETAIL: MATHEMATICAL THEORY, NUMERICAL ALGORITHM, COMPUTATIONAL STRUCTURE, AND CALCULATIONS.

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

FRANCE + *NUMERICAL METHOD + MATHEMATICAL TREATMENT + *FLOW + FLOW THEORY AND EXPERIMENTS + FLOW STABILITY

143210

PORRACCHIA A

STUDY OF THE RANGE OF VELOCITY OF GAS INSIDE A BUBBLE RISING THROUGH A LIQUID CODE C.B.U. I (IN FRENCH)

CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE

S.E.S.T.R., 12 + FRNSR-152 +, 30 PPS, 7 FIGS, 13 REFS, JUNE 20, 1978

143710 *CONTINUED*

THIS DOCUMENT DESCRIBES A CODE THAT GIVES ACCESS TO THE FIELD OF SPEEDS INSIDE A GAS BUBBLE RISING THROUGH A LIQUID. DURING A SECOND PHASE, WHICH IS BEING DEVELOPED, IT WILL EXPLAIN THE INFLUENCE EXERCISED BY THE MOTION OF THE GAS ON THE BEHAVIOUR OF PARTICLES OR AEROSOLS PLACED IN THE BUBBLE. (ML*)

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

*BUBBLE * COMPUTER PROGRAM * AEROSOL * FRANCE * GAS

147103

RUIT JC + LEFORT G

THE SAFETY OF FRENCH INSTALLATIONS FOR THE STORAGE OF IRRADIATED FUEL ELEMENTS FROM LIGHT WATER REACTORS (IN FRENCH)

CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE

CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE OSN 235 + FRRSR-184 +, 9 PPS, FROM CONFERENCE HELD IN MADRID, JUNE 20-23, 1978

THE OPERATION OF THE LWRs REQUIRES THE STORAGE OF IRRADIATED FUEL ELEMENTS IN COOLING POOLS WHICH HAVE ACCESS DIRECTLY FROM THE REACTOR CORE. AFTER TRANSPORTATION TO THE REPROCESSING PLANTS, THE STORAGE MUST BE CONTINUED IN STORAGE POOLS LOCATED AT THE ENTRY OF THE PLANT. REQUIREMENTS FOR SAFE STORAGE HAVE BEEN BASED ON EXPERIENCE ACQUIRED RELATIVE TO NORMAL OPERATING CONDITIONS: COOLING, CONTAINMENT, SHIELDING, HANDLING, WASTE AND EFFLUENTS PROCESSING, ETC. THESE REQUIREMENTS ARE DISCUSSED.

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

FRANCE * SPENT FUEL * FUEL STORAGE * SAFETY ANALYSIS * ACCIDENT ANALYSIS * ON SITE * FUEL REPROCESSING * REACTOR, LWR

143875

CAUMEITE P + CHEISSOUX JL + GARCIA JL

CALCULATING PLASTIC DEFORMATION OF STRUCTURES (IN FRENCH)

CEA CENTRE D'ETUDES NUCLEAIRES DE CADARACHE, FRANCE

DRE/STREZ/LMA 78/154 + FRRSR-149 +, 54 PPS, FIGS, JULY 1978

THE METHODS FOR CALCULATING PLASTIC DEFORMATION OF STRUCTURES ARE PRESENTED, AS THEY ARE USED IN THE CASTER SYSTEM, DEVELOPED BY THE DEPARTEMENT DES ETUDES MECANQUES ET THERMIQUES AT SACLAY. BASICS ON THE THEORY OF PLASTICITY, THE FINITE ELEMENT FORMULATION AND THE ALGORITHMS OF PLASTICITY, IN THE CASE OF ISOTROPIC HARDENING, ARE PRESENTED. (FAH)

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

FRANCE * PLASTICITY * DEFORMATION * STRUCTURE * MATHEMATICAL TREATMENT

143777

BLIN A + CARNING A + GEORGIN JP + SIGNURET JP

USE OF MARKOV PROCESSES FOR RELIABILITY PROBLEMS (IN ENGLISH)

CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE

OSN-234(E) + FRRSR-167 +, 28 PPS, 5 FIGS, 8 REFS, JULY 1978

IT IS NOT POSSIBLE TO USE A CLASSICAL METHOD SUCH AS FAULT TREE ANALYSIS TO ASSESS THE RELIABILITY OR THE AVAILABILITY OF TIME-EVOLUTIVE SYSTEMS. STOCHASTIC PROCESSES HAVE TO BE USED AND AMONG THEM THE MARKOV PROCESSES ARE THE MOST INTERESTING ONES. THE BASIC THEORY OF MARKOV PROCESSES IS DESCRIBED IN THIS PAPER IN CONNECTION WITH RELIABILITY PROBLEMS. THEN THE MARK-GE CODE DEVELOPED BY THE FRENCH CEA IS PRESENTED WITH AN EXAMPLE OF RELIABILITY ASSESSMENT OF A COMPLEX SYSTEM: AC POWER SUPPLY OF A 900 MW PWR. (EWH)

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

*RELIABILITY ANALYSIS * ANALYTICAL TECHNIQUE * FRANCE * COMPUTER PROGRAM * ELECTRIC POWER, AUXILIARY * REACTOR, PWR * MATHEMATICAL TREATMENT * ACCIDENT, LOSS OF POWER * FAILURE, COMMON MODE

143755

BONNETON M

DATA REDUCTION OF THE FIRST TEST SERIES OF BLOWDOWN ON A TUBULAR TEST SECTION ON OMEGA LOOP (IN FRENCH)

CEA CENTRE D'ETUDES NUCLEAIRES DE GRENOBLE, FRANCE

TT 580 + FRRSR-162 +, 80 PPS, 3 TABS, 50 FIGS, 2 REFS, AUG. 1978

THIS REPORT DEALS WITH THE FIRST BLOWDOWN TEST SERIES, OPERATED ON THE OMEGA LOOP WITH A VERTICAL, TUBULAR HEATED TEST SECTION. THE GENERAL METHOD OF DATA REDUCTION IS ANALYSED AND A CRITICAL STUDY OF ALL THE MEASUREMENTS IS MADE: PRESSURES, VOID FRACTIONS, MASS FLOW RATES, FLUID TEMPERATURES, AND WALL TEMPERATURES. FOUR TYPICAL BLOWDOWN TESTS, WHICH ARE THE MOST REPRESENTATIVE OF THE SERIES, ARE PRESENTED. TWO OF THEM CORRESPOND TO DOWNSTREAM BREAKS (LARGE AND SMALL); THE TWO OTHERS CORRESPOND TO UPSTREAM BREAKS (LARGE AND SMALL).

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

FRANCE * DATA PROCESSING * EXPERIMENT * BLOWDOWN * PRESSURE, INTERNAL * VOID FRACTION * FLOW * TEMPERATURE * MEASUREMENT * ACCIDENT, LOSS OF COOLANT

143778
BLIN A + DUJHEMIN B + CARNINO A
PATREC, A COMPUTER CODE FOR FAULT TREE CALCULATIONS (IN ENGLISH)
CEA DEPARTMENT OF SURETE NUCLEAIRE, FRANCE
DSN 235(E) + FRRSR-168 +, 13 PPS, 3 FIGS, 25 REFS, SEPT, 1978

A COMPUTER CODE FOR EVALUATING THE RELIABILITY OF COMPLEX SYSTEMS USING FAULT TREES IS DESCRIBED IN THIS PAPER. IT USES PATTERN RECOGNITION APPROACH AND PROGRAMMING TECHNIQUES FROM IBM PL/I LANGUAGE. IT CAN TAKE INTO ACCOUNT MANY OF THE PRESENT DAY PROBLEMS: MULTI-DEPENDENCIES TREATMENT, DISPERSION IN THE RELIABILITY DATA PARAMETERS, INFLUENCE OF COMMON MODE FAILURE . . .
* THE CODE HAS BEEN RUNNING STEADILY FOR TWO YEARS. (EWH)

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA, 22161
*COMPUTER PROGRAM + *FAULT TREE ANALYSIS + RELIABILITY ANALYSIS + FAILURE, COMMON MODE + ANALYTICAL TECHNIQUE
+ PROBABILITY + FRANCE

143929
FRANK R + RIQUE R + BOURGINE R
BLOWDOWN OF A PART OF THE LOOP OMEGA INCLUDING A 36 DIRECT HEATED ROD BUNDLE (IN FRENCH)
COMMISSARIAT A L'ENERGIE ATOMIQUE, FRANCE
TT 152 + FRRSR-169 +, 107 PPS, 44 FIGS, 2 REFS, SEPT, 1978

THIS REPORT DESCRIBES A SET OF BLOWDOWN EXPERIMENTS PERFORMED WITH A 36 ROD BUNDLE TEST SECTION WHICH SIMULATE A LOSS OF COOLANT ACCIDENT IN A PRESSURIZED WATER REACTOR. THE MASS FLOW RATE AND VOID FRACTION ARE MEASURED USING A VENTURI AND A GAMMA-DENSITOMETER. THE CALCULATION OF THE MASS FLOW RATE IS MADE ASSUMING THAT THE FLOW IS HOMOGENEOUS. IN A FIRST PART WE DESCRIBE THE EXPERIMENTAL SET UP. IN A SECOND PART THE MEASUREMENTS OF PRESSURE, TEMPERATURE, MASS FLOW RATE AND VOID FRACTION ARE DESCRIBED. IN A THIRD PART WE DESCRIBE THE PROCEDURE WHICH IS USED TO CORRECT THE MEASUREMENTS AND PROCESS THE EXPERIMENTAL DATA. WE FINALLY GIVE A PHYSICAL ANALYSIS OF SOME OF THE PARAMETER EVOLUTIONS IN THE DIFFERENT CASES WHICH WERE EXAMINED. THE MAJOR RESULT OF THESE FIRST EXPERIMENTS IS THAT THE MEASUREMENT OF THE MASS FLOW RATE AND VOID FRACTION IS MADE WITH AN ACCURACY BETTER THAN 5%.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA, 22161
*ACCIDENT, LOSS OF COOLANT + *REACTOR, PWR + *BLOWDOWN + *MEASUREMENT + TEMPERATURE + VOID FRACTION + PRESSURE DROP

146872
BOULAIS J + BRUARD D + ROCHE R
EXPERIMENTAL TESTS ON RATCHET OF 304 AUSTENITIC STEEL, AT ROOM TEMPERATURE (IN FRENCH)
CEA CENTRE D'ETUDES NUCLEAIRES DE SACLAY, FRANCE
CEA-N-2058 + FRRSR-185 +, 50 PPS, 8 TABS, 29 FIGS, 18 REFS, SEPT, 1978

THERE IS A NEED FOR EXPERIMENTAL TESTS ON BASIC STRUCTURES EASY TO USE TO DETERMINE MATERIAL CHARACTERISTICS. TESTS ON THIN TUBULAR SPECIMEN ARE VERY INTERESTING BECAUSE STRESS, STRAIN AND TEMPERATURE FIELDS ARE UNIFORM. THE PRIMARY STRESS P IS AN AXIAL TENSILE ONE (DEAD WEIGHT), THE SECONDARY STRESS, WITH DELTA Q RANGE, IS DUE TO A CYCLIC ANGLE CONTROLLED TWIST. THE INCREMENTAL ELONGATION IS OBTAINED AS A FUNCTION OF THE NUMBER OF CYCLES N FOR DIFFERENT VALUES OF P AND DELTA Q. DIAGRAMS REPRESENTING THE ISOCURVES OF CUMULATED ELONGATION (FOR A GIVEN NUMBER OF CYCLES) AS A FUNCTION OF P AND DELTA Q ARE SHOWN.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA, 22161
STEEL, STAINLESS + TESTING + DEFORMATION + ANALYTICAL TECHNIQUE + FRANCE

143383
JANVIER JC
CONTAMINATION OF A PRESSURIZED WATER REACTOR'S PRIMARY CIRCUIT BY FUEL RODS SHOWING MANUFACTURING FAULTS (IN FRENCH)
CEA CENTRE D'ETUDES NUCLEAIRES DE GRENOBLE, FRANCE
DMG 98778 + FRRSR-148 +, 15 PPS, 4 FIGS, 5 REFS, SEPT, 11, 1978

INCREASING IMPORTANCE IS BEING ATTACHED TO CONTAMINATION OF THE PRIMARY LOOP OF PWR'S RESULTING FROM FUEL ELEMENT FAILURES, ESPECIALLY THOSE THAT ARE MANUFACTURER'S DEFECTS. A RESEARCH PROGRAM ON THESE FAILURES IS BEING CARRIED OUT AT THE CENTRE D'ETUDES NUCLEAIRES, AT GRENOBLE. WITH THE OBJECTIVE OF ANALYZING THE BEHAVIOR OF FAILED FUEL ELEMENTS, A DISTINCTION IS MADE BETWEEN TWO TYPES OF FUEL ELEMENT FAILURES, ACCORDING TO WHETHER PRIMARY WATER PENETRATES INTO THE FUEL ROD AS SOON AS CIRCUIT PRESSURIZATION TAKES PLACE (MANUFACTURE DEFECT), OR FAILURE OCCURS WHILE IN OPERATION. THE EMISSION OF GASEOUS FISSION PRODUCTS AND HALOGENS HAS BEEN ANALYSED ACCORDING TO VARIOUS OPERATION PATTERNS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA, 22161
R AND D PROGRAM + *FAILURE, FUEL ELEMENT + *MAIN COOLING SYSTEM + *CONTAMINATION + FISSION GAS RELEASE + FAILURE, FABRICATION ERROR + FAILURE, INHERENT + REACTOR, PWR + FRANCE

148734

14714 *COURTINCOFF

*PIRETA X

ANA EXPERIMENTAL AND THEORETICAL STUDY OF THE BLOWDOWN OF THE SECONDARY SIDE OF A STEAM GENERATOR (IN FRENCH)
FRAMATOME, FRANCE
TRXCT/78/340 + FRRSR-179 +, 7 PPS, 1 FIG, SEPT, 29, 1978

IN ORDER TO ASSESS THE HYDRAULIC FORCES ON THE STEAM GENERATOR (SG) INTERNALS AND THE ENERGY RELEASED IN THE CONTAINMENT THE DESIGNER MUST STUDY THE BLOWDOWN OF THE SECONDARY SIDE OF THE STEAM GENERATOR WHICH MAY OCCUR AS A CONSEQUENCE OF A RUPTURE IN THE STEAM LINE. THIS PAPER SUMMARIZES SOME THEORETICAL AND EXPERIMENTAL RESEARCH WORK PERFORMED AT FRAMATOME IN COLLABORATION WITH THE CEA IN ORDER TO STUDY THE BLOWDOWN OF THE SECONDARY SIDE OF A STEAM GENERATOR. IT SUMMARIZES THE WORK RELATED TO THE STUDY AND THE MODELLING OF THE EARLY PART OF THE BLOWDOWN.

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

COOLING SYSTEM, SECONDARY + HYDRAULIC ANALYSIS + BLOWDOWN + DESIGN CRITERIA + ACCIDENT, STEAM LINE RUPTURE + ANALYTICAL MODEL + COMPUTER PROGRAM + FRANCE + STEAM GENERATOR

144595

KURKA G + HAPPER A + CHENEBAULT P

ANALYSIS OF THE FISSION PRODUCT RELEASE FROM A DEFECTED FUEL ROD - EFFECT OF THERMAL CYCLING (IN ENGLISH)
CEA CENTRE D'ETUDES NUCLEAIRES DE GRENOBLE, FRANCE
FRRSR-147 +, 8 PPS, 1 TAB, 3 FIGS, OCT, 1978

THE FOLLOWING EXPERIMENTAL WORK IS DEALING WITH THE STUDY OF THE MECHANISM OF FISSION PRODUCT RELEASE INTO THE PRIMARY CIRCUIT OF A PWR FROM FUEL ROD PRESENTING AN INITIAL DEFECTIVE LEAK TEST. EACH RAPID POWER VARIATION WAS FOLLOWED BY A PEAK OF ACTIVITY, THE AMPLITUDE OF WHICH WAS MORE IMPORTANT FOR IODINE ISOTOPES THAN FOR RARE GASES. THIS EFFECT CAN BE EXPLAINED BY VARIOUS HYPOTHESES: IODINE ISOTOPES TRAPPED ON THE FUEL OR CLADDING SURFACE, CAN BE RELEASED BY WATER FLOWING INTO AND OUT OF THE FUEL ROD; AND STRESSES ON THE FUEL BRING A PARTIAL RELEASE OF THE FISSION GASES ACCUMULATED ON GRAIN BOUNDARIES.

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

FISSION PRODUCT RELEASE + FISSION PRODUCT TRANSPORT + THERMAL EXPERIMENT + FRANCE

143138

DUCCO J + GOBERT Y

PROTECTION OF NUCLEAR POWER PLANTS (NPPS) AGAINST EXTERNAL EVENTS: EARTHQUAKES, FIRES, EXPLOSIONS AND AIRCRAFT CRASHES (IN ENGLISH)
COMMISSARIAT A L'ENERGIE ATOMIC, FRANCE + ELECTRICITE DE FRANCE
FRRSR-154 +, 10 PPS, PAPER PRESENTED AT SESSION D477 OF NUCLEX '78: 5TH INTERNATIONAL FAIR & TECHNICAL MEETINGS OF NUCLEAR INDUSTRIES; BASEL, SWITZERLAND, OCT, 3-7, 1978

THIS PAPER OUTLINES PRESENT GENERAL PRACTICE IN FRANCE AS CONCERNS THE SAFETY ANALYSIS OF NUCLEAR POWER PLANTS IN RELATION TO EXTERNAL IMPACTS DUE TO EARTHQUAKES, FIRES, EXPLOSIONS AND AIRCRAFT CRASHES. SOME TRENDS FOR THE FUTURE RESULTING FROM STUDIES NOW UNDER WAY IN THIS FIELD ARE SKETCHED.

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

SWITZERLAND + FRANCE + EARTHQUAKE + FIRE + EXPLOSION + AIRCRAFT + IMPACT SHOCK + SAFETY EVALUATION + POWER PLANT, NUCLEAR

143870

DUPRESNE J + CARNINO A + QUERO J + LUCIZ AC

FRACTURE PROBABILITY EVALUATION OF A LWR PRESSURE VESSEL (IN ENGLISH)

CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE

FRRSR-158 +, 13 PPS, PAPER PRESENTED AT SESSION D4 OF NUCLEX '78: 5TH INTERNATIONAL FAIR & TECHNICAL MEETINGS OF NUCLEAR INDUSTRIES; BASEL, SWITZERLAND, OCT, 3-7, 1978

IN ADDITION TO THE EVALUATION OF FRACTURE PROBABILITY OF A NUCLEAR PRESSURE VESSEL, THIS PROGRAM IS CARRIED OUT, TO GET THE FOLLOWING INFORMATIONS: ASSESSMENT OF THE INDIVIDUAL EFFECTS OF THE MAIN PARAMETERS ON THE FINAL RESULT; COMPARISON OF THE VARIOUS POSSIBILITIES IN THE FIELD OF FABRICATION OR OPERATION; AND BASIS FOR THE DETERMINATION OF THE INTERVALS FOR IN-SERVICE INSPECTIONS. IT IS EXPECTED THAT THIS WORK WILL BE EXTENDED TO A COMPARISON OF THE FAILURE PROBABILITY OF THE DIFFERENT COMPONENTS OF THE REACTOR COOLANT PRESSURE BOUNDARY. (FAH)

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

SWITZERLAND + FRANCE + PROBABILITY + FAILURE + PRESSURE VESSELS + REACTOR, LWR

147021

COGNE F + TANGUY P

USE OF PROBABILISTIC METHODS IN THE SAFETY EVALUATION OF NUCLEAR INSTALLATIONS (IN ENGLISH)

CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE

DSN 238(E) + FRRSR-188 +, 17 PPS, PRESENTED AT NUCLEX '78: BASEL, SWITZERLAND, OCT, 3-7, 1978

DISCUSSES THE ROLE AND EXTENT TO WHICH PROBABILISTIC METHODS FOR SAFETY EVALUATION OF NUCLEAR

147021 *CONTINUED*

POWER PLANTS IN THE LICENSING PROCESS OF SUCH PLANTS, A CLASSIFICATION SYSTEM IS PRESENTED FOR USING PROBABILISTIC METHODS IN TERMS OF WHAT IS KNOWN AND WHAT NEEDS TO BE DONE. (A). USE WASH-1400 METHODOLOGY AS AN ASSISTANCE TO SAFETY ASSESSMENTS WITH THE PRESENT SAFETY RULES. (B). INTRODUCE NEW PROBABILISTIC SAFETY CRITERIA AND/OR REPLACE SOME DETERMINISTIC CRITERIA BY PROBABILISTIC ONES.

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

*ACCIDENT, PROBABILITY OF + PROBABILITY + *ANALYTICAL TECHNIQUE + *LICENSING PROCESS + SAFETY ANALYSIS + FRANCE + REACTOR, LMFR

144415

BROSSARD J + DUCC J + GUERT T

EXPERIMENTAL STUDY OF THE OVERPRESSURE GENERATED BY THE DETONATION OF SPHERICAL AIR-HYDROGEN GASEOUS MIXTURES (IN ENGLISH)

CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE

FRRSR-155 +. 15 PPS, FROM ENS/ANS TOPICAL MEETING OF NUCLEAR POWER REACTOR SAFETY; BRUSSELS, BELGIUM, OCT. 16-19, 1978

THE CHARACTERISTICS OF THE PRESSURE WAVES TRANSMITTED BY DETONATION OF GASEOUS MIXTURES TO THE SURROUNDING AIR WERE MEASURED BY TESTS MADE NEAR THE GROUND LEVEL IN 1 TO 54 M CUBED SPHERICAL BALLOONS CONTAINING AIR-ACETYLENE OR AIR-ETHYLENE MIXTURES. AS CONCERNS THE PEAK OVERPRESSURE DELTA P, A THEORETICAL DIMENSIONAL ANALYSIS IN ACCORDANCE WITH THE EXPERIMENTAL RESULTS SHOWS THAT DELTA P CAN BE EXPRESSED AS A FUNCTION OF TWO INDEPENDENT VARIABLES, WHICH ARE THE RADIAL DISTANCE R AND THE VOLUME V OF THE BALLOON. A SEMI-EMPIRICAL FORMULA, INCLUDING GROUND EFFECTS, IS PROPOSED AND ITS PRESENT VALIDITY RANGE IS GIVEN.

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

ANALYTICAL MODEL + EXPLOSION + THERMAL EXPERIMENT + THEORETICAL INVESTIGATION + FIRE + PRESSURE PULSE + COMBUSTION + BELGIUM + HYDROGEN + FRANCE

143103

TANGUY P

THE SAFETY OF NUCLEAR REACTORS IN FRANCE (IN ENGLISH)

CEA INST. PROTECTION SURETE NUCLEAIRE

FRRSR-159 +. 6 PPS, PAPER PRESENTED AT ENS/ANS MEETING ON SAFETY OF NUCLEAR POWER REACTORS; BRUSSELS, OCT. 16-19, 1978

THE SPEAKER TALKED ABOUT THE NUCLEAR ENERGY PROGRAM IN FRANCE RELATING HIS COMMENTS TO PWR REACTORS, THE ADVANCED REACTORS, SAFETY, AND RESEARCH. THE PWR REACTORS ARE OF AMERICAN DESIGN BEING 900 MWE AND 1300 MWE PLANTS. SAFETY ASPECTS FOR THE PHENIX AND SUPER PHENIX HAVE BEEN DEVELOPED FROM PWR PHILOSOPHY AND RESEARCH SINCE THERE IS REALLY NO EXPERIENCE RECORDS THAT CAN BE RELIED UPON. RESEARCH OF NUCLEAR SAFETY IS GIVEN GREAT IMPORTANCE IN FRANCE. (FAH)

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

FRANCE + REACTOR, PWR + SUPERPHENIX (LMFR) + R AND D PROGRAM + SAFETY PROGRAM

143758

RECREUX M + SUREAU H + COURTAUD M + THIBAudeau J

FRENCH THERMO-HYDRAULIC STUDIES FOR THE DEVELOPMENT OF SAFETY ADVANCED CODE FOR PWR (IN FRENCH)

COMMISSARIAT A L'ENERGIE ATOMIQUE, FRANCE

FRRSR-163 +. 11 PPS, PAPER PRESENTED AT ENS/ANS MEETING ON SAFETY OF NUCLEAR POWER REACTORS; BRUSSELS, OCT. 16-19, 1978

AN ADVANCED CODE IS BEING WRITTEN IN FRANCE BY CEA-EDF AND FRAMATOME. IN THE THERMOHYDRAULIC FIELD SOME IMPROVEMENTS HAVE BEEN MADE IN THIS CODE WHICH ARE PRESENTED IN THE FOLLOWING SECTIONS. THIS CONCERNS TWO-PHASE FLOW MODELING, PUMP MODELING, HEAT TRANSFER DURING BLOWDOWN AND REFLUOD, SAFETY INJECTION AND SYSTEM CODE DEVELOPMENT.

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

FRANCE + *THERMAL HYDRAULIC ANALYSIS + REACTOR, PWR + FLOW, TWO PHASE + PUMPS + MODEL + BLOWDOWN + SAFETY INJECTION + HEAT TRANSFER + CORE REFLUODING

143339

BOUSCATIE F + FOURCADE P + GEORGIN JP + ROY C

MODEL OF THE FAILURE RATES OF THE VALVES OF ST. LAURENT DES EAUX POWER PLANT ACCORDING TO INFLUENTIAL PARAMETERS

CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE

FRRSR-164 +. 17 PPS, PAPER PRESENTED AT ENS/ANS MEETING ON SAFETY OF NUCLEAR POWER REACTORS; BRUSSELS, OCT. 16-19, 1978

THE STUDY IS A SEQUENCE OF CONVENTIONAL STATISTICAL STUDIES PERFORMED AT THE DEPARTEMENT DE SURETE NUCLEAIRE OF THE COMMISSARIAT A L'ENERGIE ATOMIQUE ON THE INCIDENT FILE OF THE ST-LAURENT DES EAUX NUCLEAR POWER PLANT. ALTHOUGH THIS FILE HAD NOT BEEN DESIGNED AT THE START IN THE SENSE OF A RELIABILITY FILE, IT MADE IT POSSIBLE, TO CLASSIFY VALVES ACCORDING TO PARAMETERS (CONTROL MODE, FLUID GOING THROUGH THE VALVE,...) AND TO GIVE FOR EACH TYPE THE FAILURE RATES AND THE

143119 *CONTINUED*
ASSOCIATED CONFIDENCE INTERVALS.

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

FRANCE + POWER PLANT, NUCLEAR + ANALYTICAL MODEL + FAILURE MODE ANALYSIS + VALVES + FAILURE, EQUIPMENT + RELIABILITY ANALYSIS + INCIDENT COMPILATION

143779

CARNINO A + NAMY P + LLORY M + QUENEY R

A FIRST APPROACH OF THE RARE EVENT PROBLEM BY THE STUDY OF THE RELIABILITY OF THE PROTECTION SYSTEM OF THE FESSENHEIM 1 PWR REACTOR (IN ENGLISH)

CEA DEPARTMENT DE SURETE NUCLEAIRE, FRANCE + FRAMATOME, FRANCE

FRSR-185 +, 19 PPS, PAPER PRESENTED AT ENS/ANS MEETING ON SAFETY OF NUCLEAR POWER REACTORS; BRUSSELS, OCT. 16-19, 1978

THE STUDY PRESENTED CORRESPONDS TO CONCERNS SPECIFIC TO THE NUCLEAR SAFETY DEPARTMENT OF THE "COMMISSARIAT A L'ENERGIE ATOMIQUE" ON THE RARE EVENT PROBLEM. FOR THE SAFETY ASSESSMENT OF NUCLEAR POWER PLANTS EVENTS HAVING THE OCCURENCE PROBABILITIES OF VALUES COMPRISED BETWEEN 10 (10⁻⁶) AND 10 (10⁻⁸) PER REACTOR YEAR AND WHICH COULD RESULT IN MORE OR LESS SERIOUS CONSEQUENCES ARE CONSIDERED. (EWH)

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

RELIABILITY ANALYSIS + FAILURE, COMMON MODE + REACTOR PROTECTION SYSTEM + REACTOR, PWR + FRANCE + ANALYTICAL TECHNIQUE

147131

DRUYERE R

CHARACTERISTICS AND RESOLUTIONS OF THE SYSTEM OF EQUATIONS DERIVED FROM PARTIAL HYPERBOLICS (IN FRENCH)

DEPARTMENT DES REACTEURS A L'EAU, FRANCE

DRF/STHE/LMATA 76/173 + FRSR-173 +, 21 PPS, REFS, NO. 72, 1978

DISCUSSES A WELL-POSED PROBLEM OF AN HYPERBOLIC SET OF PARTIAL DIFFERENTIAL EQUATIONS WITH TWO VARIABLES AND BOUNDARIES CONDITIONS. TWO METHODS OF SOLUTIONS ARE PRESENTED: CHARACTERISTICS AND FINITE DIFFERENCE.

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

FRANCE + ANALYTICAL TECHNIQUE + MATHEMATICAL TREATMENT + MATHEMATICS, DIFFERENCE EQUATION + EQUATION

149000

CROIX JM + LIEGEOIS A

STUDY ON THE CONDENSATION OF AIR AND STEAM MIXTURES, IN TRANSIENT CONDITIONS, ON A STAINLESS STEEL TEST SECTION (IN FRENCH)

CEA CENTRE D'ETUDES NUCLEAIRES DE GRENOBLE, FRANCE

TT-596 + FRSR-187 +, 55 PPS, 15 FIGS, DEC, 1978

THE OBJECTIVE IS TO OBTAIN HEAT TRANSFER COEFFICIENT DATA DURING CONDENSATION OF STEAM IN TRANSIENT CONDITIONS AND IN PRESENCE OF AIR TO GIVE SOME INFORMATIONS FOR THE COMPUTATION OF PRESSURE TRANSIENT IN THE CONTAINMENT OF A PWR DURING A LOCA. THE MEASUREMENTS ARE PERFORMED IN THE ECOTRA INSTALLATION. A 16 CM DIAMETER TEST SECTION IS MAINTAINED AT ROOM TEMPERATURE AND INSULATED BY A MASK FROM A STEADY STATE FLOW OF AN AIR-STEAM MIXTURE. WHEN THE MASK IS SUDDENLY REMOVED, THERE IS CONDENSATION IN TRANSIENT CONDITIONS ON THE SURFACE OF THE WALL. TEMPERATURES AT THE SURFACE AND INSIDE THE WALL ARE MEASURED FROM WHICH ARE CALCULATED HEAT FLUX DENSITIES AND HEAT TRANSFER COEFFICIENTS.

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

HEAT TRANSFER + HEAT TRANSFER, TWO PHASE + AIR + STEAM + PRESSURE TRANSIENT + HEAT TRANSFER COEFFICIENT

143900

PINET R + JEANDEY CH

EXPERIMENTAL STUDY OF CRITICAL TWO-PHASE FLOW (IN ENGLISH)

COMMISSARIAT A L'ENERGIE ATOMIQUE, FRANCE

FRSR-160 +, 21 PPS, FROM OECD SPECIALISTS MEETING ON TRANSIENT TWO-PHASE FLOW, PARIS, FRANCE, JUNE 1978

NEW EXPERIMENTAL STUDIES ON CRITICAL TWO PHASE FLOW PERFORMED ON THE MOBY DICK LOOP IN GRENOBLE ARE REPORTED HERE. PREVIOUS EXPERIMENTS CLEARLY DEMONSTRATED THE INFLUENCE OF THERMAL NONEQUILIBRIA BETWEEN THE TEMPERATURES OF WATER AND STEAM. EXTENSIVE EXPERIMENTAL DATA HAVE BEEN OBTAINED FOR TWO PHASE GAS WATER FLOW IN A STRAIGHT DIFFUSER. ACCURATE PRESSURE AND VOID FRACTION PROFILES WERE OBTAINED FOR A WIDE RANGE OF TWO PHASE FLOW RATES AND TEMPERATURES IN THE LOW QUALITY REGION. CRITICITY WAS ALSO EXPERIMENTALLY PROVED. ALTHOUGH THE GEOMETRY AND THE QUALITY RANGE WERE DIFFERENT, RESULTS ARE IN BROAD AGREEMENT WITH THE RESULTS OF SMITH AND AL (1967).

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

FLOW, TWO PHASE + FLOW, CRITICAL + MASS TRANSFER + VOID FRACTION

143078
 GIRARD P + HUNEAU M + HABASSE C + LEYER JC
 FLAME PROPAGATION THROUGH UNCONFINED AND CONFINED HEMISPHERICAL STRATIFIED GASEOUS MIXTURES (IN ENGLISH)
 UNIVERSITE DE POITIEUX, FRANCE
 FRRSR-166 +. 24 PPS, 9 FIGS, 20 REFS, 1978

TO OBSERVE THE NONSTEADY FLAME PROPAGATION ACROSS GASEOUS MIXTURES OF NON UNIFORM COMPOSITION, A TECHNIQUE, BASED ON AN IMPROVEMENT OF THE SOAP BUBBLE METHOD, IS PROPOSED HERE. TWO APPLICATIONS OF THE METHOD ARE PRESENTED. RESULTS, WHICH RELATE MAINLY TO THE CORRELATION BETWEEN THE GENERATED PRESSURE FIELD AND THE FLAME FRONT VELOCITY VARIATIONS INDUCED BY THE CONCENTRATION STEPS INTEND TO DESCRIBE SOME OF THE CONSEQUENCES OF NON UNIFORM COMPOSITION ON THE BLAST EFFECTS OF ACTUAL VAPOR CLOUD EXPLOSIONS. AVAL: NTS

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

*COMBUSTION + *GAS + EXPLOSION + VAPOR PRESSURE + *EXPERIMENT + TESTING + FRANCE

145801
 PONTE R + MAIGNE JP
 STUDY OF POLLUTANT DISPERSION IN WATER AND AIR (IN ENGLISH)
 CEA DEPARTMENT DE SURETE NUCLEAIRE, FRANCE
 DSN 2431E + FRRSR-175 +. 12 PPS, FROM ENS/ANS MEETING ON NUCLEAR POWER SAFETY; BRUSSELS, 1978

THIS REPORT SETS FORTH: 1) THE "PUFF" MODEL USED FOR PREDICTING DISPERSION IN BOTH WATER AND AIR, WITH THE ASSUMPTIONS AND SPECIFIC DATA REQUIRED FOR ITS USE IN THE CASE OF EACH OF THESE MEDIA; 2) A COMPARISON WITH EXPERIMENTAL RESULTS; 3) A COMPARISON FOR AIR DISPERSION, WITH OTHER PREDICTION METHODS SUCH AS THOSE OF PASQUELL-GIFFORD AND LE QUINIO.

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

FRANCE + *DISPERSION + ATMOSPHERIC DIFFUSION + COMPARISON, THEORY AND EXPERIENCE + CONCENTRATION + ATMOSPHERIC POLLUTION + POLLUTION

145869
 DUFRESNE J
 A PROBABILISTIC STUDY OF VESSEL BURST IN LIGHT WATER NSSS (IN FRENCH)
 CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
 DSN 216 + FRRSR-183 +. 112 PPS, FIGS, REFS, 1978

VARIOUS CRITERIA FOR BURSTING WERE ANALYZED, AND TWO METHODS HAVE BEEN SELECTED, THOSE OF TOWNLEY AND MENKLE. THE CRACK PROPAGATION TESTS IN COMPLEX MODE HAVE STARTED, THE SAMPLES AND MEASURING PROCEDURES ARE NOW FULLY DEVELOPED. BIBLIOGRAPHICAL RESEARCH HAS BEEN UNDERTAKEN ON ACCIDENTAL VARIATIONS AFFECTING PRIMARY WATER COMPOSITION IN OPERATING PWRs. 35 NUCLEAR PLANTS WERE ANALYSED BETWEEN 1974 AND 1977; 9 INCIDENTS BEARING ON WATER COMPOSITION WERE NOTED, THE AMPLITUDE OF WHICH WAS RELATIVELY LOW. TAKING THESE DATA INTO ACCOUNT, THE ANTICIPATED INCIDENTS FOR A GIVEN REACTOR HAVE BEEN ESTIMATED, ALONG WITH THE COMPOSITION OF THE WATER TO BE USED DURING FATIGUE TESTS. (FAH)

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

FRANCE + PROBABILITY + PRESSURE VESSELS + FAILURE + REACTOR, LWR + COMPUTER PROGRAM + COOLANT CHEMISTRY

146871
 ROCHE R
 SHORT ANALYSIS OF A PROGRESSIVE DISTORSION PROBLEM (TENSION AND CYCLE TORSION) (IN FRENCH)
 CEA CENTRE D'ETUDES NUCLEAIRES DE SACLAY, FRANCE
 CEA-N-2038 + FRRSR-186 +. 20 PPS, FIGS, 1978

A THIN TUBE IS SUBJECTED TO A CONSTANT TENSILE LOAD AND TO A CYCLIC TWIST. THIS PAPER IS A THEORETICAL ANALYSIS OF THAT CASE. A UNIFORM STRAIN AND STRESS FIELD IS CONSIDERED WITH A CONSTANT TENSILE STRESS P (PRIMARY STRESS) AND A CYCLIC SHEARING STRAIN. THE SHEARING STRAIN IS KNOWN BY THE CORRESPONDING ELASTIC EQUIVALENT STRESS INTENSITY. THE CYCLIC RANGE OF THE STRESS INTENSITY IS DELTA Q (SECONDARY STRESS RANGE). SPECIAL ATTENTION IS GIVEN TO PERFECT PLASTICITY AND BILINEAR KINEMATIC HARDENING. RESULTS ARE PRESENTED, BUT IT IS BELIEVED THAT THESE MATERIAL MATHEMATICAL MODELS ARE SIMPLISTIC AND SPECIAL EXPERIMENTAL TESTS ARE PROPOSED. (FAH)

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

FRANCE + THEORETICAL INVESTIGATION; + TUBING + ANALYTICAL MODEL + DEFORMATION

145870
 ALIX M + ROCHE R
 EXPERIMENTAL TESTS ON BUCKLING OF ELLIPSOIDAL VESSEL HEADS UNDER INTERNAL PRESSURE (IN FRENCH)
 CEA CENTRE D'ETUDES NUCLEAIRES DE SACLAY, FRANCE
 CEA-N-2075 + FRRSR-172 +. 87 PPS, TABS, FIGS, JAN. 1979

EXPERIMENTAL TESTS ON ELLIPSOIDAL VESSEL HEADS HAVE BEEN CONDUCTED AT SACLAY. SEVENTEEN HEADS MADE OUT OF METAL SHEETS, BY COLD WORKING, WERE TESTED. THREE DIFFERENT METALS WERE USED; CARBON STEEL, AUSTENITIC STEEL, AND ALUMINIUM ALLOY. GEOMETRICAL DEFINITION HEADS HAD A GOOD

146879 *CONT. OF*

AXISYMETRIC SHAPE, BUT THE THICKNESS WAS VARYING ALONG THE ELLIPSE. THE THICKNESS WAS MEASURED, AFTER TESTING, ALONG A RADIAL CUT FOR EACH HEAD. MATERIAL CHARACTERISTIC OF EACH HEAD WAS GIVEN BY A TENSILE TEST (STRAIN-STRESS CURVE) MADE ON SAMPLES CUT OUT OF THE TESTED HEAD. THE RESULTS ARE MAINLY THE PRESSURE DEFLECTION RECORDINGS, STRAIN MEASUREMENTS AND VISUAL OBSERVATIONS OF THE GEOMETRY. (FAN)

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
BUCKLING * PRESSURE VESSELS * PRESSURE, INTERNAL * TESTING * STEEL * STEEL, STAINLESS * ALUMINUM * FRANCE

146735

RIEDEL B

EXPERIENCE SUPER-CANON (IN FRENCH)

CEA SERVICE DES TRANSFERTS THERMIQUES, FRANCE

IT/SCHE/79-2-07/R * FRUSR-182 * APPROX. 90 PPS, FIGS, FEB. 6, 1979

DESCRIBES THE SUPER-CANON BLOWDOWN EXPERIMENTAL FACILITY AND INSTRUMENTATION FOR MEASURING PRESSURE, TEMPERATURE, VOID FRACTION, AND THRUST. RESULTS OF THE EXPERIMENTS ARE PRESENTED FOR A NUMBER OF TEMPERATURES, PRESSURES, AND BREAK SIZES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
FRANCE * EXPERIMENT * SYSTEM DESCRIPTION * ACCIDENT, LOSS OF COOLANT * BLOWDOWN * DATA COLLECTION

146669

VOIN R

THE PRACTICE OF QUALITY ASSURANCE BY FRAMATOME (IN ENGLISH)

FRAMATOME, FRANCE

FRUSR-180 * 7 PPS, PRESENTED AT EUROPEAN NUCLEAR CONFERENCE; HAMBURG, MAY 6-11, 1979

FRAMATOME HAS MORE THAN TWENTY YEARS OF EXPERIENCE IN THE ENGINEERING, MANUFACTURING, TESTING AND COMMISSIONING OF NSSS OF THE PWR TECHNOLOGY. THIS PAPER DESCRIBES THE ORGANIZATION WHICH HAS BEEN IMPLEMENTED DURING THIS TIME TO PROVIDE THE QUALITY ASSURANCE OF FRAMATOME'S PRODUCT LINE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
FRANCE * INDUSTRY, NUCLEAR * QUALITY ASSURANCE * REVIEW * DESIGN CRITERIA * FABRICATION * INSTALLATION * CONSTRUCTION * POWER PLANT, NUCLEAR

147175

NAMY D

SELECTION OF EVENTS FOR A PROBABILISTIC EVALUATION OF PWR SAFETY (IN ENGLISH)

FRAMATOME, FRANCE

FRUSR-181 * 8 PPS, FROM HAMBURG CONFERENCE; MAY 6-9, 1979

THIS PAPER PRESENTS A METHOD WHICH CAN BE USEFULLY FOLLOWED TO SELECT INITIATING EVENTS TO BE RETAINED FOR A RISK ANALYSIS OF A NUCLEAR POWER PLANT. THE MAIN STEPS ARE THE FOLLOWING: 1. DETERMINATION AND JUSTIFICATION OF THE INITIATING EVENTS CHOSEN. 2. QUANTIFICATION AND RELIABILITY ANALYSIS OF ACCIDENT SEQUENCES INDUCED BY THE INITIATING EVENTS. 3. RADIOLOGICAL ANALYSIS OF THESE ACCIDENT SEQUENCES. THIS PAPER PRESENTS A GENERAL METHOD OF SELECTION WHICH HAS BEEN USED IN THE LICENSING PROCESS OF KOBBERG NUCLEAR POWER PLANT TO ANSWER THE FIRST STEP OF THE RISK ANALYSIS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
FRANCE * RISK * ANALYTICAL TECHNIQUE * LICENSING PROCESS * REACTOR, PWR * ACCIDENT, PROBABILITY OF * PROBABILITY

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

THE FOLLOWING IS A LISTING OF REPORTS RECEIVED FROM THE FEDERAL REPUBLIC OF GERMANY DURING THE FIRST HALF OF 1979 UNDER THE TECHNICAL EXCHANGE AGREEMENT.

148297
EXPERIMENTAL DETERMINATION OF THE HEAT TRANSFER COEFFICIENT IN THE CONTAINMENT DURING A COOLING SYSTEM BLOWDOWN (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R.G. GERMANY
BF-RS-50-62-4 + GERRSR-356 +. 31 PPS, 3 TABS, 9 FIGS, JUNE 1976

IN THE RESEARCH PROJECT RS 50 SUPPORTED BY THE WEST GERMAN MINISTRY OF RESEARCH AND TECHNOLOGY, RUPTURE OF A MAIN COOLANT PIPE OF A LIGHT-WATER REACTOR IS INVESTIGATED USING A MODEL CONTAINMENT DIVIDED INTO SEVERAL COMPARTMENTS. THE EXPERIMENTAL SET-UP AND THE PROCEDURE OF EVALUATION OF THE MEASURED RESULTS ARE BRIEFLY DESCRIBED. IT WAS FOUND THAT THE VALUES DETERMINED EXPERIMENTALLY IN THE COMPARTMENT IN WHICH RUPTURE TAKES PLACE ARE SUBSTANTIALLY HIGHER AND THOSE IN THE COMPARTMENT REMOTE FROM THE SITE OF RUPTURE ARE MARKEDLY LOWER THAN THE VALUES AFTER TAGAMI/UCHIDA.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
REACTOR, LWR + ACCIDENT, LOSS OF COOLANT + MODEL TESTING + HEAT TRANSFER + TEMPERATURE + HEAT TRANSFER EXPERIMENT + HEAT TRANSFER COEFFICIENT

143329
INVESTIGATION OF THE PROCESSES IN A MULTIPLE COMPARTMENT CONTAINMENT BY PRESSURE IN WATER-COOLED REACTORS WITH REFRIGERATED CONDENSER (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R.G. GERMANY
BF RS 50-32-C19-1 + GERRSR-311 +. APPROX. 200 PPS, FIGS, JULY 1976

NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
GERMANY + *CONTAINMENT ANALYSIS + *COMPARTMENT + CONTAINMENT, ICE CONDENSER + PRESSURE, INTERNAL + REACTOR, LWR

145875
INVESTIGATION OF THE PRESSURE TRANSIENT IN A MULTICOMPARTMENTED CONTAINMENT FROM THE COOLANT BLOWDOWN OF A WATER-COOLED REACTOR - INTERIM RESEARCH REPORT C 13 (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R.G. GERMANY
BF-RS 50-32-C13-1 + GERRSR-359 +. APPROX. 200 PPS, FIGS, JULY 1976

THERE WAS NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
*PRESSURE TRANSIENT + *CONTAINMENT + *COMPARTMENT + BLOWDOWN + REACTOR, LWR + GERMANY

145872
JAX P
PRELIMINARY EXPERIMENT ON LEAKAGE MONITORING USING SONIC EMISSION ANALYSIS: EXPANDED INSTRUMENTATION AND EVALUATION PROGRAM (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R.G. GERMANY
BF-R-62,944-2 + RS 193 + GERRSR-355 +. 76 PPS, 6 TABLES, 30 FIGS, 8 REFS, MAY 1977

THERE WAS NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
*EXPERIMENT + *LEAK DETECTION + *MONITOR + *ACOUSTICS + CONTAINMENT LEAK MONITOR + GERMANY

144196
VON KLOT R + SAHM A + EISENBLATTER J + JUST H
PROPAGATION OF SIMULATED SONIC EMISSION-IMPULSES IN THICK WALLED STRUCTURES (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R.G. GERMANY
BF-R-62,945-1 + GERRSR-313 +. 101 PPS, 41 FIGS, DEC. 1977

NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED. (GTM)

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
GERMANY + *ACOUSTICS + SIMULATION + NOISE ANALYSIS + *PRESSURE VESSELS

145846
INVESTIGATION OF THE PHENOMEN OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS, QUICK LOOK REPORT EXPERIMENT D15 (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R.G. GERMANY
BF-RS 50-30-D15 + GERRSR-343 +. APPROX. 200 PPS, FIGS, MARCH 1978

THIS REPORT, WRITTEN IN GERMAN, IS THE QUICK LOOK REPORT FOR EXPERIMENT D15. THIS REPORT CONTAINS DIAGRAMS OF THE EXPERIMENTAL APPARATUS AND SEVERAL PLOTS OF DATA TAKEN DURING THE TEST.

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

145846 *CONTINUED*
SAFETY ANALYSIS + CONTAINMENT + CONTAINMENT INSTRUMENTATION + MEASUREMENT, TEMPERATURE + INSTRUMENT, PRESSURE

145847
INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUITS IN WATER-COOLED REACTORS, SUPPLEMENTAL RESEARCH DOCUMENTATION D15 (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R.G. GERMANY
DF-RS 50-32-D15 + GERRSR-341 +. APPROX. 100 PPS, FIGS, APRIL 1978

THIS REPORT, WRITTEN IN GERMAN, IS THE TECHNICAL REPORT FOR EXPERIMENT D15. THIS REPORT CONTAINS DIAGRAMS OF THE EXPERIMENT APPARATUS, AND SEVERAL PLOTS OF DATA TAKEN DURING THE EXPERIMENT. BRIEF DISCUSSIONS OF THE DATA ARE GIVEN.

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

SAFETY ANALYSIS + CONTAINMENT INSTRUMENTATION + INSTRUMENT, TEMPERATURE + MEASUREMENT, TEMPERATURE + INSTRUMENT, PRESSURE + CONTAINMENT

144198
JAX P + LORENZ H + OCHS J
IMPROVEMENT IN THE MEASUREMENT TECHNIQUES OF SONIC-EMISSION ANALYSIS (SEA) (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R.G. GERMANY
DF-R-63,244-1 + GERRSR-314 +. 54 PPS, 4 TABS, 9 FIGS, 8 REFS, MAY 1978

NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

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

GERMANY + ACOUSTICS + NOISE ANALYSIS + MEASUREMENT + TECHNOLOGY

145272
SCHALL M
COMPREHENSIVE SUMMARY OF THE THEORETICAL STUDIES ON THE D-SERIES OF THE RESEARCH PROGRAM RS 50 (MODEL CONTAINMENT) PART 1 (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R.G. GERMANY
DF-RS 50A-1 + GERRSR-357 +. APPROX. 270 PPS, FIGS, SEPT. 1978

IN RESEARCH PROJECT RS 50, "PRESSURE DISTRIBUTION IN A REACTOR CONTAINMENT AFTER A LOSS OF COOLANT ACCIDENT", INTEGRAL BLOWDOWN EXPERIMENTS ARE PERFORMED IN A MODEL CONTAINMENT (V EQUAL 600 M³ CUBED). THE EXPERIMENTAL RESULTS ARE TO BE USED FOR VERIFICATION AND IMPROVEMENT OF CONTAINMENT ANALYSIS CODES. FOR THE EXPERIMENTS OF THE D-SERIES, WHICH WERE PERFORMED UNDER SIMPLIFIED CONDITIONS (VAPOR FLOW IN THE SHORT TERM PERIOD, CHAIN-TYPE ARRANGEMENT OF THE COMPARTMENTS), A POST TEST ANALYSIS WAS PERFORMED. IN THIS FINAL REPORT A COMPREHENSIVE SUMMARY OF THIS WORK AND OF ADDITIONAL STUDIES IS GIVEN.

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

REACTOR, PWR + ACCIDENT, LOSS OF COOLANT + CONTAINMENT + COMPUTER PROGRAM + CONTAINMENT ANALYSIS + CONTAINMENT, LOW PRESSURE

145798
SCHALL M
ANALYSIS OF THE D SERIES EXPERIMENTS OF RESEARCH PROJECT RS 50 (MODEL CONTAINMENT) PART 2 (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R.G. GERMANY
DF-RS 50A-1 + RS 50A + GERRSR-358 +. 283 PPS, FIGS, SEPT. 1978

IN PART 2 OF THE FINAL REPORT ON THE RESEARCH PROJECT RS 50 A, "ANALYSIS OF THE D-SERIES EXPERIMENTS OF RESEARCH PROJECT RS 50 (MODEL CONTAINMENT)" THE PLOTS DISCUSSED IN PART 1 ARE PRESENTED CONTAINING THE RESULTS OF THE MODEL CALCULATIONS AND EXPERIMENTAL RESULTS.

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

*DATA PROCESSING + CONTAINMENT ANALYSIS + EXPERIMENT + R AND D PROGRAM + GERMANY + *CONTAINMENT R AND D

144582
EXPERIMENTAL INVESTIGATION OF THE HYDROGEN DISTRIBUTION IN A LIGHT WATER REACTOR CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT, QUICK LOOK REPORT 1 (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R.G. GERMANY
DF-RS 246-1 + GERRSR-312 +. 54 PPS, 3 TABS, 28 FIGS, OCT. 1978

THE OBJECTIVE OF THE "HYDROGEN DISPERSION IN THE CONTAINMENT" PROJECT IS TO STUDY BY MEANS OF EXPERIMENTS THE CONVECTION AND DIFFUSION PROCESSES BY WHICH HYDROGEN IS DISPERSED IN AIR. FOR THIS THE MODEL CONTAINMENT AVAILABLE AT BATTELLE-INSTITUT IS USED, WHICH IS ALSO USED FOR THE EXPERIMENTS OF THE RS 50 PROGRAMME.

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

DIFFUSION + ANALYTICAL MODEL + HYDROGEN + GERMANY

143806
 THE CONTAINMENT TEST FACILITY (EXPERIMENTS C AND D) (IN GERMAN)
 BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
 DF-RS-50-21-1 + GERRSR-323 +. 130 PPS, FIGS, REFS, OCT, 1978

TO VERIFY AND IMPROVE CONTAINMENT COMPUTER CODES, LOSS-OF-COOLANT ACCIDENTS ARE CARRIED OUT IN A MODEL CONTAINMENT (VOLUME 600 M³ CUBED) AT BATTELLE-FRANKFURT. THE PRESENT REPORT DESCRIBES IN DETAIL THE MECHANICAL COMPONENTS OF THE CONTAINMENT TEST FACILITY AND GIVES A BRIEF SURVEY OF ITS MEASURING SYSTEMS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101
 GERMANY + COMPUTER PROGRAM + ACCIDENT, LOSS OF COOLANT + CONTAINMENT + *CONTAINMENT ANALYSIS + EXPERIMENT + C AND D PROGRAM

144286
 EXPERIMENTAL INVESTIGATION OF THE HYDROGEN DISTRIBUTION IN A LIGHT WATER REACTOR CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT, QUICK LOOK REPORT 2 (IN GERMAN)
 BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
 DF-RS 246-2 + GERRSR-331 +. 33 PPS, 19 FIGS, DEC, 1978

THE PURPOSE AND GOAL OF THE PROJECT IS TO STUDY EXPERIMENTALLY THE DISTRIBUTION PROCESSES OF HYDROGEN IN AIR AS A RESULT OF CONVECTION AND DIFFUSION. IF THE GAS INJECTION SOURCE IS NOT AT FLOOR LEVEL, A DISTINCT VERTICAL CONCENTRATION GRADIENT CAN BE OBSERVED IN THE COMPARTMENT WHERE THE SOURCE IS LOCATED (9). A HORIZONTAL CONCENTRATION GRADIENT BETWEEN THE EXPERIMENTAL COMPARTMENTS OCCURS ONLY IF THE CONNECTING OPENING IS RELATIVELY SMALL.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101
 DIFFUSION + ANALYTICAL MODEL + GERMANY + HYDROGEN

144273
 INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS, CONDENSATION IN CONTAINMENT BY EXPERIMENTS C04 AND C1 TO C16 (IN GERMAN)
 BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
 DF-RS 50-31-5 + GERRSR-387 +. APPROX. 400 PPS, FIGS, FEB, 1979

TO VERIFY AND IMPROVE CONTAINMENT COMPUTER CODES LOSS-OF-COOLANT-ACCIDENT EXPERIMENTS WERE CARRIED OUT WITH A MODEL CONTAINMENT. FIRST EVALUATIONS OF THE EXPERIMENTAL RESULTS HAVE SHOWN THAT ALL DETAILS OF THE HEAT TRANSFER PROCESSES (MAINLY CONDENSATION) BETWEEN CONTAINMENT ATMOSPHERE AND CONTAINMENT STRUCTURES HAVE TO BE TAKEN INTO ACCOUNT IN THE MODEL CALCULATIONS. THE PRESENT REPORT CONTAINS A LIST OF THE INTERNAL SURFACES OF THE MODEL CONTAINMENT FOR EXPERIMENTS C04, C1 TO C16.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101
 REACTOR + REACTOR, SAFETY RESEARCH + CONTAINMENT + COMPUTER PROGRAM + HEAT TRANSFER + CONTAINMENT STRUCTURE + CONTAINMENT ANALYSIS

145874
 DRESCHER HP + HODDER P
 COMPARATIVE INVESTIGATIONS OF A COOLING SYSTEM BLOWDOWN ACCIDENT AND THE SUBSEQUENT THERMAL TRANSIENT IN LIGHT-WATER AND HIGH TEMPERATURE REACTORS (IN GERMAN)
 HONNENBERG + DRESCHER INGENIEURGESSELLSCHAFT MBH, F.R. GERMANY
 GERRSR-361 +. 138 PPS, 23 FIGS, 99 REFS, JULY 1975

THERE WAS NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101
 ACCIDENT, LOSS OF COOLANT + ACCIDENT ANALYSIS + *BLOWDOWN + *THERMAL TRANSIENT + REACTOR, LWR + HIGH TEMPERATURE + REACTOR + COMPARISON + GERMANY

144280
 BEHRENS K + SCHNEIDER H
 INITIATION OF DETONATION OF HYDROGEN-AIR MIXTURES AND PROPAGATION OF SHOCK WAVES IN THE ENVIRONMENT (IN GERMAN)
 ERNST-MACH-INSTITUT, F.R. GERMANY
 RS-102-06-6 + GERRSR-342 +. 37 PPS, 10 TABS, 11 FIGS, 8 REFS, DEC, 1977

THERE WAS NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22101
 HYDROGEN + AIR + *EXPLOSION + *SHOCK WAVE + ENVIRONMENT + GERMANY

143803

14393 *CONTINUED*
 LANGHEIM H
 BEHAVIOR OF SPECIFIC REACTOR MATERIALS AND COMPONENT PARTS AT IMPACT OF FRAGMENTS AND PROJECTILES OF DIFFERENT MASS AND VELOCITY (IN GERMAN)
 ERNST-MACH-INSTITUT, F.R. GERMANY
 EMI F75778 + RS 102-07-9 + GERRSR-329 +, 28 PPS, 4 TABS, 21 FIGS, 8 REFS, MAY 1978

THERE WAS NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
 *COMPONENTS + STRUCTURAL INTEGRITY + *MISSILE GENERATION AND PROTECTION + GERMANY + *IMPACT PROPERTY

14648
 LOTTERMOSER J + KAPLANKIES E + ZENNER P
 LABORATORY INVESTIGATIONS FOR THE ATTAINMENT OF INTERPRETATIONAL MODELS FOR THE ESTIMATION OF FLOW USING ULTRASONIC TESTS ON NUCLEAR REACTORS (IN GERMAN)
 FRAUNHOFER-GESELLSCHAFT, F.R. GERMANY
 REPORT 750236-TW + RS 196 + GERRSR-365 +, 215 PPS, FIGS, REFS, JUNE 29, 1978

THERE WAS NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THE DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
 *EXPERIMENT + ANALYTICAL MODEL + *FLOW + MEASUREMENT + *ULTRASONICS + GERMANY

146799
 JAKOBS E + DEUSTER G
 EXAMINATION OF 3 HSST PLATES RUPTURED IN AIR (IN GERMAN)
 FRAUNHOFER-GESELLSCHAFT, F.R. GERMANY
 REPORT 78-730-TW + RS 247 + GERRSR-339 +, 135 PPS, 28 TABS, 24 FIGS, JULY 31, 1978

***THERE WAS NO ENGLISH ABSTRACT

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
 EXPERIMENT + *STRUCTURAL INTEGRITY + *STEEL + EXAMINATION + GERMANY

149793
 DOBMAN G
 MAGNETIC FLUX METHOD, FINAL REPORT (IN GERMAN)
 FRAUNHOFER-GESELLSCHAFT, F.R. GERMANY
 REPORT 780333-TW + GERRSR-362 K+, 55 PPS, 26 FIGS, 7 REFS, 1978

THERE WAS NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
 *ANALYTICAL TECHNIQUE + MATHEMATICAL TREATMENT + GERMANY

149370
 WALTER F + MULLER W
 ERROR ANALYSIS OF THE AMPLITUDE CURVE, FINAL REPORT (IN GERMAN)
 FRAUNHOFER-GESELLSCHAFT, F.R. GERMANY
 REPORT 780852-TW + RS 102-17 + GERRSR-384 +, 116 PPS, 92 FIGS, 17 REFS (NO DATE)

THERE WAS NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
 *ERROR ANALYSIS + ANALYTICAL TECHNIQUE + GERMANY

149368
 BARBIAN GA + GROHS B
 DISTURBANCE AND ERROR RECONSTRUCTION WITH HELP OF TIME-OF-FLIGHT DATA, FINAL REPORT (IN GERMAN)
 FRAUNHOFER-GESELLSCHAFT, F.R. GERMANY
 REPORT 780404-TW + RS-102-17 + GERRSR-380 +, 36 PPS, 20 FIGS, 10 REFS, JAN. 9, 1979

THERE WAS NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
 *ERROR ANALYSIS + *ANALYTICAL TECHNIQUE + GERMANY + DATA PROCESSING

146796
 INSTRUMENTATION SYSTEM FOR THE DAS-MULTIPLE TUBE RESEARCH PROGRAM (IN GERMAN)
 GESELLSCHAFT FUR KERNENERGIEVERWERTUNG, F.R. GERMANY
 REPORT 73 03 AR B 57 + GERRSR-353 +, 72 PPS, 17 FIGS, NOV. 23, 1978

146796 *CONTINUED*

THERE WAS NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
R AND D PROGRAM + FUEL ROD + INSTRUMENT, FUEL SCANNING + SYSTEM DESCRIPTION + GERMANY

146488

EXPERIMENTAL RESEARCH ON SINGLE AND MULTIPLE TUBE ARRAYS IN THE PRESSURE TRANSIENT OF A NUCLEAR POWER PLANT IN THE LARGE EXPERIMENT AREA OF THE GASS (IN GERMAN)
GESELLSCHAFT FÜR KERNENERGIEVERWERTUNG, F.R.G. GERMANY
REPORT 73 03 AR B 59 + GERRSR-354 +, 32 PPS, 9 TABS, 13 FIGS, 4 REFS, DEC. 13, 1970

A DESCRIPTION OF THE INSTRUMENTATION CONCEPTS FOR THE UNDERSTANDING OF THE PRESSURE TRANSIENT PHENOMENA IS PROVIDED. ***NO ADDITIONAL TRANSLATION WAS AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.***

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
FUEL ROD + FUEL ELEMENTS + PRESSURE TRANSIENT + EXPERIMENT + MEASUREMENT + INSTRUMENT, PRESSURE + GERMANY

143902

VOJTEK I

EVALUATION OF THE 25-ROD BUNDLE TEST (RS-37C) WITH THE CALCULATIONAL PROGRAM (IN GERMAN)
GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R.G. GERMANY
GRS-A-208 + GERRSR-333 +, 273 PPS, 126 FIGS, SEPT. 1978

THE PURPOSE OF THESE EXPERIMENTS WAS TO INVESTIGATE THE TRANSIENT CRITICAL HEAT-FLUX (CHF) PHENOMENA AND POST-CHF FILM BOILING HEAT TRANSFER COEFFICIENTS (HTC) DURING DEPRESSURIZATION. IN THE FRAME OF GERMAN BNFT RESEARCH PROGRAM ON REACTOR SAFETY (RS 263) THE POST-EXPERIMENTAL ANALYSIS HAS BEEN CARRIED OUT BY GRS. THE OBTAINED VALUES OF HTC HAVE BEEN COMPARED TO SEVERAL POST-CHF HEAT TRANSFER CORRELATIONS. THE GOOD AGREEMENT IN THE ENTIRE RANGE OF TEST PARAMETERS WAS OBTAINED ONLY WHEN CONDIE-BENGTSEN IV CORRELATION WAS USED FOR THE CALCULATION OF HTC. THE RESULTS OF POST-EXPERIMENTAL ANALYSIS HAS SHOWN THAT NONE OF THE EMPLOYED CHF-CORRELATIONS PREDICTED CHF WITH SUFFICIENT ACCURACY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
GERMANY + HEAT TRANSFER EXPERIMENT + TRANSIENT + HEAT FLUX, CRITICAL + CORRELATION + HEAT TRANSFER COEFFICIENT + FUEL ROD + THERMAL TRANSIENT + FILM BOILING + R AND D PROGRAM + HEAT TRANSFER, BOILING

143966

SCHMIDT A

REPORT ON THE CONVERSION OF THE LASL-CODE TRAC-PI (VERSION 16.3) TO IBM STANDARD OPERATING SYSTEM MVS (WITH FORTRAN-H-EXTENDED COMPILER) (IN ENGLISH)
GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R.G. GERMANY
GRS-A-206 + GERRSR-335 +, 137 PPS, 20 FIGS, SEPT. 1978

DESCRIBES THE CONVERSION OF TRAC PI VERSION 16.3 FOR USE ON GRS'S AMDAHL 470 V/S COMPUTER EQUIPMENT (IBM-COMPATIBLE) WITH IBM OPERATING SYSTEM MVS. THE WORK THAT HAS TO BE DONE WITH THE FORTRAN SOURCE MAY BE SPLIT INTO TWO PARTS: CONVERSION OF SINGLE PRECISION FLOATING POINT CALCULATION (CD=48 BIT MANTISSA) TO DOUBLE PRECISION (IBM-56 BIT MANTISSA); REMOVAL OR REPLACEMENT OF NON-STANDARD FORTRAN FEATURES, WHICH CAN BE APPLIED AT LASL, SINCE THEY ARE RUNNING A SPECIAL NON-CD OPERATING SYSTEM FROM LIVERMORE WITH EXTENDED FORTRAN FACILITIES.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
GERMANY + COMPUTER PROGRAM + REACTOR PHYSICS + REACTOR TRANSIENT + ACCUMULATORS

143903

WANBA AB

REFLUX-GRS ANALYSIS OF THE REFLLOOD EXPERIMENTS (RS 62) (IN ENGLISH)
GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R.G. GERMANY
GRS-A-199 + GERRSR-336 +, 46 PPS, 20 FIGS, 29 REFS, SEPT. 1978

A METHOD IS PRESENTED FOR THE DETERMINATION OF THE LOCAL SURFACE HEAT FLUX BEHAVIOUR FROM THE MEASURED VARIATION IN THE WALL TEMPERATURE DURING FLOODING. USING THE HISTORY OF SURFACE HEAT FLUX AT DIFFERENT AXIAL POSITIONS, THE PROPAGATION OF THE QUENCH FRONTS IS DETERMINED. THE DEPENDENCE OF SURFACE HEAT FLUX ON SURFACE TEMPERATURE IS USED TO PROVIDE INFORMATION ON THE HEAT TRANSFER REGIMES PRESENT. IN ORDER TO PREDICT THE TEMPERATURE HISTORY OF THE INTERNAL SURFACE OF THE TUBE DURING FLOODING, REWETTING AND HEAT CONDUCTION MODELS FROM THE GRS REFILL AND FLOODING PROGRAM FLUT WERE IMPLEMENTED IN THE MIT-PROGRAM REFLUX.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
GERMANY + HEAT TRANSFER ANALYSIS + THERMAL HYDRAULIC ANALYSIS + WETTING + CORE REFLLOODING + HEAT TRANSFER EXPERIMENT + R AND D PROGRAM + HYDRAULIC EXPERIMENT + EMERGENCY COOLING

143894

143094 *CONTINUED*
 POITNER * RINGER F
 BLOWDOWN - EXPERIMENT RS 109 (LOFT) CONTROL OF THE ELECTRICAL HEATING POWER (IN GERMAN)
 GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R., GERMANY
 GRS-A-209 + GERRSR-337 +, 23 PPS, 15 FIGS, SEPT, 1978

IN THE LOFT BLOWDOWN TESTS THE HEATING POWER OF THE TEST-BUNDLE IS TO BE CONTROLLED IN ORDER TO SIMULATE THE BEHAVIOR OF A FUEL BUNDLE DURING BLOWDOWN. WITHIN DMFT-CONTRACT RS 109 THE GRS IS RESPONSIBLE FOR ESTIMATING THE POWER VERSUS TIME CURVES. THIS REPORT DESCRIBES THE CALCULATIONAL METHOD FOR A DOUBLE-ENDED BREAK BETWEEN PUMP AND PRESSURE VESSEL. IN THIS CASE SIMULATION OF A FUEL-BUNDLE CAN BE ACHIEVED WITH THE FOLLOWING TIME DEPENDENCE OF POWER: HEATING POWER IS KEPT CONSTANT AT STEADY STATE VALUE (100%) TILL DNB IS DETECTED, AND THEN REDUCED TO 8%. DNB CAN BE ASSUMED WHEN THE WALL TEMPERATURE OF THE TEST RODS EXCEEDS 400C.

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

*REACTOR, PWR + *BLOWDOWN + *FUEL ROD + ELECTRIC POWER + FEATERS + *DNB + *NUCLEATE BOILING

143905
 ULLICH R
 RELAP-R/GRS ANALYSIS OF THE NONNUCLEAR LOFT-TESTS L1-4 (PRE AND POST TEST CALCULATIONS) (IN GERMAN)
 GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R., GERMANY
 GRS-A-212 + GERRSR-338 +, APPROX. 125 PPS, FIGS, REFS, SEPT, 1978

LOFT L 1-4 WAS AN ISOTHERMAL BLOWDOWN TEST SIMULATING A 20% COLD LEG BREAK WITH ECC INJECTION INTO THE INTACT LOOP COLD LEG 73, 247. THE RESULTS OF BOTH THE RELAP-R/GRS-PRETEST AND POSTEST ANALYSIS CAN BE SUMMARIZED AS FOLLOWS: 1. BOTH CALCULATIONS SHOWED FAIRLY GOOD RESULTS FOR THE BLOWDOWN PHASE UNTIL THE START OF ECC INJECTION. 2. THE PRETEST SYSTEM SIMULATION FAILED AFTER ECC INJECTION BECAUSE OF WATER PACKING PROBLEMS. REFILL AND REFLOOD OF THE RPV WERE CALCULATED SEPARATELY IN A TWO ZONE REPRESENTATION. 3. THE POSTEST ANALYSIS WAS DONE BY AN IMPROVED VERSION OF RELAP-4/GRS, WHICH AVOIDED STABILITY PROBLEMS. THE CALCULATION WAS DONE IN ONE RUN UP TO ABOUT 47 SECONDS OF PROBLEM TIME.

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

*GERMANY + COMPUTER PROGRAM + ACCIDENT, LOSS OF FLOW + *BLOWDOWN + THERMAL HYDRAULIC ANALYSIS + EMERGENCY COOLING + CORE REFLOODING + HYDRAULIC EXPERIMENT + *LOFT (S-RR)

143379
 BRACHT X
 THE COURSE OF EVENTS IN THE CONCRETE - FAILURE PHASE OF THE HYPOTHETICAL CORE MELTDOWN ACCIDENT: CALCULATION TO IDENTIFY THE INFLUENCE OF VARIOUS PARAMETERS (IN GERMAN)
 GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R., GERMANY
 GRS-A-221 + GERRSR-315 +, 65 PPS, FIGS, OCT, 1978

NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

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

GERMANY + *ACCIDENT ANALYSIS + *CORE MELTDOWN + *CONCRETE + STRUCTURAL INTEGRITY + CONTAINMENT INTEGRITY + MATHEMATICAL TREATMENT

147102
 WÄRNEMUNDE R + MAY H
 THE ESSENTIAL SAFETY ASPECTS OF A CONFINED NUCLEAR FUEL CYCLE (IN GERMAN)
 GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R., GERMANY
 GRS-S-23 + GERRSR-371 +, 60 PPS, 8 TABS, 31 FIGS, NOV, 1978

***THERE WAS NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

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

GERMANY + *FUEL CYCLE + *SAFETY ANALYSIS + SAFEGUARDS, NUCLEAR MATERIAL

145757
 PANA P + SCHWINGES B
 COMPUTER MODEL FOR THE TWO-DIMENSIONAL CALCULATION OF THE WATER POOL-SWELL IN THE CONDENSATION CHAMBER OF A REACTOR SYSTEM (IN GERMAN)
 GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R., GERMANY
 GRS-A-237 + GERRSR-350 +, 23 PPS, 9 FIGS, 4 REFS, NOV, 1978

THE MODEL IS BASED ON THE PARTIAL DIFFERENTIAL-EQUATION OF EULER TO DESCRIBE THE UNSTEADY, TWO DIMENSIONAL FLUID MOTION IN THE POOL. THE DIFFERENTIALS ARE CONVERTED INTO FINITE DIFFERENCES. THE AIR REGION ABOVE THE WATER SURFACE AND THE FLUID REGION ARE DIVIDED INTO CELLS OF THE SAME SIZE. THE FINITE DIFFERENCE EQUATIONS FOR EVERY VERTEX FORM A SYSTEM OF LINEAR EQUATIONS, WHICH CAN BE SOLVED WITH THE DETERMINANT THEOREM. DEFINING INITIAL VALUES AND THE DIFFERENT BOUNDARY CONDITIONS THE VELOCITY AND PRESSURE FIELD CAN BE CALCULATED STEP BY STEP.

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

145757 *CONTINUED*
 GERMANY + ACCIDENT, LOSS OF COOLANT + REACTOR, BWR + SWELLING + PRESSURE PULSE + PRESSURE TRANSIENT +
 CONTAINMENT, PRESSURE SUPPRESSION

144854
 LIST OF REPORTS FROM THE REACTOR SAFETY RESEARCH PROGRAMS OF BMFT, USNRC, EPRI AND JSTA, JULY 1-SEPTEMBER 30,
 1978 (IN GERMAN & ENGLISH)

GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R., GERMANY
 GRS-F-67 + GERRSR-127 +, 65 PPS, DEC, 1978

THIS LIST REVIEWS REPORTS FROM THE FEDERAL REPUBLIC OF GERMANY, FROM THE UNITED STATES AND FROM
 JAPAN CONCERNING SPECIAL PROBLEMS IN THE FIELD OF REACTOR SAFETY RESEARCH. THE LIST PURSUES THE
 FOLLOWING ORDER: COUNTRY OF ORIGIN, PROBLEM AREA CONCERNED, ACCORDING TO THE REACTOR SAFETY
 RESEARCH PROGRAM OF BMFT, REPORTING ORGANIZATION, (GTM)

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

REVIEW + R AND D PROGRAM + UNITED STATES + GERMANY + JAPAN + SAFETY ANALYSIS

145165
 REPORT ON THE RESEARCH PROGRAM SPONSORED BY BMFT IN THE AREA OF REACTOR SAFETY, JULY 1-SEPTEMBER 30, 1978 (IN
 GERMAN)

GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R., GERMANY
 GRS-F-70 + GERRSR-144 +, APPROX. 200 PPS, DEC, 1978

INVESTIGATIONS ON THE SAFETY OF LIGHT WATER REACTORS (LWR) BEING PERFORMED IN THE FRAMEWORK OF
 THIS RESEARCH PROGRAM ON REACTOR SAFETY (RS-PROJECTS) ARE SPONSORED BY THE BMFT (FEDERAL MINISTER
 FOR RESEARCH AND TECHNOLOGY). THE DEJECTIVE 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. BESIDES THE INVESTIGATIONS OF LWR TASKS, PROJECTS ON THE SAFETY OF ADVANCED
 REACTORS SPONSORED BY THE BMFT ARE ALSO PRESENTED.

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

GERMANY + REACTOR, LWR + R AND D PROGRAM + SAFETY ANALYSIS + REACTOR, LMFBR

145536
 HELLINGS G + MANSFELD G
 CO, FLOW - A COMPUTER CODE FOR THE DETERMINATION OF PRESSURE TRANSIENTS IN FULL-PRESSURE CONTAINMENTS OF WATER-
 COOLED NUCLEAR POWER PLANTS (IN GERMAN)
 GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R., GERMANY
 GRS-A-254 + GERRSR-349 +, 145 PPS, FIGS, 12 REFS, DEC, 1978

COFLOW IS A COMPUTER CODE FOR DETERMINATION OF BOTH THE TIME HISTORY AND THE LOCAL DISTRIBUTION OF
 TEMPERATURE AND PRESSURE AFTER A LOSS-OF-COOLANT ACCIDENT IN FULL-PRESSURE CONTAINMENTS OF WATER
 COOLED NUCLEAR POWER REACTORS. THE DYNAMIC PRESSURE OF THE CURRENT IN THE CONTAINMENT CAN BE
 TAKEN INTO CONSIDERATION AS WELL AS THE HEAT TRANSFER TO SOLID STRUCTURES AND HEAT CONDUCTION
 WITHIN THEM. THE PHYSICAL AND MATHEMATICAL BASIS, THE ORGANIZATION AND THE APPLICATION OF THE
 COMPUTER CODE COFLOW ARE DESCRIBED.

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

*CONTAINMENT ANALYSIS + CONTAINMENT + PRESSURE TRANSIENT + COMPUTER PROGRAM + GERMANY + ACCIDENT, LOSS OF
 COOLANT + THERMAL TRANSIENT

148674
 FIRNHABER M
 POST-EXPERIMENT CALCULATION OF THE NON-NUCLEAR LOFT TEST L1-5 (IN GERMAN AND ENGLISH)
 GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R., GERMANY
 GRS-A-252 +, GERRSR-351 +, 120 PPS, 7 TABS, 88 FIGS, DEC, 1978

WITHIN THE FRAMEWORK OF THE AGREEMENT ON THE PROMOTION PROJECT RS 182 UNDER THE SHORT TITLE
 "PARTICIPATION ON THE LOFT PROGRAM OF USNRC", GRS IS ENGAGED IN CALCULATIONS OF THE LOFT
 EXPERIMENTS. THIS REPORT PRESENTS A DOCUMENTATION OF THE LOFT L 1-5 CALCULATIONS CONDUCTED BY
 THE GRS. LOFT L 1-5 WAS AN ISOTHERMAL NONNUCLEAR BLOWDOWN TEST SIMULATING A 200% COLD LEG BREAK
 WITH ECC INJECTION INTO THE INTACT LOOP COLD LEG. INSTEAD OF THE CORE SIMULATOR AN UNPOWERED
 NUCLEAR CORE WAS INSTALLED. RESULTS OF THE ANALYSIS ARE DISCUSSED.

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

*GERMANY + *THERMAL HYDRAULIC ANALYSIS + *LOFT (S-RR) + REACTOR, SAFETY RESEARCH + COMPARISON, THEORY AND
 EXPERIENCE + ACCIDENT, LOSS OF COOLANT + BLOWDOWN + TESTING

148668
 BUHL W + LIESCH KJ
 RESULTS OF THE LOFT EXPERIMENT L1-4: POST RUN CALCULATIONS USING THE COMPUTER CODE "DRUFAN" (IN GERMAN)
 GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R., GERMANY
 GRS-A-243 + GERRSR-382 +, 48 PPS, FIGS, REFS, DEC, 1978

148908 *CONTINUED*

THE TEST L1-4 WAS RECALCULATED APPLYING THE COMPUTER CODE DRUFAN. FOR THE RESULTS OF THESE COMPUTATIONS, COMPREHENSIVE COMPARISON MATERIAL GAINED FROM EXPERIMENTAL WORK WAS AVAILABLE. IT WAS THE AIM OF THESE INVESTIGATIONS TO APPLY DRUFAN TO A COMPLEX SYSTEM SUCH AS THE LUFT FACILITY AND, IF NECESSARY, TO MODIFY THE PROGRAM IN ORDER TO SHOW THE THERMO- AND FLUIDDYNAMIC PHENOMENA IN THE PRIMARY SYSTEM OF A PWR DURING A LOCA.

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

COMPUTER PROGRAM * LUFT (S-RR) * ACCIDENT, LOSS OF COOLANT * REACTOR, PWR * GERMANY * PRESSURE TRANSIENT * THERMAL TRANSIENT * FLOW THEORY AND EXPERIMENTS

148909

GRS ANNUAL PROGRESS REPORT-1978 (IN GERMAN)
GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R., GERMANY
JAHRESBERICHT 1978 * GERRSR-391 * 105 PPS, 1978

THERE WAS NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

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

*GERMANY * R AND D PROGRAM * INDUSTRY, NUCLEAR

148913

LIST OF REPORTS FROM THE REACTOR SAFETY RESEARCH PROGRAMS OF BMFT, USNRC, EPRI, AND JSTA, REPORT PERIOD OCTOBER 1 - DECEMBER 31, 1978 (IN GERMAN & ENGLISH)
GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R., GERMANY
GRS-F-72 * GERRSR-352 * 28 PPS, JAN. 1979

THIS LIST REVIEWS REPORTS FROM THE FEDERAL REPUBLIC OF GERMANY, FROM THE UNITED STATES OF AMERICA AND FROM JAPAN CONCERNING SPECIAL PROBLEMS IN THE FIELD OF REACTOR SAFETY RESEARCH. THE LIST PURSUES THE FOLLOWING ORDER: COUNTRY OF ORIGIN, PROBLEM AREA CONCERNED, ACCORDING TO THE REACTOR SAFETY RESEARCH PROGRAM OF BMFT, REPORTING ORGANIZATION. THE LIST OF REPORTS APPEARS QUARTERLY.

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

*R AND D PROGRAM * SAFETY PROGRAM * SAFETY ANALYSIS * UNITED STATES * JAPAN * GERMANY

148905

PITTS JH
ANALYSIS OF BOILING-WATER REACTOR STEAM CHUGGING (IN ENGLISH)
GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R., GERMANY
GRS-A-259 * GERRSR-369 * 82 PPS, 2 TABS, 15 FIGS, 31 REFS, JAN. 1979

RESULTS OF A TRANSIENT ANALYSIS, WHICH PREDICTS THE GENERAL CHARACTERISTICS OF STEAM CHUGGING, COMPARED WELL WITH TWO LARGE SCALE EXPERIMENTS, GKM II TEST 21 AND GKSS TEST 16. THE ANALYSIS INCLUDES EFFECTS OF AIR IN THE DRYWELL, MOMENTUM LOSS AND HEAT TRANSFER IN THE CONDENSATION PIPE, DIRECT CONTACT CONDENSATION HEAT TRANSFER AT THE GAS-WATER INTERFACE, AND MOMENTUM AND HEAT TRANSFER IN THE WETWELL WATER POOL. BUBBLE SHAPE IS CALCULATED IN TWO-DIMENSIONAL CYLINDRICAL COORDINATES.

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

*REACTOR, BWR * STEAM * HYDRAULIC EFFECT * HYDRAULIC ANALYSIS * HEAT TRANSFER ANALYSIS * COMPARISON, THEORY AND EXPERIENCE * COMPUTER PROGRAM * GERMANY

147176

SAFETY CONTAINMENT OF NUCLEAR POWER PLANTS (IN GERMAN)
GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R., GERMANY
GRS-13 * GERRSR-370 * 75 PPS, TABS, FIGS, JAN. 1979

THERE WAS NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

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

GERMANY * CONTAINMENT * CONTAINMENT ANALYSIS * CONTAINMENT R AND D * POWER PLANT, NUCLEAR * SAFETY ANALYSIS

148672

SCHAEFER A
DEVELOPMENT OF A CALCULATIONAL PROGRAM FOR THE SOLUTION OF THE NEUTRONIC EQUATION OF A MULTIDIMENSIONAL HTGR MODEL (IN GERMAN)
GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) MBH, F.R., GERMANY
GRS-14 * GERRSR-379 * 70 PPS, 2 TABS, 47 FIGS, 5 REFS, FEB. 1979

A NEW CODE FOR EFFICIENT SOLUTION OF THE MULTIDIMENSIONAL STATIONARY MULTI-GROUP-DIFFUSION EQUATION, TO BE USED WITHIN A HTGR-CODE MODEL, IS PRESENTED. THE APPROXIMATION AND ITERATION METHODS ARE DESCRIBED. SPACIAL APPROXIMATION IS BASED ON THE QUADBOX-COARSE-MESH METHOD, BUT ITERATION METHODS ARE DIFFERENT FROM QUADBOX TO GIVE LINEAR DEPENDENCE OF COMPUTATION TIME ON THE NUMBER OF ENERGY GROUPS. RESULTS FOR VARIOUS MULTIDIMENSIONAL MULTI-GROUP PROBLEMS, AMONG THEM

148672 *CONTINUED*
THE HTGR PEBBLE BED REACTOR ARE ANALYZED.

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

*COMPUTER PROGRAM * ANALYTICAL TECHNIQUE * DIFFUSION * EQUATION * REACTOR, HTGR * REACTOR PHYSICS * GERMANY

147483
REPORT OF THE FEDERAL MINISTER FOR RESEARCH AND TECHNOLOGY CONCERNING RESEARCH PROJECTS IN THE AREA OF REACTOR SAFETY REPORTING PERIOD OCTOBER-DECEMBER 31, 1973 (IN GERMAN)
GESELLSCHAFT FÜR REAKTOR SICHERHEIT (GRS) MBH, F.R. GERMANY
GRS-F-74 + GERRSR-372 +, APPROX. 300 PPS, MARCH 1979

INVESTIGATIONS ON THE SAFETY OF LIGHT WATER REACTORS BEING PERFORMED IN THE FRAMEWORK OF THIS RESEARCH PROGRAM ON REACTOR SAFETY ARE SPONSORED BY THE BMFT (FEDERAL MINISTER FOR RESEARCH AND TECHNOLOGY). OBJECTIVES OF THIS PROGRAM ARE TO INVESTIGATE IN GREATER DETAIL THE SAFETY MARGINS OF NUCLEAR ENERGY PLANTS AND THEIR SYSTEMS AND THE FURTHER DEVELOPMENT OF SAFETY TECHNOLOGY. BESIDES THE INVESTIGATIONS OF LWR TASKS, ALSO PROJECTS ON THE SAFETY OF ADVANCED REACTORS ARE SPONSORED BY THE BMFT. EACH PROGRESS REPORT REPRESENTS A COMPILATION OF INDIVIDUAL REPORTS ABOUT OBJECTIVES, THE WORK PERFORMED, THE RESULTS, THE NEXT STEPS OF THE WORK, ETC.

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

*GERMANY * R AND D PROGRAM * REACTOR, LWR * REACTOR, PWR * REACTOR, BWR * REACTOR, LMFBR * SAFETY ANALYSIS * ACCIDENT ANALYSIS

147467
ANNUAL REPORT ON REACTOR SAFETY RESEARCH PROJECTS SPONSORED BY THE MINISTRY FOR RESEARCH AND TECHNOLOGY OF THE FEDERAL REPUBLIC OF GERMANY, 1973 (IN ENGLISH)
GESELLSCHAFT FÜR REAKTOR SICHERHEIT (GRS) MBH, F.R. GERMANY
GRS-F-76 + GERRSR-373 +, APPROX. 400 PPS, TABS, FIGS, MARCH 1979

INVESTIGATIONS ON THE SAFETY OF LIGHT WATER REACTORS (LWR) BEING PERFORMED IN THE FRAMEWORK OF THIS RESEARCH PROGRAM REACTOR SAFETY (RS-PROJECTS) ARE SPONSORED BY THE BMFT. OBJECTIVE OF THIS PROGRAM IS TO INVESTIGATE IN GREATER DETAIL THE SAFETY MARGINS OF NUCLEAR POWER PLANTS AND THEIR SYSTEMS AND THE FURTHER DEVELOPMENT OF SAFETY TECHNOLOGY. BESIDES THE INVESTIGATIONS OF LWR TASKS, ALSO, PROJECTS ON THE SAFETY OF ADVANCED REACTORS ARE SPONSORED BY THE BMFT ARE REPORTED ON. EACH PROGRESS REPORT REPRESENTS A COMPILATION OF INDIVIDUAL REPORTS ABOUT OBJECTIVES, THE WORK PERFORMED, THE RESULTS, THE NEXT STEPS OF THE WORK, ETC.

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

*R AND D PROGRAM * GERMANY * REACTOR, LWR * REACTOR, LMFBR * SAFETY ANALYSIS * ACCIDENT ANALYSIS

148363
MULLER-CHRISTIANSE + WOLLESEN M
ARTICLES ON QUESTIONS PERTAINING TO NUCLEAR ENERGY: PLUTONIUM (IN GERMAN)
GESELLSCHAFT FÜR REAKTOR SICHERHEIT (GRS) MBH, F.R. GERMANY
GRS-S-27 + GERRSR-378 +, 46 PPS, 8 TABS, 10 FIGS, 22 REFS, APRIL 1979

THERE WAS NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

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

GERMANY * PLUTONIUM * N-POWER, SAFETY OF * SOCIO/PHILOSOPHICAL CONSIDERATION

148670
BERNNAT W + DIETL G + HALM G
COMPREHENSIVE DAMAGE ANALYSIS FOR HIGH TEMPERATURE GAS REACTORS, PHASE II, WATER INGRESS, AIR INGRESS, REACTIVITY EXCURSIONS (IN GERMAN)
HOCHTEMPERATUR-REAKTORBAU GMBH, F.R. GERMANY
RS-252 + GERRSR-386 +, 116 PPS, FIGS, REFS, JAN. 1979

THERE WAS NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

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

GERMANY * REACTOR, HTGR * DAMAGE * WATER * AIR * REACTIVITY, EXCESS * EXCURSION, LARGE * ACCIDENT ANALYSIS

144596
MICHAEL I
EXPERIMENTAL INVESTIGATIONS OF THE RADIOACTIVITY IN THE PRIMARY SYSTEM OF PRESSURIZED WATER REACTORS (IN GERMAN)
KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. GERMANY
KFK-2664 + GERRSR-318 +, 79 PPS, 40 FIGS, 48 REFS, SEPT. 1978

THE REPORT DESCRIBES WORK CARRIED OUT WITHIN THE FRAMEWORK OF THE REACTOR SAFETY RESEARCH PROGRAM AND CONCERNED WITH THE ANALYSIS OF RADIATION EXPOSURES CAUSED BY THE OPERATION OF NUCLEAR POWER PLANTS EQUIPPED WITH PRESSURIZED WATER REACTORS, AND WITH PROBLEMS OF THE RELEASE AND TRANSPORT

144996 *CONTINUED*

OF RADIOACTIVE SUBSTANCES IN PRIMARY CIRCUITS. THE EFFORTS ARE CONCENTRATED MAINLY ON THE RESPECTIVE REDUCTION MEASURES.

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

MASS TRANSFER * FISSION PRODUCT TRANSPORT * CORROSION * RADIATION EXPOSURE * ANALYTICAL TECHNIQUE * REACTOR, PWR * GERMANY * STEAM * UXTGEN

143906

ERBACHER F

FUEL ELEMENT BEHAVIOR DURING A LOSS-OF-COOLANT ACCIDENT AND INTERACTION WITH THE EMERGENCY CORE COOLING (IN GERMAN)

KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R.G. GERMANY

KFK-2691 * GERRSR-319 * 52 PPS, 28 FIGS, SEPT, 1978

THE PROCESS OF EMERGENCY CORE COOLING IN A LOCA OF A PRESSURIZED WATER REACTOR IS SUMMARIZED. THE THERMOHYDRAULICS IN THE REACTOR CORE AND THE LOADING OF THE FUEL ROD CLADDINGS DURING A LOCA ARE COVERED IN MORE DETAIL. SOME RECENT EXPERIMENTAL RESULTS ON ZIRCALOY CLADDING DEFORMATION IN A LOCA ARE DISCUSSED. THEY INDICATE THAT AXIAL AND AZIMUTHAL CLADDING TEMPERATURE DIFFERENCES, WHICH ARE ENHANCED BY COOLING DURING REFLOODING, ARE LIMITING THE STRAINS OF THE ZIRCALOY CLADDING TUBES AND THE RESULTING COOLANT CHANNEL BLOCKAGE IN THE FUEL ELEMENTS.

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

*REACTOR, PWR * ACCIDENT, LOSS OF COOLANT * FLOW BLOCKAGE * EMERGENCY COOLING SYSTEM * FLOW BLOCKAGE

143745

SOECK M * CLASS 3 * ERBACHER F * FIECK A

STATUS AND RESULTS OF THE THEORETICAL AND EXPERIMENTAL INVESTIGATIONS ON THE LWR FUEL ROD BEHAVIOR UNDER ACCIDENT CONDITIONS (IN GERMAN)

KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R.G. GERMANY

KFK-EXT 28/78-1 * GERRSR-322 * 143 PPS, FIGS, REFS, SEPT, 1978

PRESENTS INFORMATION ACCUMULATED THROUGH 1977 ON FUEL ROD BEHAVIOR IN LWRs DURING LOSS-OF-COOLANT ACCIDENTS. RESULTS PRESENTED HAVE BEEN DERIVED FROM STUDIES ON THE FUEL ROD BEHAVIOR PERFORMED WITHIN THE FRAMEWORK OF THE NUCLEAR SAFETY PROJECT (PNS). THE RESULTS FROM COOPERATING RESEARCH ESTABLISHMENTS AND FROM INTERNATIONAL EXCHANGE OF EXPERIENCE ARE REFERRED TO ALSO.

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

GERMANY * REACTOR, LWR * FUEL ROD * ACCIDENT, LOSS OF COOLANT * EXPERIMENT * ACCIDENT ANALYSIS

143089

SOECK M

CREEP RUPTURE AT NON-STEADY STRESS AND TEMPERATURE LOADING CONDITIONS (IN ENGLISH)

KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R.G. GERMANY

KFK-2699 * GERRSR-320 * 60 PPS, 17 FIGS, OCT, 1978

ASSUMING THE VALIDITY OF THE LIFE FRACTION RULE (LFR) THE TIME TO RUPTURE AS WELL AS THE RESPECTIVE STRESS AND TEMPERATURE AT FAILURE HAVE BEEN CALCULATED FOR SEVERAL RAMP LOADING CONDITIONS. THE RESULTS OF RAMP RUPTURE TESTS CAN BE PREDICTED SOLELY FROM ISO-STRESS RUPTURE EXPERIMENTS WITHOUT ANY FITTING PROCEDURE. THE CALCULATIONS ARE COMPARED WITH RESULTS FROM TUBE BURST EXPERIMENTS AS WELL AS WITH THOSE FROM TENSILE TESTS ON ZIRCALOY-4. FOR THIS MATERIAL THE LFR IS OBEYED IN THE TEMPERATURE RANGE EXAMINED (873K * 1110K). THE AGREEMENT BETWEEN THE CALCULATIONS AND THE EXPERIMENTAL RESULTS IS SURPRISINGLY GOOD. (FAH)

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

CREEP * STRESS * TEMPERATURE * FAILURE * ZIRCALOY

143211

KEDZIOR F * MOSINGER H

COMPARISON BETWEEN A ONE- AND TWO-DIMENSIONAL CALCULATION OF A WATER-VAPOR NOZZLE FLOW (IN GERMAN)

KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R.G. GERMANY

KFK-2623 * GERRSR-321 * 36 PPS, FIGS, REFS, OCT, 1978

THE STEADY WATER-VAPOR FLOW THROUGH A CONVERGENT NOZZLE IS SIMULATED WITH THE TWO-PHASE COMPUTER CODES DRIX-2D (TWO-DIMENSIONAL, TRANSIENT) AND CUESE (ONE-DIMENSIONAL, STATIONARY). THE RESULTS OF BOTH CODES ARE COMPARED AND INTERPRETED UNDER CONSIDERATION OF THEIR DIFFERENT MODELING, ESPECIALLY WITH RESPECT TO THE DIMENSIONALITY AND THE TIME-BEHAVIOR. THE MAIN RESULT OF THESE COMPARISONS IS THE UNDERSTANDING, THAT IN PRINCIPLE THE TWO-DIMENSIONAL CALCULATION RENDERS A LARGE PRESSURE-DROP OF THE NOZZLE-FLOW THAN THE ONE-DIMENSIONAL ONE. (MLW)

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

GERMANY * NOZZLE * WATER VAPOR * FLOW * COMPARISON * COMPUTER PROGRAM * PRESSURE DROP * ANALYTICAL MODEL

147794

147794 *CONTINUED*
 NUCLEAR SAFETY PROJECT FIRST SEMI-ANNUAL REPORT 1978 (IN GERMAN)
 KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. GERMANY
 KFK-2700 + GERRSR-388 +. APPROX. 400 PPS, FIGS, REFS, NOV. 1978

THE 13TH SEMI-ANNUAL REPORT 1978 IS A DESCRIPTION OF WORK WITHIN THE NUCLEAR SAFETY PROJECT PERFORMED IN THE FIRST SIX MONTHS OF 1978 IN THE NUCLEAR SAFETY FIELD BY KFK INSTITUTES AND DEPARTMENTS AND BY EXTERNAL INSTITUTIONS ON BEHALF OF KFK. THE FOLLOWING PROGRAMS ARE REPORTED ON: DYNAMIC LOADS AND STRAINS OF REACTOR COMPONENTS UNDER ACCIDENT CONDITIONS; FUEL BEHAVIOR UNDER ACCIDENT CONDITIONS; INVESTIGATION AND CONTROL OF LWR CORE-MELTDOWN ACCIDENTS; MODEL DEVELOPMENT FOR ANALYTICAL DESCRIPTION OF CORE-MELTDOWN ACCIDENTS; IMPROVEMENT OF FISSION PRODUCT RETENTION AND REDUCTION OF RADIATION LOAD; OFF GAS CLEANING FOR REPROCESSING PLANTS; AND BEHAVIOR, IMPACT AND REMOVAL OF RELEASED NUCLEAR POLLUTANTS.

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

*GERMANY + *REACTOR, LWR + *R AND D PROGRAM + ACCIDENT ANALYSIS + STRUCTURAL ANALYSIS, DYNAMIC + COMPONENTS + CORE MELTDOWN + ANALYTICAL MODEL + FISSION PRODUCT RETENTION + FUEL REPROCESSING + OFF GAS + RADIOACTIVITY RELEASE + ENVIRONMENT + SOIL, RADIONUCLIDE MOVEMENT THROUGH

146804
 SCHUMANN U
 EFFICIENT COMPUTATION OF THREE-DIMENSIONAL FLUID-STRUCTURE INTERACTIONS DURING BLOWDOWN OF A PRESSURIZED WATER REACTOR-FLUX (IN GERMAN)
 KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. GERMANY
 KFK-2645 + GERRSR-367 +. 250 PPS, TABS, FIGS, 105 REFS, JAN. 1979

THE MODEL USED IN THIS METHOD IS BASED ON THE FOLLOWING ESSENTIAL ASSUMPTIONS: THREE-DIMENSIONAL POTENTIAL FLOW, CONSTANT SPEED OF SOUND, LINEAR-ELASTIC STRUCTURE AND SMALL STRUCTURAL DEFORMATIONS. NOT NEGLECTED ARE THE FLUID-STRUCTURE INTERACTIONS AND THE NON-LINEAR INERTIA FORCES IN THE FLUID. IN THE PRESENT PROGRAM VERSIONS (FOR INCOMPRESSIBLE AND COMPRESSIBLE FLUID) THE DYNAMICAL PROPERTIES OF THE CORE BARREL ARE DESCRIBED BY MEANS OF THE EXISTING SHELL MODEL CYLDZ. THE RELEVANT CONSERVATION EQUATIONS ARE APPROXIMATED BY AN IMPLICIT FINITE DIFFERENCE SCHEME.

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

GERMANY + COMPUTER PROGRAM + CORE COMPONENTS + DEFORMATION + HYDRAULIC EFFECT

146750
 ENDERLE G
 FLUST-2D - A COMPUTERCODE FOR THE CALCULATION OF THE TWO-DIMENSIONAL FLOW OF A COMPRESSIBLE MEDIUM IN COUPLED RECTANGULAR AREAS (IN GERMAN)
 KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. GERMANY
 KFK-2679 + GERRSR-368 +. 188 PPS, 74 FIGS, REFS, JAN. 1979

IN A FINITE DIFFERENCE SCHEME THE PROGRAM COMPUTES PRESSURE, DENSITY, INTERNAL ENERGY AND VELOCITY. STARTING WITH A BASIC SET OF EQUATIONS, THE DIFFERENCE EQUATIONS IN A RECTANGULAR GRID ARE DEVELOPED. THE PROGRAM WAS USED TO PRECALCULATE THE BLOWDOWN EXPERIMENTS OF THE HDR EXPERIMENTAL PROGRAM. DOWNCOMER, PLANA, INTERNAL VESSEL REGION, BLOWDOWN PIPE AND A CONTAINMENT AREA HAVE BEEN MODELLED TWO-DIMENSIONALLY. THE MAJOR RESULTS OF THE PRECALCULATIONS ARE PRESENTED. THIS REPORT ALSO CONTAINS A DESCRIPTION OF THE CODE STRUCTURE AND USER INFORMATION.

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

GERMANY + COMPUTER PROGRAM + FLOW + HYDRODYNAMIC ANALYSIS + FLOW, MIXING

147860
 CALDAROLA L
 FAULT TREE ANALYSIS WITH MULTISTATE COMPONENTS (IN ENGLISH)
 KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. GERMANY
 KFK-2761 + EUR-5750 E + GERRSR-392 +. 49 PPS, FIGS, REFS, FEB. 1979

A GENERAL ANALYTICAL THEORY HAS BEEN DEVELOPED WHICH ALLOWS ONE TO CALCULATE THE OCCURRENCE PROBABILITY OF THE TOP EVENT OF A FAULT TREE WITH MULTISTATE (MORE THAN TWO STATES) COMPONENTS. IT IS SHOWN THAT, IN ORDER TO CORRECTLY DESCRIBE A SYSTEM WITH MULTISTATE COMPONENTS, A SPECIAL TYPE OF BOOLEAN ALGEBRA IS REQUIRED. THE PROBLEM OF STATISTICAL DEPENDENCE AMONG PRIMARY COMPONENTS IS DISCUSSED. THE PAPER INCLUDES A SMALL DEMONSTRATIVE EXAMPLE TO ILLUSTRATE THE METHOD. THE EXAMPLE INCLUDES ALSO STATISTICAL DEPENDENT COMPONENTS. (EWH)

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

*FAULT TREE ANALYSIS + ANALYTICAL TECHNIQUE + MATHEMATICAL TREATMENT

148673
 HOFMANN P
 SIMULATION OF THE CHEMICAL STATE OF IRRADIATED OXIDE FUEL; INFLUENCE OF THE INTERNAL CORROSION ON THE MECHANICAL PROPERTIES OF ZRY-4 TURBINE (IN ENGLISH)
 KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. GERMANY
 KFK-2785 + GERRSR-389 +. 22 PPS, 2 TABS, 10 FIGS, 17 REFS, MARCH 1979

148671 *CONTINUED*

ZIRCALOY IS NOT COMPATIBLE WITH OXIDE FUEL NOR WITH SOME FISSION PRODUCT ELEMENTS. THEREFORE, CHEMICAL INTERACTION BETWEEN THE IRRADIATED OXIDE FUEL AND THE ZIRCALOY CLADDING WILL TAKE PLACE, ESPECIALLY AT TEMPERATURES THAT CAN BE REACHED DURING REACTOR INCIDENTS (ATWS, LOCA). IN ORDER TO FIND OUT WHICH INFLUENCE THE CHEMICAL INTERACTION BETWEEN THE FISSION PRODUCTS AND THE ZIRCALOY CLADDING MATERIAL HAVE ON THE MECHANICAL PROPERTIES OF ZIRCALOY TUBING OUT-OF-PILE BURST EXPERIMENTS AND CREEP RUPTURE TESTS HAVE BEEN PERFORMED AT TEMPERATURES UP TO 600C WITH SHORT TUBE SPECIMENS CONTAINING SIMULATED FISSION PRODUCTS.

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

*ZIRCALOY + *CORROSION + *PROPERTY, MECHANICAL + *FUEL ROD + CLADDING + CHEMICAL REACTION + FUEL, NUCLEAR + OXIDE + OUT OF PILE EXPERIMENT + CREEP + TESTING

148006

ALJIRICK DC + BAYER A + SCHUNCKLER M
A PROPOSED WIND SHIFT MODEL FOR THE GERMAN REACTOR SAFETY STUDY (IN ENGLISH)
KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R.G. GERMANY
KFK-2791 + GERRSR-190 +, 10 PPS, 3 FIGS, 3 REFS, APRIL 1979

NEITHER THE U.S. NOR THE GERMAN REACTOR SAFETY STUDY IN THEIR PRESENT FORM INCLUDE HOURLY CHANGES IN WIND DIRECTION. FOR RELEASES OF SHORT DURATION THIS ASSUMPTION SHOULD HAVE A RELATIVELY SMALL EFFECT ON THE CALCULATION OF ACCIDENT CONSEQUENCES. FOR RELEASES OF LONGER DURATION THIS ASSUMPTION COULD RESULT IN AN OVERESTIMATION OF CENTERLINE RADIOACTIVE CONCENTRATIONS. TO ACCOUNT FOR HOURLY WIND DIRECTION CHANGES, A WIND SHIFT MODEL HAS BEEN PROPOSED. USING HOURLY RECORDED WIND SPEED AND DIRECTION DATA, THE MODEL MODIFIES THE ANGULAR DISTRIBUTION OF RADIOACTIVE CONCENTRATIONS CALCULATED BY A STRAIGHTLINE MODEL, AND IS INTENDED TO BETTER REPRESENT THE CONCENTRATIONS IN AREAS CLOSE TO THE REACTOR WHERE POTENTIAL DOSES MIGHT EXCEED THE THRESHOLD LEVEL FOR EARLY FATALITIES.

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

GERMANY + SAFETY ANALYSIS + *METEOROLOGY + *WIND STATISTICS + *WIND PROFILE + *ACCIDENT, CONSEQUENCES + *ANALYTICAL MODEL + RADIOACTIVITY RELEASE + DOSE

145638

NOJOTNY B + DAUBERSKY P
IMPACT OF STEEL PROJECTILES ON REINFORCED CONCRETE, CALCULATION AND COMPARISON WITH EXPERIMENTAL TESTS (IN GERMAN)
KERNTECHNIK, ENTWICKLUNG, DYNAMIK, F.R.G. GERMANY
RS 226 + GERRSR-348 +, 160 PPS, FIGS, AUG. 1978

NUCLEAR POWER PLANTS HAVE TO BE DESIGNED TO RESIST AN AIRPLANE CRASH WITHOUT ANY DANGER FOR THE ENVIRONMENT. THE INVESTIGATION OF THE EFFECTS OF THE AIRPLANE IMPACT IS A COMPLICATED PROBLEM, WHICH HAS NOT BEEN SOLVED BEFORE WITH EXPERIMENTS AND CALCULATIONS. THE PURPOSE OF THIS RESEARCH PROJECT IS TO EVALUATE A MATHEMATICAL MODEL FOR REINFORCED CONCRETE AND TO CHECK IT AGAINST EXPERIMENTS. USING THE MATHEMATICAL MODEL, EXPERIMENTAL RESULTS SHOULD BE EXTRAPOLATED LATER, ESPECIALLY TO THE EFFECTS OF AN AIRPLANE CRASH ON A NUCLEAR POWER PLANT. (FAH)

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

GERMANY + STEEL + MISSILE GENERATION AND PROTECTION + CONCRETE, REINFORCED + IMPACT SHOCK + ANALYTICAL MODEL + AIRCRAFT + TESTING

148671

SCHAEICKERT A
RESEARCH PROGRAM ON REACTOR SAFETY, 3D EXPERIMENT (IN GERMAN)
KRAFTWERK UNION, ERLANGEN, F.R.G. GERMANY
REPORT RE 23/001/78 + BNFT RS 268 + GERRSR-381 +, 186 PPS, 13 TABS, 55 FIGS, JAN. 1978

A BASIC DESIGN WAS FORMULATED FOR THE "3 D - EXPERIMENT" WHICH IS TO INVESTIGATE THE THERMOHYDRAULIC PHENOMENA IN THE UPPER PLENUM OF A PWR AFTER A LOCA. ONLY THE REFILL AND REFOOD PHASE, BEGINNING AT 5 BAR, WILL BE VERIFIED. A TEST FACILITY WAS DESIGNED AND THE REQUIREMENTS FOR INSTRUMENTATION, DATA ACQUISITION AND TEST EVALUATION WERE DISCUSSED. A BASIC TEST MATRIX WAS PLANNED. MOREOVER TECHNICAL REQUIREMENTS FOR THE "2 D - EXPERIMENT" WERE SUMMARIZED. SIX YEARS ARE REQUIRED FOR PLANNING AND CONSTRUCTING THE TEST FACILITY AND DOING THE TESTS.

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

GERMANY + *R AND D PROGRAM + EXPERIMENT + *THERMAL HYDRAULIC ANALYSIS + REACTOR, PWR + PLENUM + *ACCIDENT, LOSS OF COOLANT + CORE REFOODING

143768

SAUER A
DEVELOPMENT AND SYNTHESIS OF AN EDUCATIONAL SYSTEM USING COMBINATION OF MEDIA FOR THE INTENSIVE TRAINING AND INSTRUCTION OF OPERATING PERSONNEL AT NUCLEAR POWER PLANTS (IN GERMAN)
KRAFTWERK UNION, ERLANGEN, F.R.G. GERMANY
BNFT RS 152 + GERRSR-326 +, APPROX. 240 PPS, FIGS, REFS, SEPT. 1978

A FEASIBLE COMBINATION OF MEDIA WAS WORKED OUT FOR THE OPTIMUM PLANT TRAINING OF THE CONTROL ROOM

14379R *CONTINUED*

PERSONNEL OF NUCLEAR POWER PLANTS AFTER AN EVALUATION OF MEDIA, TAKING INTO ACCOUNT THE PRODUCTION AND REPRODUCTION CRITERIA FOR THE HARDWARE AND SOFTWARE TOGETHER WITH TECHNICAL AND ECONOMIC ASPECTS, A STANDARD METHOD IS RECOMMENDED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA, 22101
GERMANY * POWER PLANT, NUCLEAR * OPERATOR ACTION * LICENSED OPERATOR * TRAINING

14398I

SCHUSTER E * FUCHS A * KARNATH G
ACTIVATED CORROSION PRODUCTS IN SWR LOCPs (IN GERMAN)
KRAFTWERK UNION, ERLANGEN, F.R.G. GERMANY

DMFT-RS-20 * RE 23/057777 * GERNSR-316 *, 58 PPS, 4 TABS, 21 FIGS, 7 REFS, OCT, 1978

ROUTINELY MEASURED ACTIVITY CONCENTRATIONS OF SOME CORROSION PRODUCT RADIONUCLIDES IN THE COOLANT OF DIFFERENT POWER STATIONS WERE EVALUATED. COMPILATIONS APPLIED HAVE DEMONSTRATED THAT THERE ARE SUFFICIENT DATA FOR PWR'S ALLOWING THEIR COMPARISON. THE AVAILABLE DATA FOR BWR'S ARE NOT SUFFICIENT FOR SUCH AN ANALOGUS EVALUATION. THE COMPARISON WAS DONE WITH ACTIVITY CONCENTRATIONS OF 58CO AND 60CO IN THE COOLANT OF FOUR PWR'S OPERATING AT FULL LOAD. FURTHER ON ANALYTICAL METHODS FOR THE DETERMINATION OF THE ELEMENTAL SPECIFIC ACTIVITIES OF 60CO AND 58FE IN SAMPLES FROM THE COOLANT AND FROM DIFFERENT COMPONENTS OF THE PRIMARY CIRCUITS HAVE BEEN OPTIMIZED. (FAH)

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA, 22101
GERMANY * REACTOR, PWR * CORROSION * RADIONUCLIDE * COBALT * IRON * MAIN COOLING SYSTEM

14415B

ENGEL H
DOSE REDUCTION (IN GERMAN)
KRAFTWERK UNION, ERLANGEN, F.R.G. GERMANY

DMFT RS 204 * RE 23/010778 * GERNSR-317 *, 15 PPS, 1 FIG, OCT, 1978

TO IMPROVE THE HYDROGEN/OXYGEN MEASUREMENTS WITHIN THE GASEOUS WASTE PROCESSING SYSTEM AT PWR'S, INVESTIGATIONS WERE PERFORMED TO DETERMINE WHAT PARAMETERS INFLUENCED THE MEASUREMENTS. SUCH PARAMETERS AS GAS HUMIDITY, PRESSURE, FLOW, INFLUENCE OF OXYGEN, HELIUM, AND ARGON CONCENTRATIONS WERE CONSIDERED. (GWM)

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA, 22101
GERMANY * HYDROGEN * OXYGEN * MEASUREMENT * GASEOUS WASTE TREATMENT, GAS * REACTOR, PWR

14515B

KNOEDLER D
PRELIMINARY EMPIRICAL DESCRIPTION OF THE FUEL ROD BEHAVIOUR DURING LOCA (IN GERMAN)
KRAFTWERK UNION, ERLANGEN, F.R.G. GERMANY

RE 23/005778 * DMFT RS 177 * GERNSR-324 *, 65 PPS, 22 FIGS, 12 REFS, OCT, 1978

A MODIFIED NORTON EQUATION IS USED TO DESCRIBE THE STRAIN BEHAVIOUR OF ZIRCALOY TUBES AT TEMPERATURES AS CALCULATED FOR HYPOTHETICAL LOCAs. THE BURST STRAIN AT WHICH THE STRAIN CURVE IS CUT OFF, IS DERIVED EMPIRICALLY AS A FUNCTION OF TEMPERATURE AND HEATING RATE. THE MODELS ARE CALIBRATED AGAINST DATA FROM DIRECTLY HEATED SINGLE ROD EXPERIMENTS, WHICH IN CONTRAST TO REACTOR CONDITIONS EXHIBIT VERY HOMOGENEOUS TEMPERATURES. THIS LEADS TO PARTICULARLY HIGH BURST STRAINS. IT IS SHOWN HOW THESE MODELS CAN BE APPLIED TO CASES WITH AXIUMERIAL AND AXIAL TEMPERATURE VARIATIONS. (FAH)

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA, 22101
GERMANY * ZIRCALOY * FUEL ROD * ACCIDENT, LOSS OF COOLANT * ANALYTICAL MODEL

14515S

DORNER H
INVESTIGATION PROGRAM FOR THE TESTING OF A FRACTURE SAFETY DEVICE PROTECTION SYSTEM FOR REACTOR COMPONENTS (IN GERMAN)

KRAFTWERK UNION, ERLANGEN, F.R.G. GERMANY
RE 23/021/78 DMFT RS 104 GERNSR-325 *, 150 PPS, TABS, FIGS, OCT, 1978

RESULTS OF INVESTIGATIONS ON THE MATERIAL BEHAVIOUR OF INSULATION-CONCRETE SUBJECTED TO TWO-PHASE JET LOADS ARE DESCRIBED. FURTHERMORE THIS REPORT DEALS WITH THE RESULTS OF THE BURST TESTS WITH PIPES WHICH WERE CARRIED OUT UNDER PWR CONDITIONS. TEST EQUIPMENT, INSTRUMENTATION, THE MEASURING TECHNIQUES AND THE TEST PROCEDURE ARE DESCRIBED. VARIOUS VOLUME INCREASES AND LEAKAGE AREAS OCCUR DURING THE PIPE FAILURE, WHICH INFLUENCE TO A GREAT EXTENT THE THERMODYNAMIC PHENOMENA. THE LOADING OF THE PIPES AND OF THE BURST-PROTECTION ELEMENTS IS DETERMINED. (FAH)

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA, 22101
GERMANY * REACTOR * COMPONENTS * CONCRETE * PIPES AND PIPE FITTINGS * REACTOR, PWR

144575
HELDENBRAND G
KRAFTWERK POWER RAMP FUEL ROD IRRADIATION TESTS 1976/77 (GERMAN)
KRAFTWERK UNION, ERLANGEN, F.R.G. GERMANY
BMFT RS 203 + RE 23/024778 + GERRSR-345 +, 35 PPS, 2 TABS, 13 FIGS, 8 REFS, OCT, 1978

IRRADIATION EXPERIMENTS IN HER REITEN WERE CARRIED OUT TO DETERMINE THE OPERATIONAL BEHAVIOR OF FUEL RODS IN LIGHT WATER REACTORS DURING POWER RAMP. 36 PWR TEST FUEL RODS, WHICH HAD BEEN PRE-IRRADIATED IN A NUCLEAR POWER STATION UP TO BURNUPS OF ABOUT 25 GWDT (U), HAVE BEEN RAMPED IN A HELDANABLE PRESSURE BOILING CAPSULE. ON ALL FUEL RODS WITH HIGH RAMP TERMINAL POWERS, PEAKS OF FISSION PRODUCTS AT PELLET INTERFACES AND CRACKS, INSIGNIFICANT WIGGLS, PARTIAL DISH CLOSURE IN THE HIGH POWER REGION AND AN INCREASED APPEARANCE OF TRANSVERSE PELLET CRACKS HAVE BEEN DETERMINED.

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

GERMANY + IRRADIATION TESTING + REACTOR, LWR + FUEL ROD + CONTAINMENT INTEGRITY + CRACK + FISSION PRODUCT RELEASE

144296
SCHWICKERT H
EMERGENCY COOLING DEPRESSURIZATION RESEARCH; BLOCKED COOLING CHANNELS WITH BWR GEOMETRY (IN GERMAN)
KRAFTWERK UNION, ERLANGEN, F.R.G. GERMANY
REPORT RE 23/024778 + GERRSR-346 +, 145 PPS, 92 FIGS, 9 REFS, OCT, 1978

IN A TEST FACILITY OF TWO PARALLEL BWR-FUEL ASSEMBLIES EXPERIMENTS WERE CARRIED OUT WITH TOP SPRAY AND BOTTOM FLOODING. FOR THE SIMULATION OF GALLING OF THE FUEL ROD CLADDING (FLOW AREA RESTRICTIONS) ONE OF THE BUNDLES WAS PROVIDED WITH BLOCKAGE PLATES. THE TEST PARAMETERS WERE THE PRESSURE, THE SPRAY AND THE FLOODING RATES, THE HEATUP POWER AND THE INITIAL CLAD TEMPERATURE OF THE HEATERS. THE TEST RESULTS SHOWED, EXCEPT IN THE BLOCKED REGION, NO SIGNIFICANT VARIATIONS FROM THOSE WITHOUT BLOCKAGE. AN IMPROVED HEAT TRANSFER WAS OBSERVED IN A CLOSE REGION ABOVE THE BLOCKAGE IN THE CASE OF BOTTOM FLOODING AND BELOW IT IN THE CASE OF TOP SPRAY.

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

REACTOR, BWR + FLOW BLOCKAGE + FUEL ROD + FUEL SWELLING + TEMPERATURE + FLOW, TWO PHASE

144328
ENGEL H
DEVELOPMENT OF A SUCTION SYSTEM FOR INSTALLATIONS AND FITTINGS (IN GERMAN)
KRAFTWERK UNION, ERLANGEN, F.R.G. GERMANY
RE 23/027778 + BMFT RS 218 + GERRSR-323 +, 120 PPS, 1 TABS, 48 FIGS, 2 REFS, NOV, 1978

DESCRIBES A GLAND LEAK-OFF SYSTEM WITH FILTERS AND/OR ADSORBERS WHICH CONTINUOUSLY CLEANS UP A SIDE STREAM OF CONTAMINATED AIR FROM THE REACTOR LID, GLANDS, TANKS, CONTAINMENT PENETRATIONS, AND OTHER CRITICAL POINTS OF THE CONTAINED SYSTEM (CAB)

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

AIR CLEANING + ADSORPTION + CHARCOAL ADSORBER + FILTERS + WASTE TREATMENT, EQUIPMENT + REACTOR

147473
ENGEL H
INVESTIGATION AND DEVELOPMENT OF SYSTEMS LIMITING THE H₂-CONCENTRATION IN THE BWR CONTAINMENT (IN GERMAN)
KRAFTWERK UNION, ERLANGEN, F. GERMANY
BMFT RS 223 + RE 23/028773 + GERRSR-364 +, 163 PPS, 1 TABS, 1 FIGS, REFS, NOV, 1978

THE PURPOSE OF THE R + D PROGRAM IS TO IMPROVE OUR KNOWLEDGE OF HYDROGEN GENERATION AND DISTRIBUTION IN THE BWR CONTAINMENT DURING REACTOR OPERATION AND AFTER LOCA, AND ESPECIALLY TO DEVELOP AND TEST CONCEPTS AND METHODS FOR MEASUREMENTS AND LIMITATION OF H₂ CONCENTRATIONS IN THE CONTAINMENT ATMOSPHERE.

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

ACCIDENT, LOSS OF COOLANT + ACCIDENT ANALYSIS + R AND D PROGRAM + CONTAINMENT + TESTING + REACTOR, BWR + HYDROGEN + MEASUREMENT + GERMANY

147417
JAGER EH + EPFER HD
RS236 - FINAL REPORT CONTROLLED-BLASTING DEMOLITION OF RADIOACTIVE PRIMARY LOOP COMPONENTS OF DECOMMISSIONED NUCLEAR POWER PLANTS (IN GERMAN)
MESSERSCHMITT-BOLKOW-BLOHM GMBH, F.R.G. GERMANY
SOR-629 + RS236 + GERRSR-323 +, APPROX. 200 PPS, FIGS, DEC, 28, 1978

POSSIBLE WAYS OF DISMANTLING THE RADIOACTIVE PRIMARY LOOP COMPONENTS OF A BIBLIS-B-TYPE NUCLEAR POWER PLANT BY MEANS OF EXPLOSIVE DEVICES HAVE BEEN STUDIED. THE FOLLOWING PWR LARGE COMPONENTS WERE EXAMINED: STEAM GENERATORS, REACTOR COOLANT PUMPS, REACTOR VESSEL, PRIMARY PIPING, AND BIOLOGICAL SHIELD ASSUMING THAT (A) THE PLANT HAD BEEN OPERATED FOR 40 YEARS AT A 75% POWER LEVEL, (B) THE PRIMARY LOOPS HAD BEEN THOROUGHLY DECONTAMINATED BY CHEMICAL MEANS AFTER REACTOR

147317 *CONTINUED*
 DECOMMISSIONING, AND (C) THE COMPONENTS HAVE TO BE DISMANTLED INSIDE THE REACTOR CONTAINMENT BUILDING.

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

GERMANY * POWER PLANT, NUCLEAR * DECOMMISSIONING * REACTOR, PWR * COMPONENTS * EXPLOSION * STEAM GENERATOR * PUMPS * PRESSURE VESSELS * MAIN COOLING SYSTEM * PIPES AND PIPE FITTINGS * SHIELDING

146795
 EDER D * GASCH A * KAISER F
 SPECIFICATION OF CONDITIONS OF A NUCLEAR POWER PLANT WITH A PWR FOLLOWING A LOCA FOR PURPOSES OF STUDYING THE ONGOING DECONTAMINATION AND TRANSPORT (IN GERMAN)
 NIS, NUKLEAR-INGENIEUR-SERVICE, F.R. GERMANY
 NIS-337 * GERRSR-347 *, 230 PPS, TABS, FIGS, AUG, 1978

ASSUMPTIONS ARE MADE WHICH PROVIDE A CONSERVATIVE PICTURE OF THE REFERENCE PLANT STUDIED (PWR, 1300 MW) WITH RESPECT TO THE COURSE OF THE ACCIDENT AND THE RESULTING DAMAGE AS WELL AS THE DISTRIBUTION OF RADIOACTIVITY IN THE PLANT, ASSUMING A DOUBLE-ENDED RUPTURE OF THE HOT LINE IN THE PIPING CHANGER AND A FUEL ASSEMBLY CLADDING TUBE DAMAGE OF 10% CORRESPONDING TO THE LICENSING GUIDELINES CURRENTLY VALID FOR THE RELEASE OF IODINE. THE NUCLIDE-SPECIFIC DISTRIBUTION OF THE RADIOACTIVITY IN REFERENCE CHAMBERS IN THE CONTAINMENT IS DETERMINED WITH THE "CURRAL" COMPUTER PROGRAM.

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

*REACTOR, PWR * ACCIDENT, LOSS OF COOLANT * DECOMMISSIONING * TRANSPORTATION AND HANDLING * RADIOACTIVITY RELEASE * DISTRIBUTION * WASTE MANAGEMENT * GERMANY

147779
 MEMMENT G
 UNCERTAINTY OF THE FAILURE RATE OF COMPONENTS AND THE APPARENT INFLUENCE OF THESE OCCURRENCES IN FAULT TREE ANALYSIS (IN GERMAN)
 TECHNISCHE UNIVERSITAT BERLIN, F.R. GERMANY
 DMFT-RS-228 * GERRSR-385 *, 25 PPS, 1 TAB, NO DATE

THIS REPORT IS CONCERNED WITH THE UNCERTAINTY OF RELIABILITY DATA AS WELL AS ITS INFLUENCE ON THE RESULTS OF FAULT TREE CALCULATIONS. AFTER A SHORT COMMENT ON STATISTICAL PROBLEMS CONCERNING RELIABILITY DATA, THE AVAILABLE DATA IS DISCUSSED AND THE DEPENDENCE OF THE DATA SOURCES SHOWN. FINALLY SUITABLE DISTRIBUTION FUNCTIONS ARE PROPOSED TO DESCRIBE THE EXISTENT DATA. (ENF)

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

*FAULT TREE ANALYSIS * RELIABILITY ANALYSIS * DATA COLLECTION * ANALYTICAL TECHNIQUE

143774
 MAYINGER F * VIECENZ HJ
 PHASE SEPARATION (IN GERMAN)
 TECHNISCHE UNIV. HANNOVER, F.R. GERMANY
 DMFT-FB RS 179-03 * GERRSR-332 *, 133 PPS, FIGS, 39 REFS (NO DATE)

THERE WAS NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

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

PHASE CHANGE * HYDRAULIC ANALYSIS * GERMANY * ANALYTICAL TECHNIQUE * MATHEMATICAL TREATMENT

143901
 EXPERIMENTAL AND THEORETICAL RESEARCH ON THE THERMAL HYDRAULIC BEHAVIOR IN THE INITIAL BLOWDOWN PHASE, PARTS A, B, & C (IN GERMAN)
 TECHNISCHE UNIV. HANNOVER, F.R. GERMANY
 DMFT-FB RS 163-03 * GERRSR-334 *, APPROX. 150 PPS, FIGS (NO DATE)

THREE AREAS ARE DISCUSSED: ENTRAINMENT INVESTIGATION AND POST DRYOUT, MIXING INVESTIGATIONS, AND INVESTIGATIONS OF PRIMARY SYSTEM BEHAVIOR.

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

*THERMAL HYDRAULIC ANALYSIS * BLOWDOWN * EXPERIMENT * HEAT FLUX, DRYOUT * FLOW, MIXING * R AND D PROGRAM * GERMANY

146820
 RELIABILITY ASSESSMENT OF THE SECONDARY CONTAINMENT OF A PWR (IN GERMAN)
 TECHNISCHE UNIVERSITAT MUNCHEN, F.R. GERMANY
 DMFT RS 201 * GERRSR-366 *, 270 PPS, FIGS, REFS, SEPT, 1978

THE INTENTION OF THIS REPORT IS TO CONTRIBUTE TO THE DEVELOPMENT OF METHODS FOR THE RISK ANALYSIS OF NUCLEAR POWER PLANTS. FOR THIS PURPOSE A RELIABILITY ANALYSIS OF A STRUCTURAL COMPONENT, I.E. A REACTOR CONTAINMENT STRUCTURE IS CARRIED OUT. THE PROJECT CONSISTS BASICALLY OF THREE

144929 *CONTINUED*

CONCENTRATED EFFORTS OF THE STEEL MELT FOLLOWING A LOSS OF COOLANT ACCIDENT (LOCA); THE BEHAVIOR OF CONCRETE UNDER IMPACT LOAD CONDITIONS; AND FINALLY WITH THE ANALYSIS OF THE LOAD CONDITIONS. THIS INFORMATION IS THEN ASSEMBLED TO A COMPLEX RELIABILITY ANALYSIS. (FAH)

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

GERMANY * REACTOR, PWR * CONTAINMENT * RELIABILITY ANALYSIS * RISK * ANALYTICAL TECHNIQUE

145416

KORBER H * WEHLMANN H * UNGER H

INVESTIGATION OF THE FIRST PHASE OF THE CORE MELT ACCIDENT WITH MELSIM-1 IN A PWR AND A BWR STANDARD PLANT, AND THE COUPLING OF MELSIM-1 AND BILANZ-1, PART I (IN GERMAN)

UNIVERSITÄT STUTTGART, F.R.G. GERMANY

DMFT-85 211 (PART II) * GERRSR-363 * 106 PPS, FIGS, REFS, JUNE 1978

MELSIM-1 IS COUPLED WITH THE ENERGY BALANCE CODE BILANZ-1 IN ORDER TO OBTAIN THE INFLUENCE OF MELSIM-1 MODELS ON THE ATMOSPHERIC CONDITIONS IN THE CONTAINMENT. OVERALL RESULTS AS A FUNCTION OF ENERGY TRANSPORT BETWEEN REACTOR VESSEL AND CONTAINMENT OR CONTAINMENT PRESSURE AS A FUNCTION OF TIME ARE IN GOOD ACCORDANCE WITH VALUES OBTAINED BY A SINGLE ROD MODEL. THE COUPLED CALCULATIONS OF MELSIM-1 AND BILANZ-1 SHOW THAT THE CONTAINMENT PRESSURE STAYS BELOW THE DESIGN PRESSURE DURING THE TIME PERIOD CONSIDERED.

AVAILABILITY - AEC PUBLIC DOCUMENT RDG, 1717 H STREET, WASHINGTON, D. C. 20555 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00

GERMANY * ACCIDENT, CORE DISRUPTIVE * CORE MELTDOWN * ACCUMULATORS * REACTOR, PWR * CONTAINMENT ATMOSPHERE * ACCIDENT, LOSS OF COOLANT * REACTOR, PWR * REACTOR, BWR * ACCIDENT, FUEL SLUMP

145756

DIJANZ H * KORBER H * UNGER H

INVESTIGATION OF THE VARIOUS PHASES OF THE CORE MELT ACCIDENT AFTER THE AFTER FAILURE OF THE CORE SUPPORT STRUCTURE DUE TO THE FORMATION OF MELT OR DUE TO PRESSURE VESSEL FAILURE, PART II (IN GERMAN)

UNIVERSITÄT STUTTGART, F.R.G. GERMANY

DMFT-85 211 (PART III) * GERRSR-360 * 175 PPS, FIGS, REFS, JULY 1978

THIS REPORT CALCULATES THE MANNER IN WHICH A CORE MELT PROGRESSES IN BOTH A PWR AND A BWR REACTOR. SEVERAL COMPUTER PROGRAMS ARE USED TO SIMULATE THE VARIOUS PHASES OF THE ACCIDENT. CALCULATIONS INDICATE THAT THE PWR REACTOR VESSEL WILL BE MELTED THROUGH IN ABOUT 1 HOUR AFTER THE BEGINNING OF THE ACCIDENT. FOR THE BWR REACTOR MELT THROUGH THE REACTOR VESSEL OCCURS IN ABOUT 2 HOURS. (FAH)

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

ACCIDENT * CORE MELTDOWN * FAILURE * REACTOR, LWR * GERMANY * ANALYTICAL MODEL * PRESSURE VESSELS

144197

KOSPILO H * DUBR-WESTERHEIDE P

DEVELOPMENT OF A MASS-DENSITY METHOD FOR TRANSIENT TWO PHASE STATE USING ATOMIC RESONANCE (IN GERMAN)

DMFT-85 188 * GERRSR-340 * 77 PPS, FIGS, AUG, 1978

THERE WAS NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

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

GERMANY * MASS * INSTRUMENT, DENSITY * TRANSIENT * FLOW, TWO PHASE * MEASUREMENT

3. GERMAN (FRG) FAST REACTOR SAFETY RESEARCH REPORTS RECEIVED BY NRC

THE FOLLOWING IS A LISTING OF REPORTS RECEIVED FROM THE FEDERAL REPUBLIC OF GERMANY DURING THE FIRST HALF OF 1979 UNDER THE TECHNICAL EXCHANGE AGREEMENT.

143835
FISCHER T + MULLER K
SONIC-EMISSION MEASUREMENTS IN FRACTURE MECHANICS RESEARCH ON MATERIALS USED IN FAST SODIUM-COOLED REACTORS
(IN GERMAN)
BATTELLE-INSTITUT G.V., FRANKFURT AM MAIN, F.R.G. GERMANY
DP-R-62,945-3 + GERGER-330 +, 94 PPS, 2 TABS, 41 FIGS, 15 REF., AUG, 1978

THERE WAS NO ENGLISH ABSTRACT AVAILABLE AT THE TIME THIS DOCUMENT WAS PROCESSED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA, 22161
GERMANY + ACOUSTICS + MEASUREMENT + FRACTURE TOUGHNESS + PROPERTY, MECHANICAL + REACTOR, LMFBR

4. JAPANESE LIGHT-WATER REACTOR SAFETY RESEARCH REPORTS RECEIVED BY NRC

THE FOLLOWING IS A LISTING OF REPORTS RECEIVED FROM JAPAN DURING THE FIRST HALF OF 1979 UNDER THE TECHNICAL EXCHANGE AGREEMENT.

144523
TAKEDA T + NAGAI H
AN ANALYSIS OF THE ADDITIONAL FISSION PRODUCT RELEASE PHENOMENA (IN ENGLISH & JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7855 + JPNRSR-195 +, 51 PPS, 25 FIGS, 8 REFS, AUG, 1978

THE ADDITIONAL FISSION PRODUCT RELEASE BEHAVIOR THROUGH A DEFECT HOLE ON THE CLADDING OF FUEL RODS HAS BEEN STUDIED QUALITATIVELY WITH A COMPUTER PROGRAM CQDAC-ARFF. THE ADDITIONAL FISSION PRODUCT RELEASE PHENOMENA ARE DESCRIBED AS QUALITATIVE EVALUATION. THE ADDITIONAL FISSION PRODUCT RELEASE BEHAVIOR IN COOLANT TEMPERATURE AND PRESSURE FLUCTUATIONS AND IN REACTOR START-UP AND SHUT-DOWN DEPENDS ON COOLANT WATER FLOW BEHAVIOR INTO AND FROM THE FREE SPACE OF FUEL RODS THROUGH A DEFECT HOLE.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
FAILURE, CLADDING + FISSION PRODUCT RELEASE + COMPUTER PROGRAM + FAILURE, FUEL ELEMENT + JAPAN + IODINE + IN PILE EXPERIMENT

144512
MOCHIZUKI Y + SOBAYAMA M + SUZUKI M
ANALYSIS OF LOCA EXPERIMENTS WITH RELAP4J CODE (ANALYSIS OF ROSA-II EXPERIMENTS FOR COLD LEG BREAK RUNS 413 AND 312) (IN ENGLISH & JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7835 + JPNRSR-194 +, 94 PPS, FIGS, REFS, SEPT, 1978

THE TWO TESTS WERE PERFORMED UNDER EQUAL REACTOR INITIAL PRESSURE AND TEMPERATURE. IN THE RESPECTIVE DIFFERENT LPCI LOCATIONS, TYPICAL FACTORS INFLUENCING THE PRESSURE HISTORY WERE EXAMINED ANALYTICALLY. IN CONCLUSION, THE PREDICTIONS OF MACROSCOPIC-HYDRAULIC PHENOMENA SUCH AS PRESSURE TRANSIENT IN EACH LOCATION ARE GOOD, AND THE PREDICTIONS OF MICROSCOPIC-HYDRAULIC PHENOMENA SUCH AS STEAM-WATER SLIP VELOCITY, MULTI-DIMENSIONAL FLOW IN PLENUMS OR CORE, QUENCHING VELOCITY, COOLING OF FUEL RODS BY SMALL COOLANT FLOW ARE NOT GOOD. EXPERIMENTAL PHENOMENA NOT CLARIFIED YET WITH TEST DATA ARE PREDICTED WITH THE ANALYSIS. (MLW)

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
JAPAN + COMPUTER PROGRAM + ACCIDENT, LOSS OF COOLANT + PRESSURE TRANSIENT + FLOW THEORY AND EXPERIMENTS + EMERGENCY COOLING SYSTEM + THERMAL TRANSIENT

144529
KOHAYASHI K + SATO K
ASCOT-11: A COMPUTER PROGRAM FOR ANALYZING THE THERMO-HYDRAULIC BEHAVIOR IN A PWR CORE DURING A LOCA (IN ENGLISH)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7917 + JPNRSR-196 +, 53 PPS, 6 FIGS, 25 REFS, SEPT, 1978

THE CORE IS ASSUMED TO BE AXI-SYMMETRIC TWO-DIMENSIONAL AND THE CONSERVATION LAWS ARE SOLVED BY THE METHOD OF CHARACTERISTICS. FOR THE TEMPERATURE RESPONSE OF REPRESENTATIVE FUELS OF THE CONCENTRIC ANNULAR SUBREGIONS INTO WHICH THE CORE IS DIVIDED, THE HEAT CONDUCTION EQUATIONS ARE SOLVED BY THE EXPLICIT METHOD WITH AVERAGED FLOW CONDITIONS. THE BOUNDARY CONDITIONS AT THE UPPER AND LOWER PLENUM ARE GIVEN AS INPUTS. (MLW)

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
JAPAN + COMPUTER PROGRAM + COMPUTER PROGRAM, DIGITAL + THERMAL HYDRAULIC ANALYSIS + ACCIDENT, LOSS OF COOLANT + REACTOR, PWR + TEMPERATURE + METAL WATER REACTION

144526
SASAKI S
AN ANALYSIS OF LOFT L1-2 EXPERIMENT BY ALARM-P1 COMPUTER CODE (IN ENGLISH)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7947 + JPNRSR-198 +, 88 PPS, 76 FIGS, 14 REFS, OCT, 1978

PRELIMINARY TO NUCLEAR TESTS A SIMPLE BLOWDOWN EXPERIMENT WAS PERFORMED IN WHICH THE CORE IS COMPOSED OF A CONFIGURATION SIMULATING FRICTIONAL RESISTANCE AND THE OVERALL EXPERIMENTAL FACILITY IS MAINTAINED ISOTHERMALLY WITHOUT ECC WATER INJECTION. AT THE BEGINNING OF COMPUTATION, INPUT DATA WERE CHOSEN FROM RELAP4J DATA USED BY THE LOFT ANALYSIS GROUP AND THEN CONVERTED AS RELEVANT TO THE ALARM-P1 INPUT SPECIFICATIONS. BY AND LARGE, GOOD AGREEMENTS WERE OBTAINED BETWEEN CALCULATIONAL RESULTS AND EXPERIMENTAL DATA. (MLW)

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
JAPAN + COMPUTER PROGRAM + BLOWDOWN + COMPARISON, THEORY AND EXPERIENCE

143891
ROSA-II TEST DATA REPORT 12 EFFECTS OF ECCS INJECTION AND PUMP CIRCULATION ON LOCA PHENOMENA IN LARGEST COLD LEG BREAKS (RUNS 332, 413, 423) (IN ENGLISH & JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7944 + JPNRSR-197 +, 146 PPS, TABS, FIGS, NOV, 1978

143991 *CONTINUED*

RESULTS OF THE ROSA-II TESTS SIMULATING A LOSS-OF-COOLANT ACCIDENT (LOCA) AND THE EFFECTS OF AN EMERGENCY CORE COOLING SYSTEM (ECCS) IN A PRESSURIZED WATER REACTOR (PWR) ARE REPORTED AS WELL AS TEST CONDITIONS AND INTERPRETATIONS OF THE DATA IN TEST RUNS 332, 413 AND 425. EACH TEST WAS CARRIED OUT WITH A LARGE DOUBLE-ENDED COLD LEG BREAK. TEST PARAMETERS ARE ECC INJECTION, PUMP OPERATION AND INITIAL TEMPERATURE DIFFERENCE ACROSS THE CORE INFLUENCING THE PRIMARY COOLANT SYSTEM.

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

REACTOR, PWR * ACCIDENT, LOSS OF COOLANT * REACTOR TEST FACILITY * BLOWDOWN * CORE REFLUDDING * EMERGENCY COOLING SYSTEM * CORE * TEMPERATURE

144527

KOBAYASHI K * SASAKI S

SPADE: A COMPUTER SUBROUTINE FOR GENERATING STEAM TABLES HAVING PRESSURE AND DENSITY AS THE INDEPENDENT VARIABLES (IN ENGLISH & JAPANESE)

JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI

JAERI-M-7951 * JPNRSP-199 * 34 PPS, 3 FIGS, NOV, 1978

THE SPADE DIGITAL COMPUTER PROGRAM WAS DEVELOPED TO CALCULATE VARIABLE TRANSFORMATIONS AND PARTIAL DERIVATIVES BETWEEN PROPERTY VALUES WHICH ARE NECESSARY TO SOLVE THE MASS, MOMENTUM, AND ENERGY CONSERVATION LAWS HAVING PRESSURE AND DENSITY AS INDEPENDENT VARIABLES. THE OUTPUTS ARE TABLES OF TEMPERATURE, SONIC VELOCITY AND THE PARTIAL DERIVATIVE OF H WITH RESPECT TO PWD AT CONSTANT PRESSURE HAVING PRESSURE AND DENSITY AS THE INDEPENDENT VARIABLES. (MLW)

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

*STEAM * DATA COLLECTION * COMPUTER PROGRAM * COMPUTER PROGRAM, DIGITAL * VAPOR PRESSURE * TEMPERATURE * ACOUSTICS * THERMAL PROPERTY * PROPERTY, PHYSICAL * JAPAN

143974

OHNISHI N * TANZAWA S * KITANO T

EFFECT OF HEAT GENERATION PROFILE IN PELLETS ON FUEL FAILURE BEHAVIOR (ENRICHMENT PARAMETER TEST IN NSRR) (IN ENGLISH & JAPANESE)

JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO

JAERI-M-7990 * JPNRSP-201 * 54 PPS, FIGS, REFS, NOV, 1978

THE EFFECT OF HEAT GENERATION PROFILE IN PELLETS ON FUEL FAILURE BEHAVIOR HAS BEEN EXAMINED FOR 5%, 10%, AND 20% ENRICHED FUEL RODS IN NSRR TESTS. THE FAILURE THRESHOLD ENERGY DEPOSITION DECREASES WITH INCREASING ENRICHMENT OF THE FUEL RODS. THE FAILURE THRESHOLD ENERGY DEPOSITIONS FOR 5%, 10%, AND 20% ENRICHED FUEL RODS ARE ABOUT 278, 265 AND 248 CAL/GD(SUB 2), RESPECTIVELY. FUEL FAILURE DETERIORATION OF THE CLADDING, DISP 143986 HDR

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

JAPAN * ACCIDENT, REACTIVITY * FAILURE, CLADDING * HEAT GENERATION, INTERNAL * FUEL ROD * CENTERLINE MELTING

147064

KOIZUMI Y * KIKUCHI O * SODA K

PREDICTION OF ROSA-III EXPERIMENT RUN 702 (IN JAPANESE)

JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI

JAERI-M-7970 * JPNRSP-204 * 76 PPS, FIGS, NOV, 1978

RUN 702 REPRESENTS A TYPICAL 200% DOUBLE ENDED RECIRCULATING PIPE BREAK AT PUMP SUCTION SIDE. ECCS IS NOT ACTIVATED. INITIAL CORE POWER AND FLOW RATE IS 3.73 MW AND 36.4 KG/SEC RESPECTIVELY. SOME MAJOR RESULTS ARE: 1) LOWER PLENUM FLASHING IS PREDICTED TO OCCUR AT 3.7 SEC AFTER BREAK. 2) FLOW DIRECTION IN BROKEN LOOP JET PUMP REVERSES IMMEDIATELY AFTER BREAK. 3) INTACT LOOP JET PUMP LOoses ITS FUNCTION AT 10.5 SEC. 5) SURFACE TEMPERATURE OF THE SIMULATED FUEL ROD DOES NOT EXHIBIT AN EXCURSION TO HIGH TEMPERATURE, ALTHOUGH TEMPERATURE BEGINS TO SLOWLY INCREASE WHEN QUALITY IN THE CORE BECOMES 1.0 AT ABOUT 40 SEC. AFTER BREAK.

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

JAPAN * FLOW THEORY AND EXPERIMENTS * HYDRAULIC EXPERIMENT * THERMAL EXPERIMENT * THERMAL HYDRAULIC ANALYSIS * COMPUTER PROGRAM * REACTOR, BWR * FUEL ROD * SIMULATION

143893

SUDDO Y * MURAO Y

EXPERIMENT OF THE DOWNCOMER EFFECTIVE WATER HEAD DURING A REFLUDD PHASE OF PWR LOCA (IN JAPANESE)

JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI

JAERI-M-7978 * JPNRSP-200 * 93 PPS, 62 FIGS, DEC, 1978

THE RESULTS AND ANALYSIS ARE DESCRIBED OF A DOWNCOMER EFFECTIVE WATER HEAD EXPERIMENT. DOWNCOMER EFFECTIVE WATER HEAD IS THE DRIVING FORCE TO FEED AN EMERGENCY COOLANT TO THE CORE DURING A REFLUDD PHASE OF PWR LOCA. THE TEST RIG HAS DIMENSIONS OF THE FULL-SCALE HEIGHT AND GAP. THE EFFECTIVE WATER HEAD HISTORIES OBTAINED BY EXPERIMENT WERE COMPARED WITH THOSE PREDICTED FROM THE HEAT RELEASE FROM THE DOWNCOMER WALLS. THE HEAT RELEASE WAS CALCULATED FROM THE TEMPERATURE HISTORIES INDICATED BY THERMOCOUPLES INSTRUMENTED IN AND ON THE WALLS DURING EXPERIMENT.

143997 *CONTINUED*
 AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, PWR * ACCIDENT, LOSS OF COOLANT * FLOW, TWO PHASE * VOID FRACTION * THERMAL HYDRAULIC ANALYSIS *
 PRESSURE DROP * PRESSURE TRANSIENT * CORE REFLOODING

147532
 WATO K * SASAKI S * ANAYA F
 ALARM-PII: A COMPUTER PROGRAM FOR PRESSURIZED WATER REACTOR BLOWDOWN ANALYSIS (IN ENGLISH)
 JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
 JAPRI-R-8034 * JPNRSR-205 * 103 PPS, 20 FIGS, 35 REFS, DEC, 1978

ALARM-PII MODELS THE PWR SYSTEM FLUID CONDITIONS INCLUDING FLOW, PRESSURE, MASS INVENTORY, FLUID
 QUALITY AND HEAT TRANSFER. IT SOLVES INTEGRAL FORMS OF FLUID CONSERVATION AND STATE EQUATIONS
 FOR USER DEFINED VOLUMES TREATED AS ONE-DIMENSIONAL HOMOGENEOUS, THERMAL-EQUILIBRIUM ELEMENTS
 WITH INTERCONNECTING FLOW PATHS. IT ALSO PROVIDES THE INITIAL CONDITIONS FOR ANALYSIS OF THE
 LAST PORTION OF THE LOCA TRANSIENT, A REFLOW PHASE, AND THE INFORMATION FOR CORE HEAT-UP
 ANALYSIS DURING THE WHOLE LOCA.

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

JAPAN * COMPUTER PROGRAM * REACTOR, PWR * THERMAL HYDRAULIC ANALYSIS * ACCIDENT, LOSS OF COOLANT * BLOWDOWN
 * CORE REFLOODING * FLOW, CRITICAL

148332
 TAKEDA T * HIRANO K
 FUEL-COOLANT INTERACTION EXPERIMENT BY DIRECT ELECTRICAL HEATING METHOD (ZHD12- H2O SYSTEM) (IN JAPANESE)
 JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
 JAPRI-R-8035 * JPNRSR-202 * 94 PPS, 27 FIGS, 4 REFS, JAN, 1979

IN THE PCM (POWER-COOLING MISMATCH) EXPERIMENTS, THE FCI (FUEL-COOLANT INTERACTION) TEST IS ONE OF
 NECESSARY TESTS IN ORDER TO PREDICT VARIOUS PHENOMENA THAT OCCUR DURING PCM IN THE CORE. A
 DIRECT ELECTRICAL HEATING METHOD IS USED FOR THE FCI TESTS FOR FUEL PELLETT TEMPERATURE OF OVER
 1000C. TEMPERATURE CHANGES OF COOLANT AND FUEL SURFACE, AS WELL AS THE PRESSURE CHANGE OF
 COOLANT WATER, WERE MEASURED. THE MOLTEN FUEL INTERACTED WITH THE COOLANT AND GENERATED SHOCK
 WAVES. THIS REPORT SHOWS THE MEASURED COOLANT PRESSURE CHANGES AND THE COOLANT TEMPERATURE
 CHANGES, AS WELL AS PHOTOGRAPHS OF DAMAGED FUEL PIN AND FUEL FRAGMENTS.

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

JAPAN * FUEL-COOLANT INTERACTION * REACTOR TRANSIENT * EXPERIMENT * MOLTEN FUEL * FUEL ROD * DAMAGE *
 SHOCK WAVE * REACTOR, LWR

148844
 IWAHARA T * KURCYANAGI T
 FLOW REDUCTION TRANSIENT BURNDOUT IN AN ANNULAR TEST SECTION (IN JAPANESE & ENGLISH)
 JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
 JAPRI-R-8047 * JPNRSR-203 * 106 PPS, FIGS, JAN, 1979

IN ORDER TO UNDERSTAND THE TRANSIENT BOILING PHENOMENA DURING PCM (POWER-COOLING-MISMATCH) IN
 LIGHT WATER REACTORS, TRANSIENT BURNDOUT EXPERIMENTS WERE PERFORMED USING A VERTICAL ANNULAR TEST
 SECTION UNDER ATMOSPHERIC PRESSURE. THE EXPERIMENTAL RESULTS SHOWED THAT BEYOND A FLOW REDUCTION
 RATE OF ABOUT 5 CM/SEC/SEC (1.4 MM GAP) AND ABOUT 1 CM/SEC/SEC (2.0 MM GAP), BURNDOUT MASS
 VELOCITY BECAME LOWER THAN THE STEADY STATE ONE. WHEN THE FLOW REDUCTION RATES WERE FURTHER
 INCREASED TO 20 TO 40 CM/SEC/SEC OR BEYOND, THE BURNDOUT DELAY TIME BECAME CONSTANT AT ABOUT 0.4
 SEC.

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

JAPAN * EXPERIMENT * FLOW, TWO PHASE * HEAT FLUX, BURNDOUT * HEAT FLUX, CRITICAL * FLOW, ANNULAR * REACTOR,
 LWR * TRANSIENT * BOILING

5. JAPANESE FAST REACTOR SAFETY RESEARCH REPORTS RECEIVED BY NRC

THE FOLLOWING IS A LISTING OF REPORTS RECEIVED FROM JAPAN DURING THE FIRST HALF OF 1979 UNDER THE TECHNICAL EXCHANGE AGREEMENT.

143776

OZAKI Y + HAGA K + KIKUCHI Y
ACOUSTIC NOISES WITH LOSS-OF-FLOW SODIUM BOILING EXPERIMENT IN A 19-PIN BUNDLE (IN ENGLISH)
POWER REACTOR & NUCLEAR FUEL DEVELOPMENT CORP., JAPAN
PNC N941 78-140 + JPNRSR-189F +, 9 PPS, 7 FIGS, 7 REFS, OCT, 1978

THIS PAPER DEALS WITH THE MEASUREMENT OF ACOUSTIC NOISES IN LMFBR FUEL SUBASSEMBLY. THE INTENSITY OF BOILING ACOUSTIC NOISES MEASURED WITH THE WAVEGLIDE METHOD WAS MUCH HIGHER THAN BACKGROUND NOISES. A DISTINCT PEAK COULD EASILY BE DISTINGUISHED FROM THE RESONANCE PEAKS OF THE EXPERIMENTAL SYSTEM. THE WAVEFORM OF THE BOILING ACOUSTIC NOISES WAS SIMILAR TO THE BURST TYPE ACOUSTIC EMISSION. THE PROPAGATION SPEED OF ACOUSTIC NOISES AGREED WELL WITH A PREDICTION BY THE THEORY BASED ON THE ASSUMPTION THAT THE MEASURED ACOUSTIC SIGNALS WERE TRANSMITTED ON THE PIPE AS SURFACE WAVES (RAYLEIGH WAVES) OR LAMB WAVES. (ENH)

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

MEASUREMENT + ACOUSTICS + MEASUREMENT, NOISE + SODIUM + BOILING + REACTOR, LMFBR

143907

UOTANI M + HAGA K + KIKUCHI Y + HORI M
LOCAL FLOW BLOCKAGE EXPERIMENTS IN 37-PIN SODIUM COOLED BUNDLES WITH GRID SPACERS (IN ENGLISH)
POWER REACTOR & NUCLEAR FUEL DEVELOPMENT CORP., JAPAN
PNC N941 78-141 + JPNRSR-190 +, 15 PPS, 13 FIGS, 5 REFS, OCT, 1978

A SERIES OF OUT-OF-PILE EXPERIMENTS WERE CONDUCTED ON LOCAL TEMPERATURE RISES DUE TO NON-HEAT GENERATING BLOCKAGES IN 37-PIN BUNDLES. IN THE CENTRAL BLOCKAGE EXPERIMENT, THE CENTRAL 24 SUBCHANNELS OF THE BUNDLE WERE BLOCKED WITH A 5 MM THICK STAINLESS-STEEL PLATE AT UPSTREAM END OF A GRID SPACER. THE BLOCKED AREA WAS 27% OF THE TOTAL FLOW AREA. IN THE EDGE BLOCKAGE EXPERIMENT, A STAINLESS-STEEL PLATE BLOCKED 39 SUBCHANNELS OF A 1/2 EDGE PART OF THE CROSS-SECTIONAL AREA. THE DIMENSIONLESS COOLANT RESIDENCE TIME WAS FOUND INDEPENDENT OF REYNOLDS NUMBER EXCEPT IN THE LOW NUMBER RANGE, AND THE VALUE OBTAINED IN THE EDGE BLOCKAGE EXPERIMENT WAS ABOUT 2.4 TIMES AS MUCH AS THAT OBTAINED IN THE CENTRAL BLOCKAGE EXPERIMENT. WHEN EXPERIMENTAL RESULTS WERE EXTRAPOLATED TO THE REACTOR CONDITION, AN EDGE BLOCKAGE OF MORE THAN 30% MIGHT CAUSE LOCAL BOILING IN THE WAKE REGION, WHILE A CENTRAL ONE WOULD NOT CAUSE LOCAL BOILING IN ANY BLOCKAGE RATIO LESS THAN 60%. THE TEMPERATURE RISES IN THE BLOCKED GRID SPACER WERE ALSO DISCUSSED.

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

FLOW BLOCKAGE + BOILING + TEMPERATURE + REACTOR, LMFBR + MASS TRANSFER + OUT OF PILE EXPERIMENT

143343

PROGRESS REPORT ON FAST BREEDER REACTOR DEVELOPMENT IN JAPAN, APRIL-JUNE 1978
POWER REACTOR & NUCLEAR FUEL DEVELOPMENT CORP., JAPAN
PNC N251 78-06 + JPNRSR-192F +, 11 PPS, NOV, 1978

DEVELOPMENT IN THE FOLLOWING AREAS IS DISCUSSED: THE EXPERIMENTAL FAST REACTOR JOYO; THE PROTOTYPE FBR MONJU; REACTOR PHYSICS; STRUCTURAL COMPONENTS; INSTRUMENTATION AND CONTROL; SODIUM TECHNOLOGY; FUEL MATERIALS; REACTOR CORE SAFETY; AND STEAM GENERATOR.

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

*JAPAN + R AND D PROGRAM + REACTOR, LMFBR + REACTOR, FAST + REACTOR, BREEDER + REACTOR PHYSICS + CONTROL SYSTEM + STRUCTURAL INTEGRITY + COMPONENTS + SODIUM + FUEL, NUCLEAR + SAFETY ANALYSIS + STEAM GENERATOR

6. U.K. LIGHT-WATER REACTOR SAFETY RESEARCH REPORTS RECEIVED BY NRC

THE FOLLOWING IS A LISTING OF REPORTS RECEIVED FROM THE U.K. DURING THE FIRST HALF OF 1979 UNDER THE TECHNICAL EXCHANGE AGREEMENT.

120343
MARTIN OJV
LASER HOLOGRAPHIC AND SPECKLE PHOTOGRAPHY METHODS FOR DEFECT DETECTION AND STRAIN EVALUATION IN PRESSURE VESSELS
UKAEA SAFETY & RELIABILITY DIRECTORATE, WARRINGTON, U.K.
SRD-R-60 + UKRSR-183 +, 30 PPS, 22 FIGS, 3 REFS, SEPT, 1976

HOLOGRAPHIC INTERFEROMETRY AND LASER SPECKLE PHOTOGRAPHY ARE COHERENT OPTIC TECHNIQUES CAPABLE OF SHOWING, RECORDING, AND EVALUATING THE PHYSICAL EFFECTS OF DYNAMIC EVENTS. THIS PAPER EXPLAINS THE HOLOGRAPHIC TECHNIQUES, WITH AN EXAMPLE VISUALISING BURIED DEFECTS IN TUBES, AND DESCRIBES THE USE OF A SAFETY LASER FOR HOLOGRAPHIC AND SPECKLE PHOTOGRAPHY METHODS, TO EVALUATE STRAIN, AND FINALLY GIVES CONCLUSIONS. STRAIN VALUES IN FRONT OF A CRACK TIP IN A 75 MM THICK PRESSURE VESSEL BY SPECKLE PHOTOGRAPHY IS GIVEN.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
STRESS + PRESSURE VESSELS + TEST, NONDESTRUCTIVE + LASER + FLAW

145040
BRITTAIN J + BRYCE WM + GREEN C
THE STATUS OF RELAP-UK MK III AT JULY 1976 - A PROGRAM FOR TRANSIENT THERMAL-HYDRAULIC ANALYSIS
UKAEA ATOMIC ENERGY ESTABLISHMENT, DORSET, U.K.
AEEW-R-1083 + SGHWHTG/P(77)322 + UKRSR-177 +, 105 PPS, MAY 1977

RELAP-UK MKIII WAS DEVELOPED IN SUPPORT OF THE STEAM GENERATING HEAVY WATER REACTOR. THE MAJOR CHANGE OVER EARLIER VERSIONS IS THE INTRODUCTION OF AN IMPLICIT SCHEME FOR THE INTEGRATION OF THE EQUATIONS OF HYDRODYNAMICS, WHICH RESULTS IN RUNNING TIME IMPROVEMENTS OF UP TO A FACTOR OF TEN. ANOTHER NEW FEATURE IS THE RELAP4 FOUR-QUADRANT DYNAMIC PUMP MODEL. IMPROVEMENTS ALSO INCLUDE REVISED MOMENTUM FLUX AND KINETIC ENERGY TERMS IN THE CONSERVATION EQUATIONS, AND AN OPTION TO RAMP ON/OFF VALVES OVER A FINITE TIME OF OPERATION.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
UNITED KINGDOM + THERMAL HYDRAULIC ANALYSIS + COMPUTER PROGRAM + TRANSIENT + REACTOR, LWR + FLOW, TWO PHASE + HEAT FLUX, DRYOUT

148847
NASH G
AN APPRAISAL OF SUBCOOLED BOILING AND SLIP RATIO FROM MEASUREMENTS MADE IN LINGEN BWR
UKAEA ATOMIC ENERGY ESTABLISHMENT, DORSET, U.K.
AEEW-R-1128 + UKRSR-178 +, 28 PPS, 2 TABS, 12 FIGS, 14 REFS, AUG, 1977

MEASUREMENTS OF STEAM BUBBLE VELOCITIES AND VOIDAGE HAVE BEEN MADE IN THE RELATIVELY SMALL CORE B OF LINGEN BWR. THE RESULTS OF AXIAL SCANNING IN ONE RADIAL POSITION HAVE PRODUCED EXPERIMENTAL VALUES OF SLIP RATIO, POWER (FROM A TRAVELLING INCORE PROBE), VOIDAGE AND COOLANT MEAN DENSITY OVER THE CORE HEIGHT FOR THIS POSITION. THIS ONE SET OF DISTRIBUTIONS HAS ENABLED TESTING OF CURRENT UKAEA MODELS OF SUBCOOLED BOILING AND SLIP RATIO AGAINST EXPERIMENTS. FROM THE COMPARISONS, IT APPEARS THAT THE ONSET OF VOIDING CAN BE PREDICTED WELL, BUT THE ASSUMPTION THAT A CONSTANT FRACTION OF THE HEAT FLUX FORMS STEAM IN THE SUBCOOLED REGION NEEDS MODIFYING.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
BOILING + SUBCOOLING + VOID + BUBBLE + MEASUREMENT + REACTOR, BWR + ANALYTICAL TECHNIQUE + UNITED KINGDOM + COMPARISON, THEORY AND EXPERIENCE

145877
FRASER DC
IN-SITU TESTING OF HIGH EFFICIENCY FILTERS AT AEE WINFRITH
UKAEA ATOMIC ENERGY ESTABLISHMENT, DORSET, U.K.
AEEW-M-1510 + UKRSR-181 +, 19 PPS, 2 TABS, 4 FIGS, 11 REFS, OCT, 1977

EXPERIENCE WITH IN-PLACE TESTING OF INSTALLED HEPA FILTERS, SYSTEMS, USING A CONDENSATION NUCLEI TECHNIQUE, IS DESCRIBED. ALSO INCLUDED IS A COMPARISON OF THIS METHOD WITH THE DOP TEST AND SODIUM CHLORIDE TEST.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
AIR CLEANING + TEST, FILTER SYSTEM + FILTER EFFICIENCY + FILTER, HEPA + TESTING + TEST, FILTER

145641
HOLMES JA
DEVELOPMENT OF THE BUBBLE RISE MODEL IN RELAP-UK
UKAEA ATOMIC ENERGY ESTABLISHMENT, DORSET, U.K.
AEEW-M-1540 + UKRSR-180 +, 19 PPS, 2 FIGS, 5 REFS, NOV, 1977

SEVERAL IMPROVEMENTS HAVE BEEN MADE TO THE "BUBBLE RISE CALCULATION" IN THE CODE RELAP-UK. IN PARTICULAR, THE CALCULATION OF THE BUBBLE RISE VELOCITY IS CONSISTENT WITH THE RELAP-UK DRIFT FLUX CORRELATION. IT IS NOW POSSIBLE TO REPRESENT A VERTICAL COLUMN BY A STACK OF VERTICALLY-ADJACENT BUBBLE-RISE VOLUMES. ANY MIXTURE LEVEL EXISTING WITHIN THE COLUMN CAN FREELY PASS

145641 *CONTINUED*

BETWEEN THE VOLUMES IN THE STACK, THESE FACILITIES ARE DEMONSTRATED IN THIS PAPER BY A SIMPLE COMPUTATIONAL EXAMPLE.

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

UNITED KINGDOM * BUBBLE * COMPUTER PROGRAM * VOID FRACTION * FLOW, TWO PHASE

147795

HALSALL M J

A REVIEW OF INTERNATIONAL SOLUTIONS TO NEACRP BENCHMARK BWR LATTICE CELL PROBLEMS
UKAEA ATOMIC ENERGY ESTABLISHMENT, GORSEY, U.K.

AEA-R-1052 * UKRSR-179 *, 31 PPS, 5 TABS, 9 FIGS, 11 REFS, DEC. 1977

THIS PAPER SUMMARIZES INTERNATIONAL SOLUTIONS TO A SET OF BWR BENCHMARK PROBLEMS. THE PROBLEMS, POSED AS AN ACTIVITY SPONSORED BY THE NUCLEAR ENERGY AGENCY COMMITTEE ON REACTOR PHYSICS, WERE AS FOLLOWS: (1) 9-PIN SUPERCELL WITH CENTRAL BURNABLE POISON PIN; (2) MINI-BWR WITH 4 PIN-CELLS AND WATER GAPS AND CONTROL ROD CRUCIFORM; (3) FULL 7 X 7 PIN BWR LATTICE CELL WITH DIFFERENTIAL U(235) ENRICHMENT; AND (4) FULL 8 X 8 PIN BWR LATTICE CELL WITH WATER-HOLE, PU LOADING, BURNABLE POISON, AND HOMOGENIZED CRUCIFORM CONTROL ROD. SOLUTIONS HAVE BEEN CONTRIBUTED BY DENMARK, JAPAN, SWEDEN, SWITZERLAND AND THE UK.

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

COMPARISON * COMPUTER PROGRAM * TRANSPORT THEORY * CODES AND STANDARDS * FLUX DISTRIBUTION * REACTOR, BWR * REACTOR PHYSICS * QUALITY ASSURANCE * NUMERICAL METHOD * INTERNATIONAL

143910

DULLFORCE TA * DE W JELPHS AN * HIMMER W

THERMAL INTERACTIONS BETWEEN CERROTRU AND WATER
UKAEA CULHAM LAB., OXON, U.K.

CLM-44/52/17 * UKRSR-169 *, 8 PPS, 2 TABS, 2 FIGS, 1977

FUEL-COOLANT INTERACTIONS BETWEEN WATER AND 20 G SAMPLES OF THE LOW MELTING POINT ALLOY CERROTRU HAVE, AS IN THE PREVIOUSLY REPORTED CASES OF TIN AND CERROBEND, SHOWN THE EXISTENCE OF A WELL DEFINED ZONE IN FUEL TEMPERATURE-COOLANT TEMPERATURE SPACE WITHIN WHICH FCIS MAY OCCUR SPONTANEOUSLY. THE MINIMUM FUEL TEMPERATURE REQUIRED IS SLIGHTLY DEPENDENT ON COOLANT TEMPERATURE AND IS SHOWN TO CORRESPOND VERY CLOSELY TO AN INTERFACE TEMPERATURE LINE CORRESPONDING TO HOMOGENEOUS NUCLEATION

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

*FUEL COOLANT INTERACTION * EXPLOSION * TEMPERATURE * TIN * TEMPERATURE * MEASUREMENT * THERMAL CONDUCTIVITY

148848

DULLFORCE TA * HIMMER W

THERMAL INTERACTIONS BETWEEN CERROBEND AND WATER
CULHAM LAB., OXON, U.K.

CLM-RR/52/18 * UKRSR-184 *, 13 PPS, 4 TABS, 3 FIGS, 6 REFS, 1977

DROP TYPE FCI EXPERIMENTS HAVE BEEN PERFORMED USING 8-13 G SAMPLES OF THE LOW MELTING POINT ALLOY CERROBEND AS FUEL AND WATER AS COOLANT. ALTHOUGH THE COMPLETE TEMPERATURE INTERACTION ZONE HAS BEEN DETERMINED SPONTANEOUS INTERACTIONS OCCUR ONLY FOR SPECIFIC COMBINATIONS OF FUEL AND COOLANT TEMPERATURE (AS FOR MOLTEN TIN DROPPED INTO WATER). IT IS SHOWN THAT IN THIS SYSTEM THE INTERFACE TEMPERATURE MUST EXCEED THE COOLANT HOMOGENEOUS NUCLEATION TEMPERATURE FOR FRAGMENTATION TO OCCUR WHILE FOR EXPLOSIVE INTERACTIONS MUCH HIGHER INITIAL FUEL TEMPERATURES ARE REQUIRED.

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

*FUEL COOLANT INTERACTION * EXPERIMENT * THERMAL ANALYSIS * MEASUREMENT * UNITED KINGDOM

140447

MARTIN D

HUBBLE BUBBLE II: A COMPUTER PROGRAM TO DESCRIBE THERMAL NON-EQUILIBRIUM FLOW OF WATER IN SIMPLE PIPE SYSTEMS
UKAEA SAFETY & RELIABILITY DIRECTORATE, WARRINGTON, UK

SRD R 118 * UKRSR-182 *, 38 PPS, 19 FIGS, 8 REFS, JULY 1978

DESCRIBES THE COMPUTER PROGRAM HUBBLE-BUBBLE II WHICH CONSIDERS THE FLOW OF A TWO-PHASE MIXTURE THROUGH SIMPLE PIPE SYSTEMS. THE WATER-STEAM MIXTURE IS NOT IN THERMAL EQUILIBRIUM, THE FORMATION OF THE STEAM BEING CONTROLLED BY HEAT FLOW TO THE BUBBLES. THE PROGRAM IS USED TO INVESTIGATE TRANSIENT FLOW FROM A PIPE SYSTEM CONTAINING PRESSURIZED HOT WATER, THE FLOW BEING INITIATED BY THE BURSTING OF A DISC AT ONE END OF THE PIPE SYSTEM.

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

*COMPUTER PROGRAM * FLOW, TWO PHASE * PIPES AND PIPE FITTINGS * HEAT TRANSFER ANALYSIS * BUBBLE * PRESSURE TRANSIENT * WATER * UNITED KINGDOM * STEAM

144773
 MACINNIS DA
 THE ELECTRONIC CONTRIBUTION TO THE THERMODYNAMICS OF UO₂
 UKAEA SAFETY & RELIABILITY DIRECTORATE, WARRINGTON, U.K.
 SRD R 117 + UKRSR-171 +, 8 PPS, 1 TAB, 2 FIGS, 11 REFS, AUG, 1978

THE SPECIFIC HEAT, CP(T), OF UO₂ SHOWS A RAPID INCREASE BETWEEN 1500K AND 3100K (THE MELTING POINT). IT HAS BEEN CUSTOMARY TO INTERPRET THIS PEAK IN TERMS OF FORMATION OF DEFECTS IN THE PERFECT LATTICE. HOWEVER, STUDIES OF THE ELECTRICAL CONDUCTIVITY OF UO₂ HAVE SHOWN IT TO BE THAT OF A SEMICONDUCTOR WITH A BAND GAP OF APPROXIMATELY 2 EV. THE FORMATION ENERGY OF DEFECTS IS CALCULATED TO BE BETWEEN 3.25 AND 5.5 EV, SO THE ACTIVATION ENERGY OF ELECTRONIC EXCITATION IS CONSIDERABLY LOWER THAN THAT OF DEFECT FORMATION. THIS PAPER PROPOSES A RE-INTERPRETATION OF THE PEAK IN CP(T) IN TERMS OF ELECTRONIC EXCITATION, AND SHOWS A SIMPLE BUT REALISTIC MODEL OF THE UO₂ VALENCE + CONDUCTION BAND STRUCTURE. (FAH)

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
 URANIUM DIOXIDE + THERMODYNAMICS + MICROSTRUCTURE + ELECTRICAL CONDUCTION + THERMAL PROPERTY

146483
 HALL SF
 A SIMPLE HOMOGENEOUS EQUILIBRIUM CRITICAL DISCHARGE MODEL APPLIED TO MULTI-COMPONENT, TWO-PHASE SYSTEMS - THE COMPUTER PROGRAMS CRITS AND CRITER
 UKAEA SAFETY & RELIABILITY DIRECTORATE, WARRINGTON, U.K.
 UKRSR-170 + SRD R 127 +, 33 PPS, 2 TABS, 6 REFS, SEPT, 1978

A SIMPLE HOMOGENEOUS EQUILIBRIUM MODEL OF TWO-PHASE CRITICAL DISCHARGE FROM A RESERVOIR IS DESCRIBED, FOR USE IN SAFETY CALCULATIONS. THE ASSUMPTIONS ON WHICH THE MODEL IS BASED ARE DISCUSSED AND THE SOLUTION METHOD IS DESCRIBED; A COMPUTER PROGRAM, CRITS, WHICH SOLVES THE MODEL EQUATIONS IS GIVEN. THE MODEL IS THEN EXTENDED TO THE CASE WHERE THE FLUID IN THE RESERVOIR UNDER CONSIDERATION IS A MIXTURE OF SEVERAL COMPONENTS. ADDITIONAL ASSUMPTIONS ARE MADE AND AN EASILY SOLVABLE SET OF EQUATIONS DERIVED. AGAIN, A COMPUTER PROGRAM, CRITER, IS DESCRIBED WHICH SOLVES THIS MORE GENERAL SET OF EQUATIONS. THE SINGLE COMPONENT MODEL IS COMPARED WITH CRITICAL DISCHARGE RATES FOR WATER SYSTEMS DERIVED BY A DIFFERENT METHOD BUT WITH SIMILAR ASSUMPTIONS. EXAMPLES OF THE USE OF BOTH COMPUTER PROGRAMS ARE GIVEN.

AVAILABILITY - THE EDITOR, UNITED KINGDOM ATOMIC ENERGY AUTHORITY, SAFETY & RELIABILITY DIRECTORATE, CULCHETH, WARRINGTON WA3 4NE, ENGLAND

DISCHARGE + COMPONENTS + FLOW, TWO PHASE + COMPUTER PROGRAM + EQUATION OF STATE

148845
 MACINNIS DA
 THE ELECTRONIC CONTRIBUTION TO THE THERMODYNAMICS OF MOLTEN UO₂
 UKAEA SAFETY & RELIABILITY DIRECTORATE, WARRINGTON, UK
 SRD R 130 + UKRSR-230 +, 6 PPS, 3 TABS, 14 REFS, SEPT, 1978

THE SPECIFIC HEAT OF MOLTEN UO₂ WAS ANALYZED AT ITS MELTING POINT (TSUB M) AND SUGGEST THERE EXISTS A MECHANISM WHICH CAN ABSORB INTERNAL ENERGY AND WHICH IS PRESENT IN UO₂ BUT NOT IN OTHER IONIC AB(2) COMPOUNDS SUCH AS CaF₂. THIS MECHANISM WAS IDENTIFIED WITH ELECTRONIC EXCITATION. THE CALCULATED VALUE OF C(SUB V)(T)(SUB M) WAS COMPARED WITH THAT IN CURRENT USE AND SHOW THAT A MAJOR DISCREPANCY EXISTS. IT SEEMS POSSIBLE THAT ERRONEOUS EXTRAPOLATION OF C(SUB P)(T) BETWEEN T = T(SUB M) AND T = 5000K IS THE SOURCE OF DIFFICULTY IN INTERPRETATION OF CURRENT EXPERIMENTAL WORK ON HIGH-TEMPERATURE THERMODYNAMICS OF UO₂(SUB 2).

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
 THERMODYNAMICS + MOLTEN FUEL + URANIUM DIOXIDE + HIGH TEMPERATURE + UNITED KINGDOM

148843
 BRISCOE P + VAUGHAN GJ
 LNG/WATER VAPOUR EXPLOSIONS - ESTIMATES OF PRESSURES AND YIELDS
 UKAEA SAFETY & RELIABILITY DIRECTORATE, WARRINGTON, U.K.
 SRD R 131 + UKRSR-176 +, 20 PPS, 4 TABS, 7 FIGS, 27 REFS, OCT, 1978

CRITICALLY REVIEWS THE EXPERIMENTAL DATA ON VAPOUR EXPLOSIONS BETWEEN LNG AND WATER AND OTHER HEAVIER HYDROCARBONS AND WATER. THE SUPERHEAT LIMIT THEORY WHICH PURPORTS TO EXPLAIN THE EXPERIMENTS IS CONSIDERED, AND IS USED TO CALCULATE EXPLOSION PRESSURES AND YIELDS. THE THEORY IS SHOWN TO BE DEFICIENT IN SOME RESPECTS, AND A METHOD IS DESCRIBED OF CALCULATING UPPER LIMITS TO THE EXPLOSION YIELDS, DEPENDANT ONLY ON THERMODYNAMIC EFFECTS. THESE CALCULATIONS GIVE PRESSURES AND YIELDS HIGHER THAN THOSE CALCULATED BY THE SUPERHEAT LIMIT THEORY, BUT STILL MANY TIMES SMALLER THAN THE YIELDS FROM EXPLOSIVE BURNING.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
 LIQUID + GAS + WATER VAPOR + REVIEW + DATA COLLECTION + MATHEMATICAL TREATMENT + EXPLOSION

147602
 TIRION 4 - A COMPUTER PROGRAM FOR USE IN NUCLEAR SAFETY STUDIES

147602 *CONTINUED*
 UKAEA SAFETY & RELIABILITY DIRECTORATE, WARRINGTON, U.K.
 SRD-R-134 + UKRSR-231 +, 37 PPS, 10 FIGS, 41 REFS, NOV. 1978

TIRION 4 IS A COMPUTER PROGRAM WHICH MAY BE USED TO CALCULATE THE CONSEQUENCES OF RELEASING RADIOACTIVE MATERIAL TO THE ATMOSPHERE. IT IS AN IMPROVED VERSION OF AN EARLIER PROGRAM, TIRION 2. THIS PAPER DESCRIBES THE WAYS IN WHICH THE TWO PROGRAMS DIFFER AND THE IMPROVEMENTS THAT HAVE BEEN MADE. THESE INCLUDE A SYSTEMATIC STUDY OF PLUME RISE, SEVERAL REFINEMENTS OF THE METEOROLOGICAL MODEL EMPLOYED, A MUCH MORE FLEXIBLE APPROACH TO THE RELATIONSHIP BETWEEN DOSE AND CONSEQUENCE AND AN EXAMINATION OF THE MILK INGESTION PATHWAY.

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

RADIOACTIVITY RELEASE + PLUME BEHAVIOR + GAUSSIAN PLUME FORMULA + COMPUTER PROGRAM + WAKE EFFECT + CONCENTRATION + DOSE + IODINE + STRONTIUM + CESIUM + AIRBORNE RELEASE + AIRBORNE RELEASE

148846
 MACINNES DA

DO ELECTRONIC TRANSITIONS CONTRIBUTE TO THE THERMODYNAMICS OF CONDENSED UO₂? A REVIEW OF THE ARGUMENTS
 UKAEA SAFETY & RELIABILITY DIRECTORATE, WARRINGTON, U.K.
 SRD-R-151 + UKRSR-232 +, 13 PPS, 2 TABS, 5 FIGS, 20 REFS, MARCH 1979

RECENT ANALYSIS OF THE ROLE OF ELECTRONIC TRANSITIONS IN THE THERMOPHYSICAL PROPERTIES OF UO₂ IS SURVEYED. IT IS CONCLUDED TO BE HIGHLY LIKELY THAT THE ELECTRONS ON THE U⁴⁺ METAL ION PLAY A MAJOR ROLE IN BOTH THE SPECIFIC HEAT AND THERMAL CONDUCTIVITY, IN THAT THEY ARE PRIMARILY RESPONSIBLE FOR THE LARGE 'ANOMALOUS' INCREASE DISPLAYED BY EACH OF THESE QUANTITIES BETWEEN T = 1600K AND T(SUB M) = 3100K. THIS HAS IMPORTANT IMPLICATIONS FOR REACTOR ANALYSIS, SINCE TO OBTAIN THE REQUIRED DATA FOR MOLTEN FUEL ONE MUST EXTRAPOLATE EXISTING DATA THROUGH A WIDE RANGE IN TEMPERATURE, AND THE BEHAVIOR OF THE ELECTRONIC MECHANISMS MAY BE EXPECTED TO EXTRAPOLATE QUITE DIFFERENTLY FROM THAT OF THE MECHANISM IN CURRENT USE.

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

*THERMODYNAMICS + ELECTRON + *MOLTEN FUEL + *URANIUM DIOXIDE + DATA COLLECTION + ANALYTICAL TECHNIQUE + REVIEW + UNITED KINGDOM

7. U.K. FAST REACTOR SAFETY RESEARCH REPORTS RECEIVED BY NRC

THE FOLLOWING IS A LISTING OF REPORTS RECEIVED FROM THE U.K. DURING THE FIRST HALF OF 1979 UNDER THE TECHNICAL EXCHANGE AGREEMENT.

144781
PARSONS D
ASSESSMENTS OF RISK FOLLOWING THE INHALATION OF PLUTONIUM OXIDE USING OBSERVED LUNG CLEARANCE PATTERNS
OXFORD ATOMIC ENERGY ESTABLISHMENT, DORSET, U.K.
ATTN-X-1118 + UKRSR-175 + 20 PPS, 4 TABS, 3 FIGS, 16 REFS, OCT, 1977

DOSE COMMITMENTS AND RISK ESTIMATES FOR THE INHALATION OF PLUTONIUM OXIDE ARE CALCULATED USING THE
LUNG CLEARANCE PATTERNS OBSERVED AT AEC WINFRITH. THESE RISKS ARE COMPARED WITH PUBLISHED DATA
ON RISKS ARISING FROM A LUNG CLEARANCE BASED ON THE ICRP LUNG MODEL. (GEM)

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

DOSE + DOSE MEASUREMENT, INTERNAL + *LUNG + *PLUTONIUM + *RISK + UNITED KINGDOM + INHALATION

KEYWORD INDEX

A COLLECTION OF KEYWORDS IS USED TO DENOTE THE MAIN SAFETY RELATED POINTS COVERED IN EACH ARTICLE; THE FOLLOWING INDEX IS AN ALPHABETICAL LISTING OF THE KEYWORDS GIVING REFERENCES TO EACH ARTICLE WHICH WAS KEYED TO IT.

ACCIDENT
145756 P 25
ACCIDENT ANALYSIS
147103 P 3 143779 P 15 143745 P 19 145874 P 12
147467 P 18 147473 P 23 147483 P 18 147794 P 19
148670 P 18
ACCIDENT, CONSEQUENCES
148006 P 21
ACCIDENT, CORE DISRUPTIVE
145458 P 25
ACCIDENT, FUEL SLUMP
145458 P 25
ACCIDENT, LOSS OF COOLANT
143755 P 3 143911 P 1 143928 P 1 143929 P 4
145871 P 1 146735 P 9 143745 P 19 143806 P 12
143906 P 19 145156 P 22 145458 P 25 145636 P 16
143757 P 15 145874 P 12 146795 P 24 147473 P 23
148272 P 11 148297 P 10 145668 P 16 148671 P 21
148674 P 16 143891 P 27 143893 P 28 144529 P 27
144532 P 27 147502 P 29
ACCIDENT, LOSS OF FLOW
143925 P 15
ACCIDENT, LOSS OF POWER
143777 P 3
ACCIDENT, PROBABILITY OF
147021 P 5 147175 P 9
ACCIDENT, REACTIVITY
143974 P 28
ACCIDENT, STEAM LINE RUPTURE
148734 P 4
ACCUMULATORS
143966 P 14 145458 P 25
ACOUSTICS
144126 P 10 144198 P 11 145872 P 10 143805 P 26
144527 P 28 143776 P 30
ADSORPTION
144828 P 23
AEROSOL
143210 P 2
AIR
149000 P 7 144280 P 12 148670 P 18
AIR CLEANING
144828 P 23 145877 P 31
AIRBORNE RELEASE
147602 P 33 147602 P 33
AIRCRAFT
143138 P 5 145638 P 21
A. AMINUM
146870 P 8
ANALYTICAL MODEL
143339 P 6 143930 P 1 144415 P 6 145296 P 1
146871 P 8 148734 P 4 143211 P 19 144286 P 12
144582 P 11 145156 P 22 145638 P 21 145756 P 25
146456 P 13 147794 P 19 148006 P 21
ANALYTICAL TECHNIQUE
143756 P 2 143777 P 3 143778 P 4 143779 P 7
146872 P 4 147021 P 5 147101 P 7 147175 P 9
143924 P 24 144396 P 18 146793 P 13 146820 P 24
147779 P 24 147860 P 20 148368 P 13 148370 P 13
148672 P 17 148846 P 34 148847 P 31
ATMOSPHERIC DIFFUSION
146801 P 8
ATMOSPHERIC POLLUTION
146801 P 8
AVAILABILITY
145296 P 1
BELGIUM
144415 P 6
BLOODDOWN
143751 P 2 143755 P 3 143758 P 6 143914 P 2
143929 P 4 148734 P 4 148735 P 9 143894 P 14
143901 P 24 143905 P 15 145874 P 12 145875 P 10
148674 P 16 143891 P 27 144526 P 27 147502 P 29
BOILING
148844 P 29 143776 P 30 143907 P 30 148847 P 31
BUBBLE
143210 P 2 140447 P 32 145641 P 31 148847 P 31
BUCKLING
146870 P 8
CENTERLINE MELTING
143974 P 28
CESIUM
147602 P 33
CHARCOAL ADSORBER
144828 P 23
CHEMICAL REACTION
148673 P 20
CLADDING
148673 P 20
COBALT
143083 P 22
CODES AND STANDARDS
147795 P 32
COMBUSTION
143078 P 8 144415 P 6
COMPARISON
143751 P 2 143211 P 19 145874 P 12 147795 P 32
COMPARISON, THEORY AND EXPERIENCE
143751 P 2 146801 P 8 146805 P 17 148674 P 16
144526 P 27 148847 P 31
COMPARTMENT
143329 P 10 145675 P 10
COMPONENTS
143803 P 12 145155 P 22 147794 P 19 147817 P 23
143343 P 30 146483 P 33
COMPUTER PROGRAM
143210 P 2 143777 P 3 143778 P 4 145871 P 1
146869 P 8 148734 P 4 143211 P 19 143806 P 12
143905 P 15 143966 P 14 145636 P 16 146750 P 20
146804 P 20 146805 P 17 148272 P 11 148273 P 12
148668 P 16 148672 P 17 144523 P 27 144526 P 27
144527 P 28 144529 P 27 144532 P 27 147064 P 28
147502 P 29 140447 P 32 145640 P 31 145641 P 31
146483 P 33 147602 P 33 147795 P 32
COMPUTER PROGRAM, DIGITAL
144527 P 28 144529 P 27
CONCENTRATION
146801 P 8 147602 P 33
CONCRETE
143379 P 15 145155 P 22
CONCRETE, REINFORCED
145638 P 21
CONSTRUCTION
148669 P 9
CONTAINMENT
143806 P 12 145636 P 16 145846 P 10 145847 P 11
145875 P 10 146820 P 24 147176 P 17 147473 P 23
148272 P 11 148273 P 12
CONTAINMENT ANALYSIS
143329 P 10 143806 P 12 145636 P 16 146798 P 11
147176 P 17 148272 P 11 148273 P 12
CONTAINMENT ATMOSPHERE
145458 P 25
CONTAINMENT INSTRUMENTATION
145846 P 10 145847 P 11
CONTAINMENT INTEGRITY
143379 P 15 148675 P 23
CONTAINMENT LEAK MONITOR
145872 P 10
CONTAINMENT R AND D
146798 P 11 147176 P 17
CONTAINMENT STRUCTURE
148273 P 12
CONTAINMENT, ICE CONDENSER
143329 P 10
CONTAINMENT, LOW PRESSURE
148272 P 11
CONTAINMENT, PRESSURE SUPPRESSION
145757 P 15
CONTAMINATION
143383 P 4
CONTROL SYSTEM
143343 P 30
COOLANT CHEMISTRY
146869 P 8
COOLING SYSTEM, SECONDARY
148734 P 4
CORE
143891 P 27
CORE COMPONENTS
146804 P 20
CORE MELTDOWN
143379 P 15 145458 P 25 145756 P 25 147794 P 19
CORE REFLOODING

143758 P 6	143928 P 1	143903 P 14	143905 P 15		143383 P 4
146671 P 21	143891 P 27	143893 P 28	147502 P 29		FAULT TREE ANALYSIS
CORRELATION					143778 P 4
143751 P 2	143902 P 14				147779 P 24
CORROSION					147860 P 20
143083 P 22	144596 P 18	148673 P 20			FILM BOILING
CRACK					143902 P 14
148675 P 23					FILTER EFFICIENCY
CREEP					145877 P 31
145871 P 1	143089 P 19	148673 P 20			FILTER, HEPA
DAMAGE					145877 P 31
148670 P 18	148842 P 29				FILTERS
DATA COLLECTION					144828 P 23
148735 P 9	147779 P 24	144527 P 28	146343 P 33		FIRE
148846 P 34					143138 P 5
DATA PROCESSING					144415 P 6
143755 P 3	146798 P 11	148368 P 13			FISSION GAS RELEASE
DECOMMISSIONING					143383 P 4
146795 P 24	147817 P 23				FISSION PRODUCT RELEASE
DEFORMATION					144595 P 5
143875 P 3	145871 P 1	146871 P 8	146372 P 4		148675 P 23
145804 P 20					144523 P 27
DESIGN CRITERIA					FISSION PRODUCT RETENTION
148669 P 9	148734 P 4				147794 P 19
DIFFUSION					FISSION PRODUCT TRANSPORT
144286 P 12	144582 P 11	148672 P 17			144595 P 5
DISCHARGE					144596 P 18
146483 P 33					FLAW
DISPERSION					120343 P 31
146801 P 8					FLOW
DISTRIBUTION					143752 P 2
146795 P 24					143755 P 3
DNB					143211 P 19
143894 P 14					146486 P 13
DOSE					146750 P 20
148006 P 21	147602 P 33	145281 P 35			FLOW BLOCKAGE
DOSE MEASUREMENT, INTERNAL					143911 P 1
145281 P 35					143906 P 19
DROPLET					143906 P 19
143928 P 1					148296 P 23
EARTHQUAKE					143907 P 30
143138 P 5					FLOW STABILITY
ELECTRIC POWER					143753 P 2
143894 P 14					FLOW THEORY AND EXPERIMENTS
ELECTRIC POWER, AUXILIARY					143753 P 2
143777 P 3					148668 P 16
ELECTRICAL CONDUCTION					144532 P 27
143773 P 33					147064 P 28
ELECTRON					FLOW, ANNULAR
148846 P 34					146844 P 29
EMERGENCY COOLING					FLOW, CRITICAL
143903 P 14	143905 P 15				143900 P 7
EMERGENCY COOLING SYSTEM					147502 P 29
143928 P 1	143906 P 19	143891 P 27	144532 P 27		FLOW, MIXING
ENVIRONMENT					143901 P 24
144280 P 12	147794 P 19				146750 P 20
EQUATION					FLOW, TWO PHASE
147101 P 7	148672 P 17				143751 P 2
EQUATION OF STATE					143914 P 2
146483 P 33					143928 P 1
ERROR ANALYSIS					143930 P 1
148368 P 13	148370 P 13				144197 P 25
EXAMINATION					148296 P 23
146799 P 13					143893 P 28
EXCURSION, LARGE					148844 P 29
148670 P 18					146483 P 33
EXPERIMENT					FLUX DISTRIBUTION
143078 P 8	143755 P 3	148735 P 9	143745 P 19		147795 P 32
143806 P 12	143901 P 24	145872 P 10	146486 P 13		FRACTURE TOUGHNESS
146488 P 14	146798 P 11	146799 P 13	148671 P 21		143805 P 26
148842 P 29	148844 P 29	148848 P 32			FRANCE
EXPLOSION					143078 P 8
143078 P 8	143138 P 5	144415 P 6	144280 P 12		143339 P 6
147817 P 23	143910 P 32	148843 P 33			143383 P 4
FABRICATION					143756 P 2
148669 P 9					143758 P 6
FAILURE					143779 P 7
143870 P 5	146869 P 8	143089 P 19	145756 P 25		143870 P 5
FAILURE MODE ANALYSIS					144595 P 5
143339 P 6	145296 P 1				145296 P 1
FAILURE, CLADDING					145871 P 1
143974 P 28	144523 P 27				146801 P 8
FAILURE, COMMON MODE					146872 P 4
143777 P 3	143778 P 4	143779 P 7			146871 P 8
FAILURE, EQUIPMENT					146872 P 4
143339 P 6					146873 P 9
FAILURE, FABRICATION ERROR					FUEL COOLANT INTERACTION
143383 P 4					148842 P 29
FAILURE, FUEL ELEMENT					143910 P 32
143383 P 4	144523 P 27				148848 P 32
FAILURE, INHERENT					FUEL CYCLE
					147102 P 15
					FUEL ELEMENTS
					146488 P 14
					FUEL REPROCESSING
					147103 P 3
					147794 P 19
					FUEL ROD
					145871 P 1
					143745 P 19
					143894 P 14
					145156 P 22
					146488 P 14
					146796 P 13
					148296 P 23
					148673 P 20
					146675 P 23
					143974 P 28
					147064 P 28
					148842 P 29
					FUEL STORAGE
					147103 P 3
					FUEL SWELLING
					148296 P 23
					FUEL, NUCLEAR
					148673 P 20
					143343 P 30
					GAS
					143078 P 8
					143210 P 2
					148843 P 33
					GAUSSIAN PLUME FORMULA
					147602 P 33
					GERMANY
					143083 P 22
					143211 P 19
					143329 P 10
					143379 P 15
					143745 P 19
					143768 P 21
					143803 P 12
					143806 P 12
					143901 P 24
					143902 P 14
					143903 P 14
					143904 P 24
					143905 P 15
					143966 P 14
					144158 P 22
					144196 P 10
					144197 P 25
					144198 P 11
					144280 P 12
					144286 P 12
					144582 P 11
					144596 P 18
					144854 P 16
					145155 P 22
					145156 P 22
					145165 P 16
					145458 P 25
					145638 P 21
					145756 P 25
					145757 P 15
					145872 P 10
					145874 P 12
					145875 P 10
					146486 P 13
					146488 P 14
					146750 P 20
					146793 P 13
					146795 P 24
					146796 P 13

146798 P 11	146799 P 13	146804 P 20	146825 P 17				
146813 P 17	146820 P 24	147102 P 15	147175 P 17				
147467 P 15	147473 P 23	147463 P 18	147794 P 11				
147817 P 23	148035 P 21	148363 P 18	148358 P 13				
148369 P 17	148370 P 13	148568 P 16	148670 P 18				
148671 P 21	148672 P 17	148674 P 16	148675 P 23				
143805 P 26							
HEAT FLUX, BURNOUT							
148844 P 29							
HEAT FLUX, CRITICAL							
143902 P 14	148844 P 29						
HEAT FLUX, DRYOUT							
143928 P 1	143901 P 24	145640 P 31					
HEAT GENERATION, INTERNAL							
143974 P 28							
HEAT TRANSFER							
143758 P 6	149000 P 7	148273 P 12	148297 P 10				
HEAT TRANSFER ANALYSIS							
143903 P 14	146805 P 17	140447 P 32					
HEAT TRANSFER COEFFICIENT							
149000 P 7	143902 P 14	143297 P 10					
HEAT TRANSFER EXPERIMENT							
143902 P 14	143903 P 14	148297 P 10					
HEAT TRANSFER, BOILING							
143902 P 14							
HEAT TRANSFER, TWO PHASE							
149000 P 7							
HEATERS							
143894 P 14							
HIGH TEMPERATURE							
145874 P 12	148845 P 33						
HYDRAULIC ANALYSIS							
148734 P 4	143904 P 24	146805 P 17					
HYDRAULIC EFFECT							
146804 P 20	146805 P 17						
HYDRAULIC EXPERIMENT							
143903 P 14	143905 P 15	147064 P 28					
HYDRODYNAMIC ANALYSIS							
143756 P 2	146750 P 20						
HYDROGEN							
144415 P 6	144158 P 22	144280 P 12	144286 P 12				
144582 P 11	147473 P 23						
IMPACT PROPERTY							
143803 P 12							
IMPACT SHOCK							
143138 P 5	145638 P 21						
IN PILE EXPERIMENT							
144523 P 27							
INCIDENT COMPILATION							
143339 P 6							
INDUSTRY, NUCLEAR							
148669 P 9	148369 P 17						
INHALATION							
145281 P 35							
INSTALLATION							
148659 P 9							
INSTRUMENT, DENSITY							
144197 P 25							
INSTRUMENT, FUEL SCANNING							
145796 P 13							
INSTRUMENT, PRESSURE							
145846 P 10	145847 P 11	146488 P 14					
INSTRUMENT, TEMPERATURE							
145847 P 11							
INTERNATIONAL							
147795 P 32							
IODINE							
144523 P 27	147602 P 33						
IRON							
143083 P 22							
IRRADIATION TESTING							
148675 P 23							
JAPAN							
144854 P 16	146813 P 17	143974 P 28	144523 P 27				
144526 P 27	144527 P 28	144529 P 27	144532 P 27				
147064 P 28	147502 P 29	148842 P 29	148844 P 29				
143343 P 30							
LASER							
120343 P 31							
LEAK DETECTION							
145872 P 10							
LICENSED OPERATOR							
143768 P 21							
LICENSING PROCESS							
147021 P 5	147175 P 9						
LIQUID							
148843 P 33							
LOFT (S-RR)							
143905 P 15	148668 P 16	148674 P 16					
LUNG							
145281 P 35							
MAIN COOLING SYSTEM							
143383 P 4	143083 P 22	147817 P 23					
MASS							
144197 P 25							
MASS TRANSFER							
143900 P 7	143930 P 1	144596 P 18	143907 P 30				
MATHEMATICAL TREATMENT							
143753 P 2	143777 P 3	143875 P 3	147101 P 7				
143379 P 15	143904 P 24	146793 P 13	147860 P 20				
148843 P 33							
MATHEMATICS, DIFFERENCE EQUATION							
147101 P 7							
MEASUREMENT							
143755 P 3	143914 P 2	143929 P 4	144158 P 22				
144197 P 25	144198 P 11	146486 P 13	146488 P 14				
147473 P 23	143805 P 26	143776 P 30	143910 P 32				
148847 P 31	148848 P 32						
MEASUREMENT, NOISE							
143776 P 30							
MEASUREMENT, TEMPERATURE							
145846 P 10	145847 P 11						
METAL WATER REACTION							
144529 P 27							
METEOROLOGY							
148006 P 21							
MICROSTRUCTURE							
143773 P 33							
MISSILE GENERATION AND PROTECTION							
143803 P 12	145638 P 21						
MODEL							
143751 P 2	143758 P 6						
MODEL TESTING							
148297 P 10							
MOLTEN FUEL							
148842 P 29	148845 P 33	148846 P 34					
MONITOR							
145872 P 10							
N-POWER, SAFETY OF							
148363 P 18							
NEUTRON							
143914 P 2							
NOISE ANALYSIS							
144196 P 10	144198 P 11						
NOZZLE							
143211 P 19							
NUCLEATE BOILING							
143894 P 14							
NUMERICAL METHOD							
143753 P 2	143756 P 2	143930 P 1	147795 P 32				
OFF GAS							
144158 P 22	147794 P 19						
ON SITE							
147103 P 3							
OPERATOR ACTION							
143768 P 21							
OUT OF PILE EXPERIMENT							
148673 P 20	143907 P 30						
OXIDE							
148673 P 20							
OXYGEN							
144158 P 22	144596 P 18						
PHASE CHANGE							
143904 P 24							
PIPES AND PIPE FITTINGS							
145155 P 22	147817 P 23	140447 P 32					
PLASTICITY							
143875 P 3							
PLENUM							
148671 P 21							
PLUME BEHAVIOR							
147602 P 33							
PLUTONIUM							
148363 P 18	145281 P 35						
POLLUTION							
146801 P 8							
POWER PLANT, NUCLEAR							
143138 P 5	143339 P 6	148669 P 9	143768 P 21				
147176 P 17	147817 P 23						
PRESSURE DROP							
143929 P 4	143211 P 19	143893 P 28					
PRESSURE PULSE							
144415 P 6	145757 P 15						
PRESSURE TRANSIENT							
149000 P 7	145636 P 16	145757 P 15	145875 P 10				
146488 P 14	148668 P 16	143893 P 28	144532 P 27				
140447 P 32							
PRESSURE VESSELS							
143870 P 5	146869 P 8	146870 P 8	144196 P 10				
145756 P 25	147817 P 23	120343 P 31					
PRESSURE, INTERNAL							
143755 P 3	146870 P 8	143329 P 10					

140486 P 13
UNITED KINGDOM
140487 P 32 140640 P 31 140641 P 31 140845 P 33
140886 P 34 141087 P 31 140890 P 32 145201 P 33
UNITED STATES
144054 P 16 146813 P 17
URANIUM DIOXIDE
141773 P 33 140045 P 33 140546 P 34
VALVES
141339 P 6
VAPOR PRESSURE
141078 P 3 141327 P 20
VOID
141751 P 2 140847 P 31
VOID FRACTION
141751 P 2 141755 P 3 143900 P 7 143914 P 2
143929 P 4 143930 P 1 143991 P 20 145641 P 31
WAKE EFFECT
147602 P 33

WASTE MANAGEMENT
146795 P 24
WASTE TREATMENT, EQUIPMENT
144828 P 23
WASTE TREATMENT, GAS
144158 P 22
WATER
148670 P 18 140447 P 32
WATER VAPOR
143211 P 19 140043 P 33
WETTING
143993 P 14
WIND PROFILE
140006 P 21
WIND STATISTICS
148006 P 21
ZIRCALOY
143089 P 19 145156 P 22 148673 P 20

AUTHOR INDEX

FOLLOWING IS A LIST OF AUTHORS WHOSE DOCUMENTS
HAVE BEEN ABSTRACTED IN THIS PUBLICATION

ABRAHAMSON D	143761 P 2				
ALONICK DC	143906 P 21				
ALIX M	146873 P 3				
ALIXA F	147932 P 29				
ANDRIAN SA	143358 P 13				
AYER A	143306 P 21				
BEHRENS K	144250 P 12				
BERNAT M	146870 P 18				
BESANZ R	143756 P 25				
BETH A	143777 P 3	143778 P 4			
BLOCK M	143089 P 19	143745 P 19			
BODNETON M	143755 P 1				
BODALE J	146872 P 4				
BORGINS M	143929 P 4				
BORGESANI P	143339 P 6				
BRACHT K	143379 P 15				
BRISQDE F	146843 P 33				
BRITTAIN I	146840 P 31				
BROSSARD J	144415 P 6				
BROUARD D	146872 P 4				
BROVIERE M	143756 P 2	147101 P 7			
BRUCE WH	146840 P 31				
BURE M	146868 P 16				
CALDAROLA L	147863 P 20				
CARNINO A	143777 P 3	143778 P 4	143779 P 7	143870 P 5	
CAHNETTE P	143875 P 3				
CHAGROT M	146871 P 1				
CHEISSOUX JL	143875 P 3				
CHENEHAULT P	144595 P 5				
CLASS G	143745 P 19				
COGNE F	147021 P 5				
COURTAUD M	144758 P 6				
CRUIX JM	149000 P 7				
DASCALAKIS J	143930 P 1				
DAUBLEBSKY P	146838 P 21				
DE M JELMS AN	143910 P 32				
DEUSTER G	146799 P 13				
DIEHL G	146870 P 18				
DOBEMANN G	146793 P 13				
DORNER H	145155 P 22				
DRECHER HP	146874 P 12				
DUCHEMIN B	143778 P 4				
DUCO J					
	143138 P 5	144415 P 6			
	DUPRESNE J	143870 P 5	146869 P 8		
	DUALPORCE TA	143910 P 32	146840 P 32		
	DUER G	146795 P 24			
	EISENBLATTER J	144196 P 10			
	ENDERLE G	146750 P 20			
	ENGEL H	144158 P 22	144628 P 23	147473 P 23	
	ERBACHER F	143745 P 19	143906 P 19		
	FIEGE A	143745 P 19			
	FIRNHABER M	146874 P 16			
	FISCHER T	143805 P 26			
	FOURCADE P	143339 P 6			
	FRANK R	143929 P 4			
	FRASER DC	146877 P 31			
	FUCHS A	143082 P 22			
	GARCIA JL	143875 P 3			
	GASCH A	146795 P 24			
	GEORGIN JP	143339 P 6	143777 P 3		
	GIRARD P	143078 P 8			
	GOBERT T	143138 P 5	144415 P 6		
	GRANDCTIF M	143753 P 2			
	GREEN C	146840 P 31			
	GROHS B	146368 P 13			
	HAGA K	143776 P 30	143907 P 30		
	HALL SF	146483 P 33			
	HALM G	146870 P 18			
	HALSALL WJ	147795 P 32			
	HARRER A	144595 P 5			
	HELLINGS G	146836 P 16			
	HILDENBRAND G	146875 P 23			
	HIRANO K	146842 P 29			
	HOFMANN P	146873 P 20			
	HOLMES JA	146841 P 31			
	HORI M	143907 P 30			
	HUNEAU M	143078 P 8			
	IBAMURA T	146844 P 29			
	JAGER EH	147817 P 23			
	JAKOBS E	146799 P 13			
	JANVIER JC	143383 P 4			
	JAX P	144196 P 11	145872 P 10		
	JEANDET CH	143900 P 7			
	JOST H	144196 P 10			
	KAISER F	146795 P 24			

KARNATH G			
143883 P 22			
KEOZIOR F			
143211 P 19			
KIKUCHI O			
147964 P 28			
KIKUCHI Y			
143776 P 30	143907 P 33		
KITANO T			
143974 P 28			
KNODLER O			
145156 P 22			
KODAYASHI K			
144527 P 28	144529 P 27		
KOIZUMI Y			
147064 P 28			
KORBER H			
148458 P 25	145756 P 25		
KOSFELD W			
144197 P 25			
KURKA G			
144595 P 5			
KURUYANAGI T			
148844 P 29			
LANGHEIM H			
143803 P 12			
LE BERRE F			
143756 P 2			
LE COQ G			
143911 P 1			
LEFORT G			
147103 P 3			
LEYER JC			
143078 P 8			
LIEGEOLIS A			
149000 P 7			
LIESCH KJ			
148668 P 16			
LIDRY M			
143779 P 7			
LORENZ H			
144198 P 11			
LOTTERMOSEER J			
146486 P 13			
LUCIZ AC			
143870 P 5			
MACINNES GA			
143773 P 33	148845 P 33	148846 P 34	
MAIGNE JP			
146801 P 8			
MANGFELD G			
145636 P 16			
MARTIN D			
140447 P 32			
MARTIN DJV			
120343 P 31			
MAY H			
147102 P 15			
MAYINGER F			
143904 P 24			
MENHERT G			
147779 P 24			
MENNESSIER D			
143751 P 2			
MICHAEL I			
144596 P 18			
MOCHIZUKI Y			
144532 P 27			
MOSINGER H			
143211 P 19			
MULLER K			
143805 P 26			
MULLER W			
148370 P 13			
MULLER-CHRISTIANSE			
148363 P 18			
MURAO Y			
143893 P 28			
NAGAI H			
144523 P 27			
NAMY D			
143779 P 7	147175 P 9		
NASH G			
148847 P 31			
NOBOTHY B			
145638 P 21			
OCHS J			
144198 P 11			
OHNISHI N			
143974 P 28			
OPFER HD			
147817 P 23			
OZAKI Y			
143776 P 30			
PANA F			
145757 P 15			
PINET B			
143900 P 7			
PITIS JH			
146805 P 17			
PUINTNER W			
143894 P 14			
PORRACCHIA A			
143210 P 2			
PORTE R			
146801 P 8			
PUMPH-WESTERHEIDE P			
144197 P 25			
PUIT JC			
147103 P 3			
QUENEE R			
143779 P 7			
QUERO J			
143870 P 5			
RABASSE C			
143078 P 8			
RANSDEK D			
145281 P 35			
RAYMOND P			
143911 P 1	143928 P 1		
REUCREUX M			
143758 P 6			
RICQUE R			
143929 P 4			
RIEGEL B			
143914 P 2	148735 P 9		
RIMMER W			
143910 P 32	148848 P 32		
RINGER F			
143894 P 14			
ROCHE H			
146870 P 8	146871 P 8	146872 P 4	
RODDER P			
145874 P 12			
ROUSSEAU JC			
143914 P 2			
ROY C			
143339 P 6			
SAHM A			
144196 P 10			
SASAKI S			
144526 P 27	144527 P 28	147502 P 29	
SATO K			
144529 P 27	147502 P 29		
SAUER A			
143768 P 21			
SCHAEFER A			
148672 P 17			
SCHALL M			
146798 P 11	148272 P 11		
SCHMIDT A			
143966 P 14			
SCHNEIDER H			
144280 P 12			
SCHUMANN U			
146804 P 20			
SCHUNCKLER W			
148006 P 21			
SCHUSTER E			
143083 P 22			
SCHWEICKERT A			
148671 P 21			
SCHWEICKERT H			
146296 P 23			
SCHWINGES B			
145757 P 15			
SIGNORET JP			
143777 P 3	145296 P 1		
SIRETA X			
148734 P 4			
SOBAJIMA W			
144532 P 27			
SODA K			
147064 P 28			
SUDO Y			
143893 P 28			
SUREAU H			
143758 P 6			
SUZUKI M			
144532 P 27			

TAKEDA T	143904 P 24
144523 P 27	147042 P 29
TAKUYO O	VOIN R
144403 P 5	148669 P 9
TAKIWA S	VOJTEK I
143974 P 28	143902 P 14
THIAGUEBO J	VON KLIT R
143758 P 6	144152 P 10
ULLICH R	WANDA AU
143905 P 13	143902 P 14
UNGE R	WALTE F
145458 P 25	148370 P 13
UTANI M	WARNEUNDE H
143907 P 30	147102 P 15
VACHAN GJ	WASCHIES E
144043 P 33	148426 P 13
VITHELMANN R	WOLLESEN M
145458 P 25	148363 P 18
VITCOZAZ HJ	ZENNER P
	148422 P 13

PERMUTED TITLE INDEX

THIS INDEX IS ONE IN WHICH THE REPORT TITLES ARE PERMUTED AROUND SIGNIFICANT WORDS WHICH APPEAR IN THE TITLE. THE INDEX WORDS ARE ARRANGED ALPHABETICALLY IN A COLUMN IN THE CENTER OF THE PAGE WITH THE TITLE PERMUTED AROUND THEM. (X INDICATES BEGINNING OF TITLE, * INDICATES END OF TITLE.)

- HYDRAULIC BEHAVIOR IN THE INITIAL BLOWDOWN PHASE, PARIS
#INVESTIGATION OF THE VARIOUS PHASES OF THE CORE MELT
IN GERMAN)*#PHASE ELEMENT BEHAVIOR DURING A LOSS-OF-COOLANT
#COMPARATIVE INVESTIGATIONS OF A COOLING SYSTEM BLOWDOWN
INVESTIGATIONS ON THE LWR FUEL ROD BEHAVIOR UNDER
AND THE #INVESTIGATION OF THE FIRST PHASE OF THE CORE MELT
LIGHT WATER REACTOR CONTAINMENT FOLLOWING A LOSS-OF-COOLANT
CONCRETE - FAILURE PHASE OF THE HYDROTHERMAL CORE MELTDOWN
RATES OF THE VALVES OF ST. LAURENT DES EAUX POWER PLANT
EXPERIMENT IN A 19-PEN BUNDLE (IN ENGLISH)*
- JAPANESE)* #AN ANALYSIS OF THE
THERMO-HYDRAULIC STUDIES FOR THE DEVELOPMENT OF SAFETY
#IN-SITU TESTING OF HIGH EFFICIENCY FILTERS AT
OF THE VARIOUS PHASES OF THE CORE MELT ACCIDENT AFTER THE
PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT
PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT
PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT
AND AIRCRAFT #PROTECTION OF NUCLEAR POWER PLANTS (NPPS)
#STUDY OF POLLUTANT DISPERSION IN WATER AND
#EXAMINATION OF 3 HSSC PLATES RUPTURED IN
STAINLESS STEEL TEST SECTION #STUDY ON THE CONDENSATION OF
FOR HIGH TEMPERATURE GAS REACTORS, PHASE II: WATER INGRESS,
ENVIRONMENT (EN #INITIATION OF DETECTION OF HYDROGEN-
THE OVERPRESSURE GENERATED BY THE DETONATION OF SPHERICAL
AGAINST EXTERNAL EVENTS: EARTHQUAKES, AIR EXPLOSIONS AND
#AN ANALYSIS OF LOFT L1-2 EXPERIMENT BY
BLOWDOWN ANALYSIS (IN ENGLISH)*
- #ERROR ANALYSIS OF THE
THE SECONDARY SIDE OF A STEAM GENERATOR (IN FRENCH)*
A COMPUTER PROGRAM FOR PRESSURIZED WATER REACTOR BLOWDOWN
SYSTEMS AND DESCRIPTION OF EQUATIONS FOR HYDRODYNAMIC
THE APPARENT INFLUENCE OF THESE OCCURRENCES IN FAULT TREE
IMPROVEMENT IN THE MEASUREMENT TECHNIQUE OF SONIC-EMISSION
WATER INGRESS, AIR INGRESS, #COMPREHENSIVE DAMAGE
CYCLE TORSION (IN FRENCH)* #SHORT
ENGLISH)*
DE ROSA-11 EXPERIMENTS FOR COLD LEG BREAK RUNS 413 AND
IN ENGLISH)* #AN
413 AND #ANALYSIS OF LOCA EXPERIMENTS WITH RELAP4J CODE
PHENOMENA (IN ENGLISH & JAPANESE)*
- RS 50 (MODEL CONTAINMENT) PART 2 (IN GERMAN)*
FUEL ROD - EFFECT OF THERMAL CYCLING (IN ENGLISH)*
TEST CALCULATIONS (IN GERMAN)* #RELAP-4/GRS
#REFLUX-GRS
#FAULT TREE
- III AT JULY 1976 - A PROGRAM FOR TRANSIENT THERMAL-HYDRAULIC
EXPERIMENT ON LEAKAGE MONITORING USING SONIC EMISSION
DURING A LOCA (IN ENGLISH)*#ASCOT-11: A COMPUTER PROGRAM FOR
#FLOW REDUCTION TRANSIENT BURNOUT IN AN
#UNCERTAINTY OF THE FAILURE RATE OF COMPONENTS AND THE
#A SIMPLE HOMOGENEOUS EQUILIBRIUM CRITICAL DISCHARGE MODEL
MEASUREMENTS MADE IN LINEN DMR)* #AN
RELIABILITY OF THE PROTECTION SYSTEM OF THE #A FIRST
REPORT ON FAST BREEDER REACTOR DEVELOPMENT IN JAPAN,
NONNEGLECTIBLE TEST DURATION, TEST EFFICIENCY #SYSTEMS WHICH
RESEARCH AND TECHNOLOGY CONCERNING RESEARCH PROJECTS IN THE
#REPORT ON THE RESEARCH PROGRAM SPONSORED BY BRFI IN THE
TRANSIENT OF A NUCLEAR POWER PLANT IN THE LARGE EXPERIMENT
FLOW OF A COMPRESSIBLE MEDIUM IN COUPLED RECTANGULAR
TO THE THERMODYNAMICS OF CONDENSED UCL2? A REVIEW OF THE
IN THE #EXPERIMENTAL RESEARCH ON SINGLE AND MULTIPLE TUBE
FOR GENERATING STEAM TABLES HAVING PRESSURE AND DENSITY
HYDRAULIC BEHAVIOR IN A PWR CORE DURING A LOCA (IN ENGLISH)*
GERMAN)* #THE ESSENTIAL SAFETY
OXIDE USING OBSERVED LUNG CLEARANCE PATTERNS)* #RELIABILITY
A MASS-DENSITY METHOD FOR TRANSIENT TWO PHASE STATE USING
- A, D, & C (IN GERMAN)* THEORETICAL RESEARCH ON THE THERMAL
ACCIDENT AFTER THE AFTER FAILURE OF THE CORE SUPPORT
ACCIDENT AND INTERACTION WITH THE EMERGENCY CORE COOLING (E
ACCIDENT AND THE SUBSEQUENT THERMAL TRANSIENT IN LIGHT-
ACCIDENT CONDITIONS (IN GERMAN)* AND EXPERIMENTAL
ACCIDENT WITH RELSIN-1 IN A PWR AND A DWR STANDARD PLANT,
ACCIDENT, QUICK LOCK REPORT 1 (IN GERMAN)* IN A
ACCIDENT, QUICK LOCK REPORT 2 (IN GERMAN)* IN A
ACCIDENT: CALCULATION TO IDENTIFY THE INFLUENCE OF VARIOUS
ACCORDING TO INFLUENTIAL PARAMETERS)* #MODEL OF THE FAILURE
#ACOUSTIC NOISES WITH LOSS-OF-FLOW SODIUM BOILING
#ACTIVATED CORROSION PRODUCTS IN LWR LOOPS (IN GERMAN)*
ADDITIONAL FISSION PRODUCT RELEASE PHENOMENA (IN ENGLISH &
ADVANCED CODE FOR PWR (IN FRENCH)* #FRENCH
AEE W/AFHIT)*
- AFTER FAILURE OF THE CORE SUPPORT STRUCTURE DUE TO THE
AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-
AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-
AFTER RUPTURE OF THE PRIMARY COOLING CIRCUITS IN WATER-
AFTER THE AFTER FAILURE OF THE CORE SUPPORT STRUCTURE DUE
AGAINST EXTERNAL EVENTS: EARTHQUAKES, FIRES, EXPLOSIONS
AIR (IN ENGLISH)*
AIR (IN GERMAN)*
AIR AND STEAM MIXTURES IN TRANSIENT CONDITIONS ON A
AIR INGRESS, REACTIVITY OCCURRENCES (IN GERMAN)* ANALYSIS
AIR MIXTURES AND PROPAGATION OF SHOCK WAVES IN THE
AIR-HYDROGEN GASEOUS MIXTURES (IN ENGLISH)* STUDY OF
AIRCRAFT CRASHES (IN ENGLISH)* NUCLEAR POWER PLANTS (NPPS)
ALARM-PI COMPUTER CODE (IN ENGLISH)*
ALARM-PI: A COMPUTER PROGRAM FOR PRESSURIZED WATER REACTOR
AMPLITUDE CURVE, FINAL REPORT (IN GERMAN)*
ANA EXPERIMENTAL AND THEORETICAL STUDY OF THE BLOWDOWN OF
ANALYSIS (IN ENGLISH)* #ALARM-PI
ANALYSIS (IN GERMAN)* #STUDY OF THE STABILITY OF VARIOUS
ANALYSIS (IN GERMAN)*#OF THE FAILURE RATE OF COMPONENTS AND
ANALYSIS (ISER) (IN GERMAN)* #
ANALYSIS FOR HIGH TEMPERATURE GAS REACTORS, PHASE II,
ANALYSIS OF A PROGRESSIVE DISTORTION PROBLEM (TENSION AND
ANALYSIS OF BOILING-WATER REACTOR STEAM CHUGGING (IN
ANALYSIS OF LOCA EXPERIMENTS WITH RELAP4J CODE (ANALYSIS
ANALYSIS OF LOFT L1-2 EXPERIMENT BY ALARM-PI COMPUTER CODE
ANALYSIS OF ROSA-11 EXPERIMENTS FOR COLD LEG BREAK RUNS
ANALYSIS OF THE ADDITIONAL FISSION PRODUCT RELEASE
ANALYSIS OF THE AMPLITUDE CURVE, FINAL REPORT (IN GERMAN)*
ANALYSIS OF THE D SERIES EXPERIMENTS OF RESEARCH PROJECT
ANALYSIS OF THE FISSION PRODUCT RELEASE FROM A DEPLETED
ANALYSIS OF THE NONNUCLEAR LOFT-TESTS L1-4 (PRE AND POST
ANALYSIS OF THE REFLUX EXPERIMENTS (RS 62) (IN ENGLISH)*
ANALYSIS WITH MULTISTATE COMPONENTS (IN ENGLISH)*
ANALYSIS)* #THE STATUS OF RELAP-4X WR
ANALYSIS: EXPANDED INSTRUMENTATION AND EVALUATION PROGRAM (E
ANALYZING THE THERMO-HYDRAULIC BEHAVIOR IN A PWR CORE
ANNULAR TEST SECTION (IN JAPANESE & ENGLISH)*
APPARENT INFLUENCE OF THESE OCCURRENCES IN FAULT TREE
APPLIED TO MULTI-COMPONENT, TWO-PHASE SYSTEMS - THE
APPRAISAL OF SUBCOOLED BOILING AND SLIP RATIO FROM
APPROACH OF THE HAZE EVENT PROBLEM BY THE STUDY OF THE
APRIL-JUNE 1978)* #PROGRESS
ARE UNCONNECTED AND WAITING FOR PERIODIC TESTING-
AREA OF REACTOR SAFETY REPORTING PERIOD OCTOBER-DECEMBER
AREA OF REACTOR SAFETY, JULY 1-SEPTEMBER 30, 1978 (IN
AREA OF THE GRS (IN GERMAN)* TUBE ARRAYS IN THE PRESSURE
AREAS (IN GERMAN)* THE CALCULATION OF THE 1D-DIMENSIONAL
ARGUMENTS)* #DU ELECTRONIC TRANSITIONS CONTRIBUTE
ARRAYS IN THE PRESSURE TRANSIENT OF A NUCLEAR POWER PLANT
AS THE INDEPENDENT VARIABLES (IN ENGLISH & JAPANESE)*
ASCOT-11: A COMPUTER PROGRAM FOR ANALYZING THE THERMO-
ASPECTS OF A CONFINED NUCLEAR FUEL CYCLE (IN GERMAN)*
ASSESSMENT OF THE SECONDARY CONTAINMENT OF A PWR (IN
ASSESSMENTS OF RISK FOLLOWING THE INHALATION OF PLUTONIUM
ASSURANCE BY FRACATURE (IN ENGLISH)*
ATOMIC RESONANCE (IN GERMAN)* #DEVELOPMENT OF
- 143901 P 24
143756 P 25
143906 P 19
145074 P 12
143745 P 19
142458 P 15
144082 P 11
144286 P 12
143379 P 10
143339 P 6
143776 P 20
143083 P 22
144023 P 27
143758 P 0
145877 P 21
143756 P 25
145046 P 10
140273 P 12
145047 P 11
145736 P 25
143138 P 9
146001 P 8
146790 P 13
149000 P 7
140670 P 18
144280 P 12
144415 P 6
143138 P 5
144526 P 27
147502 P 29
146370 P 13
146734 P 0
147020 P 29
143756 P 24
147779 P 2
144198 P 11
146671 P 8
140805 P 17
144032 P 27
144526 P 27
144532 P 27
144523 P 27
146370 P 13
146790 P 11
144505 P 5
143903 P 14
147800 P 20
145640 P 31
145872 P 10
144525 P 27
140944 P 29
147779 P 24
144483 P 13
140870 P 31
143779 P 7
143343 P 20
145290 P 1
147483 P 18
145105 P 16
146488 P 18
146750 P 20
140846 P 24
144527 P 25
144529 P 27
147102 P 15
140820 P 24
145281 P 25
146005 P 5
144197 P 25

OF FLOW USING ULTRASONIC LABORATORY INVESTIGATIONS FOR THE PLUTONIUM (IN GERMAN)	ATTAINMENT OF INTERPRETATIONAL MODELS FOR THE ESTIMATION OF ATTITUDES ON QUESTIONS PERTAINING TO NUCLEAR ENERGY	140486 P 13
EXPERIMENTAL TESTS ON MATECH OF JSA	AUSTRIAN STEEL AT ROOM TEMPERATURE (IN FRENCH)	140303 P 10 140874 P 8
HYDRAULIC BEHAVIOR IN THE INITIAL BLOWDOWN PHASE, PARTS A: EFFECT OF HEAT GENERATION PROFILE IN RELATION TO FUEL FAILURE WITH THE EMERGENCY CORE COOLING (IN GERMAN) #FUEL ELEMENT	5, 6 & C (IN GERMAN) THEORETICAL RESEARCH ON THE THERMAL BEHAVIOR ENRICHMENT PARAMETER TEST IN NSRR (IN ENGLISH & BEHAVIOR DURING A LOSS-OF-COOLANT ACCIDENT AND INTERACTION BEHAVIOR IN A PWR CORE DURING A LOCA (IN ENGLISH) #ASCOT	143901 P 14 143974 P 20 143900 P 15 144529 P 27 143901 P 24
1) A COMPUTER PROGRAM FOR ANALYZING THE THERMO-HYDRAULIC AND THERMICAL RESEARCH ON THE THERMAL HYDRAULIC BEHAVIOR; A CODE DESCRIBING THE THERMAL AND MECHANICAL ASPECT OF FRAGMENTS AND PROJECTILES OF DIFFERENT MASS AND EXPERIMENTAL INVESTIGATIONS ON THE LWR FUEL ROD #PRELIMINARY THEORETICAL DESCRIPTION OF THE FUEL ROD #A REVIEW OF INTERNATIONAL SOLUTIONS TO NEACRP VAPOR NOZZLE FLOW (IN GERMAN)	BEHAVIOR OF A PWR FUEL ROD DURING A LOCA (IN FRENCH) BEHAVIOR OF SPECIFIC REACTOR MATERIALS AND COMPONENT PARTS BEHAVIOR UNDER ACCIDENT CONDITIONS (IN GERMAN) THEORETICAL BEHAVIOR DURING LOCA (IN GERMAN) BENCHMARK 0WR LATTICE CELL PROBLEMS	145071 P 1 143003 P 13 143740 P 19 145156 P 22 147750 P 22
2) COMPARISON OF THERMAL INTERACTIONS BETWEEN CORROSION AND WATER	BENCHMARK 0WR LATTICE CELL PROBLEMS	143211 P 15 140048 P 20 143910 P 22
AND A BWR STANDARD PLANT, AND THE COUPLING OF MELISSA-1 AND OF DECOMMISSIONED NUCLEAR #R236 - FINAL REPORT CONTROLLED-GRID SPACERS (IN ENGLISH) #LOCAL FLOW	BETWEEN CORROSION AND WATER	140048 P 20 143910 P 22
3) EMERGENCY COOLING DEPRESSURIZATION RESEARCH, COEFFICIENT IN THE CONTAINMENT DURING A COOLING SYSTEM ELECTRICAL HEATING POWER (IN GERMAN)	BILANZ-1, PART I (IN GERMAN) WITH MELISSA-1 IN A PWR BLASTING DEMONSTRATION OF RADIOACTIVE PRIMARY LOOP COMPONENTS BLOCKAGE EXPERIMENTS IN 37-PIN SODIUM COOLED BUNDLES WITH BLOCKAGE PHENOMENON FOR A TWO PHASE FLOW (IN FRENCH) BLOCKED COOLING CHANNELS WITH BWR GEOMETRY (IN GERMAN) BLOWDOWN (IN GERMAN) DETERMINATION OF THE HEAT TRANSFER BLOWDOWN - EXPERIMENT R2 109 (LODI) CONTROL OF THE BLOWDOWN ACCIDENT AND THE SUBSEQUENT THERMAL TRANSIENT IN BLOWDOWN ANALYSIS (IN ENGLISH)	145450 P 25 147017 P 23 143907 P 25 143911 P 1 140050 P 23 140297 P 10 143054 P 10 145074 P 14 147502 P 29
4) LIGHT-WATER #COMPARATIVE INVESTIGATIONS OF A COOLING SYSTEM #ALARM-FIT #COMPUTER PROGRAM FOR PRESSURIZED WATER REACTOR HEATED ROD BUNDLE (IN FRENCH)	BLOWDOWN OF A PART OF THE LOOP OMEGA INCLUDING A 36 DIRECT BLOWDOWN OF A PRESSURIZED WATER REACTOR-FLUX (IN GERMAN) BLOWDOWN OF A WATER-COOLED REACTOR - INTERIM RESEARCH BLOWDOWN OF THE SECONDARY SIDE OF A STEAM GENERATOR (IN BLOWDOWN OF A TUBULAR TEST SECTION ON OMEGA LOOP (IN BLOWDOWN PHASE, PARTS A, B, & C (IN GERMAN) THEORETICAL BLOWDOWN) COMPARISON OF SEVERAL MODELS (IN FRENCH) THE BMT IN THE AREA OF REACTOR SAFETY, JULY 1-SEPTEMBER 30, 1970; LORAC, EPRI AND JSA, JULY 1-SEPTEMBER 30, 1970 (IN BMT, USNR, EPRI, AND JSA, REPORT PERIOD OCTOBER 1 - BOILING AND SLIP RATIO FROM MEASUREMENTS MADE IN LINDEN BWR BOILING EXPERIMENT IN A 19-PIN BUNDLE (IN ENGLISH) BOILING-WATER REACTOR STEAM CROGGING (IN ENGLISH) BREAK RUNS #12 AND #13 (IN ENGLISH & JAPANESE) WITH BREAKS (RUNS #32, #13, #23) (IN ENGLISH & JAPANESE) BREEDER REACTOR DEVELOPMENT IN JAPAN, APRIL-JUNE 1970 BUBBLE (I) A COMPUTER PROGRAM TO DESCRIBE THERMAL NON-BUBBLE FLOW MODEL IN RELAP-0K	143929 P 8 140054 P 20 145070 P 10 140734 P 5 143700 P 3 143901 P 24 143751 P 2 145165 P 16 144054 P 16 140813 P 17 140847 P 21 143776 P 20 140600 P 17 144532 P 27 143891 P 27 143243 P 22 140447 P 22 145041 P 21
5) OF THREE-DIMENSIONAL FLUID-STRUCTURE INTERACTIONS DURING IN A MULTICOMPARTMENTED CONTAINMENT FROM THE LOUANT FRENCH) #ANA EXPERIMENTAL AND THEORETICAL STUDY OF THE FRENCH) #DATA REDUCTION OF THE FIRST TEST SERIES OF RESEARCH ON THE THERMAL HYDRAULIC BEHAVIOR IN THE INITIAL CALCULATION OF THE RATE OF VOIDING DURING A RAPID FAILURE (1970 (IN #REPORT ON THE RESEARCH PROGRAM SPONSORED BY OF REPORTS FROM THE REACTOR SAFETY RESEARCH PROGRAMS OF OF REPORTS FROM THE REACTOR SAFETY RESEARCH PROGRAMS OF #AN APPRAISAL OF SUBCOOLED #ACOUSTIC NOISES WITH LOSS-OF-FLOW SODIUM #ANALYSIS OF RELAP-0 CODE (ANALYSIS OF RESEA-11 EXPERIMENTS FOR COLD LEG AND PUMP CIRCULATION ON LICA PHENOMENA IN LARGEST COLD LEG #PROGRESS REPORT ON FAST EQUILIBRIUM FLOW OF WATER IN SIMPLE PIPE SYSTEMS #MODEL OF DEVELOPMENT OF THE #STUDY OF THE RANGE OF VELOCITY OF GAS INSIDE A BUBBLE RISING THROUGH A LIQUID CORE #EXPERIMENTAL TESTS IN WITH LOSS-OF-FLOW SODIUM BOILING EXPERIMENT IN A 19-PIN A PART OF THE LOOP OMEGA INCLUDING A 36 DIRECT HEATED ROD GERMAN) #EVALUATION OF THE 25-ROD #LOCAL FLOW BLOCKAGE EXPERIMENTS IN 37-PIN SODIUM COOLED #FLOW REDUCTION TRANSIENT #A PROBABILISTIC STUDY OF VESSEL DEVELOPMENT OF SYSTEMS LIMITING THE H2-CONCENTRATION IN THE DEPRESSURIZATION RESEARCH, BLOCKED COOLING CHANNELS WITH #A REVIEW OF INTERNATIONAL SOLUTIONS TO NEACRP BENCHMARK OF THE CORE MELT ACCIDENT WITH MELISSA-1 IN A PWR AND A BOILING AND SLIP RATIO FROM MEASUREMENTS MADE IN LINDEN	143929 P 8 140054 P 20 145070 P 10 140734 P 5 143700 P 3 143901 P 24 143751 P 2 145165 P 16 144054 P 16 140813 P 17 140847 P 21 143776 P 20 140600 P 17 144532 P 27 143891 P 27 143243 P 22 140447 P 22 145041 P 21 143210 P 2 140670 P 6 143776 P 20 143929 P 8 143902 P 14 143907 P 25 140844 P 29 140869 P 6 147473 P 23 140050 P 23 147750 P 22 140450 P 25 140847 P 21	
6) VELOCITY OF GAS INSIDE A BUBBLE RISING THROUGH A LIQUID CORE GERMAN) #IMPACT OF STEEL PROJECTILES ON REINFORCED CONCRETE, #COMPARISON BETWEEN A ONE- AND TWO-DIMENSIONAL AND ENGLISH) #POST-EXPERIMENT BLOWDOWN) COMPARISON OF SEVERAL MODELS (IN #MODEL FOR THE MEDIUM IN COOLED #POST-20 - A COMPUTER CODE FOR THE CHAMBER OF A #COMPUTER MODEL FOR THE TWO-DIMENSIONAL - FAILURE PHASE OF THE HYPOTHETICAL CORE MELTDOWN ACCIDENT; USING FINITE ELEMENT METHOD (IN FRENCH) #EVALUATION OF THE 25-ROD BUNDLE TEST (RS-37C) WITH THE EQUATION OF A MULTIDIMENSIONAL HYDRO MODEL #DEVELOPMENT OF A #PATREC, A COMPUTER CODE FOR FAULT TREE #RESULTS OF THE LOFT EXPERIMENT L1-42 POST RUN OF THE NONNUCLEAR LOFT-TESTS L1-4 (PRE AND POST TEST #EXPERIENCE SUPER-#SUPER OF INTERNATIONAL SOLUTIONS TO NEACRP BENCHMARK 0WR LATTICE CELL PROBLEMS	CANDU 1 (IN FRENCH) #STUDY OF THE RANGE OF CALCULATING PLASTIC DEFORMATION OF STRUCTURES (IN FRENCH) CALCULATION AND COMPARISON WITH EXPERIMENTAL TESTS (IN CALCULATION OF A WATER-VAPOR NOZZLE FLOW (IN GERMAN) CALCULATION OF THE NON-NUCLEAR LOFT TEST L1-5 (IN GERMAN) CALCULATION OF THE RATE OF VOIDING DURING A RAPID FAILURE (CALCULATION OF THE TWO-DIMENSIONAL FLOW OF A COMPRESSIBLE CALCULATION OF THE WATER POOL-SWELL IN THE CONDENSATION CALCULATION TO IDENTIFY THE INFLUENCE OF VARIOUS CALCULATIONAL METHOD FOR TWO DIMENSIONAL VISCOUS FLOW CALCULATIONAL PROGRAM (IN GERMAN) CALCULATIONAL PROGRAM FOR THE SOLUTION OF THE NEUTRONIC CALCULATIONS (IN ENGLISH) CALCULATIONS USING THE COMPUTER CODE "ORIFAN" (IN GERMAN) CALCULATIONS (IN GERMAN) #RELAP-NAGRS ANALYSIS CANDU (IN FRENCH) CANDU EXPERIMENTS (IN ENGLISH) CELL PROBLEMS	143210 P 2 143875 P 3 145030 P 21 143211 P 15 140074 P 16 143751 P 2 146750 P 20 145757 P 15 143374 P 15 143753 P 2 143902 P 14 140072 P 17 143776 P 8 140668 P 16 143905 P 9 143914 P 2 147750 P 22 140048 P 20 143910 P 22 145757 P 15 140050 P 23 147101 P 7
7) CALCULATION OF THE WATER POOL-SWELL IN THE CONDENSATION COOLING DEPRESSURIZATION RESEARCH, BLOCKED COOLING DERIVED FROM PARTIAL HYPERBOLICS (IN FRENCH)	CHARACTERISTICS AND RESOLUTIONS OF THE SYSTEM OF EQUATIONS	147101 P 7

INTERNAL CORROSION ON THE MECHANICAL #SIMULATION OF THE CHEMICAL STATE OF IRRADIATED OXIDE FUELS INFLUENCE OF THE 148673 P 20
 #ANALYSIS OF BOILING-WATER REACTOR STEAM CRUDDING (IN ENGLISH)* 148680 P 17
 #CONTAMINATION OF A PRESSURIZED WATER REACTOR'S PRIMARY #CIRCUIT BY FUEL RODS SHOWING MANUFACTURING FAULTS (IN 143383 P 4
 CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS; CONDENSATION IN 146273 P 12
 CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS; QUICK LOOK REPORT 145846 P 10
 CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUITS IN WATER-COOLED REACTORS; SUPPLEMENTAL RESEARCH 145847 P 11
 II TEST DATA REPORT 12 EFFECTS OF ECCS INJECTION AND PUMP #CIRCULATION ON LOCA PHENOMENA IN LARGEST COLD LEG BREAKS (143891 P 27
 THE INHALATION OF PLUTONIUM OXIDE USING OBSERVED LONG CLEARANCE PATTERNS* #ASSESSMENTS OF RISK FOLLOWING 145281 P 30
 PRESSURE TRANSIENTS IN FULL-PRESSURE CONTAINMENTS OF WATER-#CO. FLOW - A COMPUTER CODE FOR THE DETERMINATION OF 145036 P 16
 RUNS 413 AND #ANALYSIS OF LOCA EXPERIMENTS WITH RELAP-2 CODE (ANALYSIS OF ROSA-II EXPERIMENTS FOR COLD LEG BREAK 144532 P 27
 #AN ANALYSIS OF LOFT L1-2 EXPERIMENT BY ALARM-PI COMPUTER CODE (IN ENGLISH)* 144526 P 27
 EXPERIMENT L1-41 POST RUN CALCULATIONS USING THE COMPUTER #CODE "DRUFAN" (IN GERMAN)* #RESULTS OF THE LOFT 145068 P 10
 OF VELOCITY OF GAS INSIDE A BUBBLE RISING THROUGH A LIQUID #CODE CABU 1 (IN FRENCH)* #STUDY OF THE RANGE 143210 P 2
 PWR FUEL ROD DURING A LOCA (IN FRENCH)* #CUPIDONT; A CODE DESCRIBING THE THERMAL AND MECHANICAL BEHAVIOR OF A 145071 P 1
 #PATREC; A COMPUTER CODE FOR FAULT TREE CALCULATIONS (IN ENGLISH)* 143778 P 4
 #FRENCH THERMO- 143758 P 6
 HYDRAULIC STUDIES FOR THE DEVELOPMENT OF SAFETY ADVANCED #CODE FOR THE DETERMINATION OF PRESSURE TRANSIENTS IN FULL- 145036 P 10
 PRESSURE CONTAINMENTS OF WATER-#CO. FLOW - A COMPUTER CODE TRAC-PI (VERSION 16.3) TO IOM STANDARD OPERATING 143966 P 14
 SYSTEM MVS (WITH #REPORT ON THE CONVERSION OF THE LASH- #EFFICIENT IN THE CONTAINMENT DURING A COOLING SYSTEM 145297 P 10
 BLOWDOWN) #EXPERIMENTAL DETERMINATION OF THE HEAT TRANSFER #COEFFICIENT IN THE CONTAINMENT DURING A COOLING SYSTEM 144532 P 27
 WITH RELAP-2 CODE (ANALYSIS OF ROSA-II EXPERIMENTS FOR #COLD LEG BREAK RUNS 413 AND 312) (IN ENGLISH & JAPANESE)* 143891 P 27
 INJECTION AND PUMP CIRCULATION ON LOCA PHENOMENA IN LARGEST #COLD LEG BREAKS (RUNS 302, 413, 425) (IN ENGLISH & 143768 P 21
 #DEVELOPMENT AND SYNTHESIS OF AN EDUCATIONAL SYSTEM USING #COMBINATION OF MEDIA FOR THE INTENSIVE TRAINING AND 145874 P 12
 ACCIDENT AND THE SUBSEQUENT THERMAL TRANSIENT IN LIGHT-#COMPARATIVE INVESTIGATIONS OF A COOLING SYSTEM BLOWDOWN 143211 P 19
 OF A WATER-VAPOR NOZZLE FLOW (IN GERMAN)* #COMPARISON BETWEEN A ONE- AND TWO-DIMENSIONAL CALCULATION 143751 P 2
 OF THE RATE OF VOIDING DURING A RAPID FAILURE (BLOWDOWN) #COMPARISON OF SEVERAL MODELS (IN FRENCH)* THE CALCULATION 145038 P 21
 STEEL PROJECTILES ON REINFORCED CONCRETE, CALCULATION AND #COMPARISON WITH EXPERIMENTAL TESTS (IN GERMAN)* #IMPACT OF 145047 P 10
 #INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI- #COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY 145047 P 11
 #INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI- #COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY 146273 P 12
 #INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI- #COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY 143329 P 10
 #INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI- #COMPARTMENT CONTAINMENT BY PRESSURE IN WATER-COOLED 143966 P 14
 REACTORS WITH #INVESTIGATION OF THE PROCESSES IN A MULTIPLE #COMPIER) (IN ENGLISH) #ALW-CODE TRAC-PI (VERSION 10.3) TO 143803 P 13
 IOM STANDARD OPERATING SYSTEM MVS (WITH FORTRAN-H-EXTENDED #COMPONENT PARTS AT IMPACT OF FRAGMENTS AND PROJECTILES OF 146483 P 33
 DIFFERENT MASS #BEHAVIOR OF SPECIFIC REACTOR MATERIALS AND #COMPONENT. TWO-PHASE SYSTEMS - THE COMPUTER PROGRAMS CRIS 147860 P 20
 EQUILIBRIUM CRITICAL DISCHARGE MODEL APPLIED TO MULTI- #COMPONENTS (IN ENGLISH)* 145155 P 22
 #FAULT TREE ANALYSIS WITH MULTISTATE #COMPONENTS (IN GERMAN)* PROGRAM FOR THE TESTING 147779 P 24
 OF A FAILURE SAFETY DEVICE PROTECTION SYSTEM FOR REACTOR #COMPONENTS AND THE APPARENT INFLUENCE OF THESE OCCURRENCES 147817 P 23
 IN FAULT TREE ANALYSIS (UNCERTAINTY OF THE FAILURE RATE OF #COMPONENTS OF DECOMMISSIONED NUCLEAR POWER PLANTS (IN 148070 P 10
 CONTROLLED-BLASTING DEMOLITION OF RADIOACTIVE PRIMARY LOOP #COMPREHENSIVE DAMAGE ANALYSIS FOR HIGH TEMPERATURE GAS 148272 P 11
 REACTORS, PHASE II, WATER INGRESS, AIR INGRESS, REACTIVITY) #COMPREHENSIVE SUMMARY OF THE THEORETICAL STUDIES ON THE D- 146750 P 20
 SERIES OF THE RESEARCH PROGRAM RS 50 (MODEL CONTAINMENT) #COMPRESSIBLE MEDIUM IN COUPLED RECTANGULAR AREAS (IN 146804 P 20
 FOR THE CALCULATION OF THE TWO-DIMENSIONAL FLOW OF A #COMPUTATION OF THREE-DIMENSIONAL FLUID-STRUCTURE 144026 P 27
 INTERACTIONS DURING BLOWDOWN OF A PRESSURIZED #EFFICIENT COMPUTER CODE (IN ENGLISH)* 145868 P 16
 #AN ANALYSIS OF LOFT L1-2 EXPERIMENT BY ALARM-PI #CODE "DRUFAN" (IN GERMAN)* #RESULTS OF 143778 P 4
 THE LOFT EXPERIMENT L1-41 POST RUN CALCULATIONS USING THE #PATREC; A COMPUTER CODE FOR FAULT TREE CALCULATIONS (IN ENGLISH)* 145036 P 16
 #FRENCH THERMO- 143758 P 6
 IN FULL-PRESSURE CONTAINMENTS OF WATER-COOLED #CO. FLOW - A #CODE FOR THE DETERMINATION OF PRESSURE TRANSIENTS 145757 P 15
 WATER POOL-SWELL IN THE CONDENSATION CHAMBER OF A REACTOR #COMPUTER MODEL FOR THE TWO-DIMENSIONAL CALCULATION OF THE 145297 P 27
 BEHAVIOR IN A PWR CORE DURING A LOCA (IN #ASCCT-1; A COMPUTER PROGRAM FOR ANALYZING THE THERMO-HYDRAULIC 147502 P 29
 ANALYSIS (IN ENGLISH)* #ALARM-PI; A COMPUTER PROGRAM FOR PRESSURIZED WATER REACTOR BLOWDOWN 147602 P 33
 #STERION 4 - A COMPUTER PROGRAM FOR USE IN NUCLEAR SAFETY STUDIES* 140447 P 32
 #HUBBLE BUBBLE III; A COMPUTER PROGRAM TO DESCRIBE THERMAL NON-EQUILIBRIUM FLOW 146471 P 33
 MODEL APPLIED TO MULTI-COMPONENT, TWO-PHASE SYSTEMS - THE #COMPUTER PROGRAMS CRIS AND CRITER* CRITICAL DISCHARGE 144527 P 28
 PRESSURE AND DENSITY AS THE INDEPENDENT VARIABLES #SPACE; A #COMPUTER SUBROUTINE FOR GENERATING STEAM TABLES HAVING 145050 P 20
 FLOW OF A COMPRESSIBLE MEDIUM IN COUPLED #FLUST-2D - A COMPUTER CODE FOR THE CALCULATION OF THE TWO-DIMENSIONAL 147473 P 23
 #INVESTIGATION AND DEVELOPMENT OF SYSTEMS LIMITING THE #CONCENTRATION IN THE BWR CONTAINMENT (IN GERMAN)* 147483 P 10
 #REPORT OF THE FEDERAL MINISTER FOR RESEARCH AND TECHNOLOGY #CONCERNING RESEARCH PROJECTS IN THE AREA OF REACTOR SAFETY 143379 P 15
 ACCIDENT: CALCULATION TO #THE COURSE OF EVENTS IN THE #CONCRETE - FAILURE PHASE OF THE HYPOTHETICAL CORE MELTDOWN 145038 P 21
 TESTS (IN GERMAN) #IMPACT OF STEEL PROJECTILES ON REINFORCED #CONCRETE, CALCULATION AND COMPARISON WITH EXPERIMENTAL 145757 P 15
 TWO-DIMENSIONAL CALCULATION OF THE WATER POOL-SWELL IN THE #CONDENSATION CHAMBER OF A REACTOR SYSTEM (IN GERMAN)* THE 146273 P 12
 OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS; #CONDENSATION IN CONTAINMENT BY EXPERIMENTS C04 AND C1 TO 149000 P 7
 CONDITIONS; ON A STAINLESS STEEL TEST SECTION #STUDY ON THE #CONDENSATION OF AIR AND STEAM MIXTURES, IN TRANSIENT 148846 P 34
 ELECTRONIC TRANSITIONS CONTRIBUTE TO THE THERMODYNAMICS OF #CONDENSED UO(2) ? A REVIEW OF THE ARGUMENTS* #UO 143329 P 10
 BY PRESSURE IN WATER-COOLED REACTORS WITH REFRIGERATED #CONDENSER (IN GERMAN)* A MULTIPLE COMPARTMENT CONTAINMENT 143089 P 19
 #REEP RUPTURE AT NON-STEADY STRESS AND TEMPERATURE LOADING #CONDITIONS (IN ENGLISH)* 143745 P 19
 INVESTIGATIONS ON THE LWR FUEL ROD BEHAVIOR UNDER ACCIDENT #CONDITIONS (IN GERMAN) #OF THE THEORETICAL AND EXPERIMENTAL 146795 P 24
 LOCA FOR PURPOSES OF STUDYING THE ENSUING #SPECIFICATION OF #CONDITIONS OF A NUCLEAR POWER PLANT WITH A PWR FOLLOWING A 149000 P 7
 ON THE CONDENSATION OF AIR AND STEAM MIXTURES, IN TRANSIENT #COMPIED HEMISPHERICAL STRATIFIED GASEOUS MIXTURES (IN 143078 P 8
 ENGLISH)* #FLAME PROPAGATION THROUGH UNCONFINED AND #CONFINED NUCLEAR FUEL CYCLE (IN GERMAN)* 147102 P 15
 #THE ESSENTIAL SAFETY ASPECTS OF A #CONTAINMENT (IN GERMAN)* #INVESTIGATION AND DEVELOPMENT 147473 P 23
 OF SYSTEMS LIMITING THE H2-CONCENTRATION IN THE BWR #CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT 145846 P 10
 OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT #CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT 148273 P 12
 OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT #CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUITS 145847 P 11
 COOLING CIRCUIT IN WATER-COOLED REACTORS; CONDENSATION IN #CONTAINMENT BY EXPERIMENTS C04 AND C1 TO C16 (IN GERMAN)* 146273 P 12
 #INVESTIGATION OF THE PROCESSES IN A MULTIPLE COMPARTMENT #CONTAINMENT BY PRESSURE IN WATER-COOLED REACTORS WITH 143329 P 10
 DETERMINATION OF THE HEAT TRANSFER COEFFICIENT IN THE #CONTAINMENT DURING A COOLING SYSTEM BLOWDOWN (IN GERMAN)* 146297 P 10
 OF THE HYDROGEN DISTRIBUTION IN A LIGHT WATER REACTOR #CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT, QUICK 144582 P 11
 OF THE HYDROGEN DISTRIBUTION IN A LIGHT WATER REACTOR #CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT, QUICK 144286 P 12
 OF THE PRESSURE TRANSIENT IN A MULTICOMPARTMENT #CONTAINMENT FROM THE COOLANT BLOWDOWN OF A WATER-COOLED 145875 P 10
 #RELIABILITY ASSESSMENT OF THE SECONDARY #CONTAINMENT OF A PWR (IN GERMAN)* 146820 P 24
 #SAFETY #CONTAINMENT OF NUCLEAR POWER PLANTS (IN GERMAN)* 47176 P 17
 #THE #CONTAINMENT TEST FACILITY (EXPERIMENTS C AND D) (IN GERMAN 143806 P 12
 ON THE D-SERIES OF THE RESEARCH PROGRAM RS 50 (MODEL #CONTAINMENT) PART 1 (IN GERMAN) #OF THE THEORETICAL STUDIE; 148272 P 11
 THE D SERIES EXPERIMENTS OF RESEARCH PROJECT RS 50 (MODEL #CONTAINMENT) PART 2 (IN GERMAN)* #ANALYSIS OF 146798 P 11
 THE DETERMINATION OF PRESSURE TRANSIENTS IN FULL-PRESSURE #CONTAINMENTS OF WATER-COOLED NUCLEAR POWER PLANTS (IN 145036 P 16
 CIRCUIT BY FUEL RODS SHOWING MANUFACTURING FAULTS (IN #CONTAMINATION OF A PRESSURIZED WATER REACTOR'S PRIMARY 143383 P 4
 REVIEW OF THE ARGUMENTS* #DOO ELECTRONIC TRANSITIONS #CONTRIBUTE TO THE THERMODYNAMICS OF CONDENSED UO(2)? A 148846 P 34
 #THE ELECTRONIC #CONTRIBUTION TO THE THERMODYNAMICS OF MOLTEN UO(2)? 148846 P 34

#THE ELECTRICAL BLOWDOWN - EXPERIMENT RS 109 (CLOSED COMPONENTS OF DECOMMISSIONED NUKLEAR - #RS236 - FINAL REPORT STANDARD OPERATING SYSTEM RYS WITH FORTRAN - #REPORT ON THE COOLING (IN GERMAN) #FUEL ELEMENT BEHAVIOR DURING A LOSS-OF- IN A LIGHT WATER REACTOR CONTAINMENT FOLLOWING A LOSS-OF- TRANSIENT IN A MULTICOMPARTMENTED CONTAINMENT FROM THE HEATING METHOD (ENGLISH) - #20 SYSTEMS (IN JAPANESE) #FUEL LOCAL FLOW BLOCKAGE EXPERIMENTS IN 37-PER SODIUM PRESSURE TRANSDUCERS IN FULL-PRESSURE CONTAINMENTS OF WATER- CONTAINMENT FROM THE COOLANT BLOWDOWN OF A WATER- MECHANICS RESEARCH ON MATERIALS USED IN FAST SODIUM- IN A MULTIPLE COMPARTMENT CONTAINMENT BY PRESSURE IN WATER- AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER- AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER- AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER- IF-COOLANT ACCIDENT AND INTERACTION WITH THE EMERGENCY CORE- #EMERGENCY COOLING DEPRESSURIZATION RESEARCH, BLOCKED MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY WITH DWR GEOMETRY (IN GERMAN) #EMERGENCY THE HEAT TRANSFER COEFFICIENT IN THE CONTAINMENT DURING A THERMAL TRANSIENT IN LIGHT-#COMPARATIVE INVESTIGATIONS OF A MODELIZATION DURING THE REFLOODING PHASE OF A PWR'S LOSS-OF-COOLANT ACCIDENT AND INTERACTION WITH THE EMERGENCY FOR ANALYZING THE THERMO-HYDRAULIC BEHAVIOR IN A PWR SMOOTH #INVESTIGATION OF THE VARIOUS PHASES OF THE STANDARD PLANT, #INVESTIGATION OF THE FIRST PHASE OF THE EVENTS IN THE CONCRETE - FAILURE PHASE OF THE HYPOTHETICAL OF THE CORE MELT ACCIDENT AFTER THE AFTER FAILURE OF THE STATE OF IRRADIATED URINE FUEL INFLUENCE OF THE INTERNAL #ACTIVATED OF THE TWO-DIMENSIONAL FLOW OF A COMPRESSIBLE MEDIUM IN WITH MELSIM-1 IN A PWR AND A DWR STANDARD PLANT, AND THE HYPOTHETICAL CORE MELT DOWN ACCIDENT: CALCULATION TO #THE EVENTS: EARTHQUAKES, FIRES, EXPLOSIONS AND AIRCRAFT CONDITIONS (IN ENGLISH) #A SIMPLE HOMOGENEOUS EQUILIBRIUM FLOW (IN FRENCH) #EXPERIMENTAL STUDY OF MULTI-COMPONENT, TWO-PHASE SYSTEMS - THE COMPUTER PROGRAMS TWO-PHASE SYSTEMS - THE COMPUTER PROGRAMS CRITS AND BEHAVIOR OF A PWR FUEL ROD DURING A LOCA (IN FRENCH) #ERROR ANALYSIS OF THE AMPLITUDE #THE ESSENTIAL SAFETY ASPECTS OF A CONFINED NUCLEAR FUEL ANALYSIS OF A PROGRESSIVE DISTURBANCE PROBLEM (TERMINUS AND RELEASE FROM A DEFECTED FUEL ROD - EFFECT OF THERMAL COOLED REACTORS, CONDENSATION IN CONTAINMENT BY EXPERIMENTAL CONDENSATION IN CONTAINMENT BY EXPERIMENTS C04 AND CONDENSATION IN CONTAINMENT BY EXPERIMENTS C04 AND C1 TO

#THE CONTAINMENT TEST FACILITY EXPERIMENTS L AND #COMPREHENSIVE SUMMARY OF THE THEORETICAL STUDIES ON THE (1), WATER INGRESS, AIR INGRESS, REACTIVITY #COMPREHENSIVE #INSTRUMENTATION SYSTEM FOR THE FRIEDRICH TEST SECTION ON OMEGA LOOP (IN FRENCH) #CIRCULATION ON LOCA PHENOMENA IN LARGEST COLO #ROSA-II TEST AND ERROR RECONSTRUCTION WITH HELP OF TIME-OF-FLIGHT DEMOLITION OF RADIOACTIVE PRIMARY LOOP COMPONENTS OF A PWR FOLLOWING A LOCA FOR PURPOSES OF STUDYING THE PASSING LASER HOLOGRAPHIC AND SPECKLE PHOTOGRAPHY METHODS FOR #ANALYSIS OF THE FISSION PRODUCT RELEASE FROM A #CALCULATING PLASTIC DECOMMISSIONED. #RS236 - FINAL REPORT CONTROLLED-ELASTING SUBROUTINE FOR GENERATING STEAM TABLES HAVING PRESSURE AND RESONANCE (IN GERMAN) #DEVELOPMENT OF A MASS-DWR GEOMETRY (IN GERMAN) #EMERGENCY COOLING #CHARACTERISTICS AND RESOLUTIONS OF THE SYSTEM OF EQUATIONS #MODEL OF THE FAILURE RATES OF THE VALVES OF ST. LAURENT PIPE SYSTEMS #BUZZLE BOBBLE III: A COMPUTER PROGRAM TO FUEL ROD DURING A LOCA (IN FRENCH) #CUPIDANT: A CODE FRENCH) #STUDY OF THE STABILITY OF VARIOUS SYSTEMS AND GERMAN) #PRELIMINARY EMPIRICAL HOLOGRAPHIC AND SPECKLE PHOTOGRAPHY METHODS FOR DEFECT CONTAINMENTS OF WATER #CO. FLOW - A COMPUTER CODE FOR THE CONTAINMENT DURING A COOLING SYSTEM BLOWDOWN (EXPERIMENTAL SHOCK WAVES IN THE ENVIRONMENT (IN GERMAN) #INITIATION OF #EXPERIMENTAL STUDY OF THE OVERPRESSURE GENERATED BY THE #INVESTIGATION PROGRAM FOR THE TESTING OF A FRACTURE SAFETY COMPONENT PARTS AT IMPACT OF FRAGMENTS AND PROJECTILES OF GERMAN) #COMPARISON BETWEEN A ONE- AND TWO- CONDENSATION CHAMBER OF A #COMPUTER MODEL FOR THE TWO-

CONTRIBUTION TO THE THERMODYNAMICS OF SODIUM 142773 P 23
CONTROL OF THE ELECTRICAL HEATING POWER (IN GERMAN) 142804 P 10
CONTROLLED-BLASTING DEMOLITION OF RADIOACTIVE PRIMARY LOOP 142812 P 23
CONVERSION OF THE LASS-CODE THREE-DIMENSIONAL TOWER TO IOM 142850 P 14
COOLANT ACCIDENT AND INTERACTION WITH THE EMERGENCY CORE 142900 P 15
COOLANT ACCIDENT, QUICK LOOK REPORT 1 (IN GERMAN) 142920 P 11
COOLANT ACCIDENT, QUICK LOOK REPORT 2 (IN GERMAN) 142920 P 11
COOLANT BLOWDOWN OF A WATER-COOLED REACTOR - INTERIM 142975 P 11
COOLANT INTERACTION EXPERIMENT BY DIRECT ELECTRICAL 143042 P 25
COOLED BUNDLES WITH GRID SPACERS (IN ENGLISH) 143097 P 10
COOLED NUCLEAR POWER PLANTS (IN GERMAN) DETERMINATION OF 143130 P 10
COOLED REACTOR - INTERIM RESEARCH REPORT C 12 (IN GERMAN) 143170 P 10
COOLED REACTORS (IN GERMAN) MEASUREMENTS IN FRACTURE 14 10 P 10
COOLED REACTORS WITH REGENERATED CONDENSER (IN GERMAN) 143229 P 10
COOLED REACTORS, CONDENSATION IN CONTAINMENT BY 143270 P 10
COOLED REACTORS, QUICK LOOK REPORT, EXPERIMENT D10 (IN 143296 P 10
COOLED REACTORS, SUPPLEMENTAL RESEARCH DOCUMENTATION D10 1 143340 P 10
COOLING (IN GERMAN) #FUEL ELEMENT BEHAVIOR DURING A LOSS- 143390 P 15
COOLING CHANNELS WITH DWR GEOMETRY (IN GERMAN) 143420 P 10
COOLING CIRCUIT IN WATER-COOLED REACTORS, CONDENSATION IN 143470 P 10
COOLING CIRCUIT IN WATER-COOLED REACTORS, QUICK LOOK 143486 P 10
COOLING CIRCUITS IN WATER-COOLED REACTORS, SUPPLEMENTAL 143547 P 10
COOLING DEPRESSURIZATION RESEARCH, BLOCKED COOLING CHANNELS 143590 P 10
COOLING SYSTEM BLOWDOWN (IN GERMAN) DETERMINATION OF 143657 P 10
COOLING SYSTEM BLOWDOWN ACCIDENT AND THE SUBSEQUENT 143674 P 10
CORE (IN FRENCH) #FLOW AND HEAT TRANSFER THERMODYNAMIC 143720 P 10
CORE COOLING (IN GERMAN) #FUEL ELEMENT BEHAVIOR DURING A 143740 P 15
CORE DURING A LOCA (IN ENGLISH) II: A COMPUTER PROGRAM 143790 P 27
CORE MELT ACCIDENT AFTER THE AFTER FAILURE OF THE CORE 143820 P 10
CORE MELT ACCIDENT WITH MELSIM-1 IN A PWR AND A DWR 143850 P 10
CORE MELTDOWN ACCIDENT: CALCULATION TO IDENTIFY THE 143870 P 25
CORE SUPPORT STRUCTURE USE TO THE FORMATION OF MELT OR UOL 143700 P 20
CORROSION IN THE MECHANICAL PROPERTIES OF ZIRCONIUM 1 143973 P 25
CORROSION PRODUCTS IN LWR COOLERS (IN GERMAN) 143980 P 24
COUPLED RECTANGULAR AREAS (IN GERMAN) FOR THE CALCULATION 144000 P 10
COUPLING OF MELSIM-1 AND OILAPP-1, PART I (IN GERMAN) 144050 P 10
COURSE OF EVENTS IN THE CONCRETE - FAILURE PHASE OF THE 144070 P 10
CRACKS (IN ENGLISH) #POWER PLANTS (EMPS) AGAINST EXTERNAL 144120 P 10
CRACK RUPTURE AT NON-STEADY STRESS AND TEMPERATURE LOADING 144180 P 25
CRITICAL DISCHARGE MODEL APPLIED TO MULTI-COMPONENT, TWO- 144240 P 10
CRITICAL FLOW AND FLOW BLOCKAGE PHENOMENON FOR A TWO PHASE 144311 P 10
CRITICAL TWO-PHASE FLOW (IN ENGLISH) 144390 P 7
CRITS AND CRITTERS - CRITICAL DISCHARGE MODEL APPLIED TO 144480 P 27
CRITTERS - DISCHARGE MODEL APPLIED TO MULTI-COMPONENT, 144580 P 10
CUPIDANT: A CODE DESCRIBING THE THERMAL AND MECHANICAL 144670 P 10
CURVE, FINAL REPORT (IN GERMAN) 144720 P 10
CYCLE (IN GERMAN) 144710 P 10
CYCLE (ARABIC) (IN FRENCH) #SHORT 144871 P 6
CYCLING (IN ENGLISH) #ANALYSIS OF THE FEEDBACK PRODUCT 144950 P 10
C04 AND C1 TO C10 (IN GERMAN) COOLING CIRCUIT IN WATER- 145070 P 10
C1 TO C10 (IN GERMAN) CIRCUIT IN WATER-COOLED REACTORS, 145070 P 10
C10 (IN GERMAN) COOLING CIRCUIT IN WATER-COOLED REACTORS, 145070 P 10

D3 (IN GERMAN) 143200 P 10
D-SERIES OF THE RESEARCH PROGRAM RS 50 (MODEL CONTAINMENT) 143270 P 10
DAMAGE ANALYSIS FOR HIGH TEMPERATURE GAS REACTORS, PHASE 143300 P 10
DAS-MULTIPLE TUBE RESEARCH PROGRAM (IN GERMAN) 143350 P 10
DATA REDUCTION OF THE FIRST TEST SERIES OF BLOWDOWN ON A 143350 P 2
DATA REPORT 12 EFFECTS OF ECCS INJECTION AND PUMP 143390 P 27
DATA, FINAL REPORT (IN GERMAN) #DISTURBANCE 143420 P 10
DECOMMISSIONED NUCLEAR POWER PLANTS (IN GERMAN) #BLASTING 143470 P 23
DECONTAMINATION AND TRANSPORT (IN GERMAN) #POWER PLANT WITH 143490 P 24
DEFECT DETECTION AND STRAIN EVALUATION IN PRESSURE VESSELS 143540 P 21
DEFECTED FUEL ROD - EFFECT OF THERMAL CYCLING (IN ENGLISH) 143550 P 0
DEFORMATION OF STRUCTURES (IN FRENCH) 143570 P 2
DEMOLITION OF RADIOACTIVE PRIMARY LOOP COMPONENTS OF 143610 P 23
DENSITY AS THE INDEPENDENT VARIABLES (IN ENGLISH) 143620 P 20
DENSITY METHOD FOR TRANSIENT TWO PHASE STATE USING ATOMIC 143620 P 20
DEPRESSURIZATION RESEARCH, BLOCKED COOLING CHANNELS WITH 143620 P 20
DERIVED FROM PARTIAL HYPERBOLICS (IN FRENCH) 143630 P 7
DES FAUX POWER PLANT ACCORDING TO INFLUENTIAL PARAMETERS 143630 P 6
DESCRIBE THERMAL NON-EQUILIBRIUM FLOW OF WATER IN SIMPLE 143640 P 22
DESCRIBING THE THERMAL AND MECHANICAL BEHAVIOR OF A PWR 143670 P 10
DESCRIPTION OF EQUATIONS FOR HYDRODYNAMIC ANALYSIS (IN 143750 P 20
DESCRIPTION OF THE FUEL ROD BEHAVIOR DURING LOCA (IN 143810 P 6
DETECTION AND STRAIN EVALUATION IN PRESSURE VESSELS #LASER 143840 P 31
DETERMINATION OF PRESSURE TRANSIENTS IN FULL-PRESSURE 143850 P 10
DETERMINATION OF THE HEAT TRANSFER COEFFICIENT IN THE 143850 P 10
DETERMINATION OF HYDROGEN-AIR MIXTURES AND PROPAGATION 143880 P 10
DETERMINATION OF SPHERICAL AIR-HYDROGEN GASEOUS MIXTURES (IN 143910 P 6
DEVICE PROTECTION SYSTEM FOR REACTOR COMPONENTS (IN GERMAN) 143930 P 20
DIFFERENT MASS AND VELOCITY (IN GERMAN) MATERIALS AND 143930 P 13
DIMENSIONAL CALCULATION OF A WATER-VAPOR NOZZLE FLOW (IN 143930 P 19
DIMENSIONAL CALCULATION OF THE WATER POOL-SWELL IN THE 143950 P 10

#PLUST-23 - A COMPUTER CODE FOR THE CALCULATION OF THE TWO-DIMENSIONAL FLOW OF A PRESSURIZED WATER REACTOR

JAPANESE** #FUEL COOLANT INTERACTION EXPERIMENT BY BLOWDOWN OF A PART OF THE LOOP (MUSK) INCLUDING A 90 DEGREE TEE

FOLLOWING A #EXPERIMENTAL INVESTIGATION OF THE HYDROGEN FOLLOWING A #EXPERIMENTAL INVESTIGATION OF THE HYDROGEN IN LIGHT WATER REACTOR

OF CONDENSED WATER A REVIEW OF THE ARGUMENTS

CIRCUITS IN WATER-COOLED REACTORS, SUPPLEMENTAL RESEARCH DOCUMENTATION

FOR LOCA (IN JAPANESE)** #EXPERIMENT OF THE LATEST HIGH-TEMPERATURE CALCULATIONS USING THE COMPUTER CODE FOR CORE SUPPORT STRUCTURE DUE TO THE FURNACE OF MELT UP AFTER THE AFTER FAILURE OF THE CORE SUPPORT STRUCTURE AND WAITING FOR PERIODIC TESTING-NONNEGOTIABLE TEST OF THE HEAT TRANSFER EFFICIENT IN THE CONTAINMENT FOR ANALYZING THE THERMO-HYDRAULIC BEHAVIOR IN A PWR CORE THE THERMAL AND MECHANICAL BEHAVIOR OF A PWR FUEL ROD DURING A LOSS-OF-COOLANT ACCIDENT AND INTERACTION WITH THE DURING A RAPID FAILURE (BLOWDOWN) COMPARISON OF SEVERAL DURING A RELOAD PHASE OF PWR LOCA (IN JAPANESE)**

PRELIMINARY EMPIRICAL DESCRIPTION OF THE FUEL ROD BEHAVIOR DURING THE RELOADING PHASE OF A PWR CORE (IN FRENCH)**

IN WATER-COOLED REACTORS, SUPPLEMENTAL RESEARCH DOCUMENTATION

DIMENSIONAL FLOW OF A COMPRESSIBLE MEDIUM IN EQUIPPED 146750 P 40

DIMENSIONAL FLUID-STRUCTURE INTERACTIONS DURING BLOWDOWN 146904 P 40

DIMENSIONAL VISCOUS FLOW USING FINITE ELEMENT METHOD (IN 147752 P 40

TWO-PHASE FLOWS (IN FRENCH)** 148320 P 4

DIRECT ELECTRICAL HEATING METHOD (ZRIC)2- H2O SYSTEM (IN 148844 P 4

DIRECT HEATED ROD BUNDLE (IN FRENCH)** 149244 P 4

DISCHARGE MODEL APPLIED TO MULTI-COMPONENT, TWO-PHASE 149483 P 23

DISPERSION IN WATER AND AIR (IN ENGLISH)** 149801 P 6

DISPERSSION PROBLEME (STATION AND CYCLE TURNING) (IN FRENCH)** 149871 P 6

DISTRIBUTION IN A LIGHT WATER REACTOR CONTAINMENT 149882 P 11

DISTRIBUTION IN A LIGHT WATER REACTOR CONTAINMENT 149886 P 12

DISRUPTION AND ERROR RECONSTRUCTION WITH HELP OF TIME-UP- 149888 P 13

DO ELECTRONIC TRANSITIONS CONTRIBUTE TO THE THERMODYNAMICS 149894 P 24

DOCUMENTATION DIS (IN GERMAN)** OF THE PRIMARY COOLING 149897 P 11

DOSE REDUCTION (IN GERMAN)** 149910 P 44

DOWNGRADER EFFECTIVE WATER HEAT DURING A RELOAD PHASE OF 149913 P 40

DURFAN* (IN GERMAN)** #RESULTS OF THE LOFT EXPERIMENT 149968 P 20

DUE TO PRESSURE VESSEL FAILURE, PART II (IN GERMAN)** OF 149700 P 15

DUE TO THE FURNACE OF MELT UP DUE TO PRESSURE VESSEL 149706 P 40

DURATION, TEST EFFICIENCY NOT 100% AND 1 OUT OF 2 STANDBY 149796 P 1

DURING A COOLING SYSTEM BLOWDOWN (IN GERMAN) (ZRIC)2 149829 P 10

DURING A LOCA (IN ENGLISH)** #SCOUT-12 A COMPUTER PROGRAM 149829 P 47

DURING A LOCA (IN FRENCH)** #SCOUT-12 A CODE DESCRIBING 149871 P 1

DURING A LOSS-OF-COOLANT ACCIDENT AND INTERACTION WITH THE 149896 P 19

DURING A RAPID FAILURE (BLOWDOWN) COMPARISON OF SEVERAL 149791 P 4

DURING A RELOAD PHASE OF PWR LOCA (IN JAPANESE)** 149893 P 48

DURING BLOWDOWN OF A PRESSURIZED WATER REACTOR-FLUX (IN 149898 P 46

DURING LOCA (IN GERMAN)** # 149910 P 42

DURING THE RELOADING PHASE OF A PWR CORE (IN FRENCH)** 149920 P 1

DIS (IN GERMAN)** RUPTURE OF THE PRIMARY COOLING CIRCUIT 149894 P 16

DIS (IN GERMAN)RUPTURE OF THE PRIMARY COOLING CIRCUITS IN 149897 P 11

OF NUCLEAR POWER PLANTS (NPPs) AGAINST EXTERNAL EVENTS: 149138 P 5

MODEL OF THE FAILURE RATES OF THE VALVES OF ST. LAURENT DES 149139 P 6

LARGEST COLD LEG #MISA-II TEST DATA REPORT 12 EFFECTS OF 149391 P 27

INTEENSIVE TRAINING AND #RECORDING AND SYNTHESIS OF AN 149768 P 41

FAILURE BEHAVIOR (EMBELEMENT PARAMETER TEST IN NSRR) (IN 149374 P 28

IN JAPANESE)** #EXPERIMENT OF THE DOWNGRADER 149595 P 5

PHENOMENON IN LARGEST COLD LEG #MISA-II TEST DATA REPORT 12 149391 P 27

IN-SITU TESTING OF HIGH 149577 P 11

FOR PERIODIC TESTING-NONNEGOTIABLE TEST DUPLICATION, TEST 149296 P 1

INTERACTIONS DURING BLOWDOWN OF A PRESSURIZED WATER 149804 P 40

JAPANESE** #FUEL COOLANT INTERACTION EXPERIMENT BY DEFECT 148844 P 4

OF THE BLOWDOWN - EXPERIMENT (RS 104 (LUG)) CONTROL OF THE 148894 P 19

CONDENSED WATER A REVIEW OF THE ARGUMENTS #DO 148845 P 23

INTERACTION WITH THE EMERGENCY CORE COOLING (IN #FUEL 147752 P 40

METHOD FOR TWO DIMENSIONAL VISCOUS FLOW USING FINITE 148846 P 40

OF FRENCH INSTALLATIONS FOR THE STORAGE OF IRRADIATED FUEL 148894 P 19

EXPERIMENTAL TESTS ON BUCKLING OF 148845 P 23

COOLING CHANNELS WITH 90° GEOMETRY (IN GERMAN)** #ELECTRONIC CONTRIBUTION TO THE THERMODYNAMICS OF MELTEN 148846 P 19

DURING A LOSS-OF-COOLANT ACCIDENT AND INTERACTION WITH THE 149396 P 14

EMBELEMENT IN THE MEASUREMENT TECHNIQUES OF SONIC- 149375 P 4

PRELIMINARY EXPERIMENT ON LEAKAGE MEASUREMENTS USING SONIC 147103 P 3

MATERIALS USED IN FAST SODIUM-COOLED REACTORS (IN #SONIC- 149870 P 4

INNOVATION OF SIMULATED SONIC 148296 P 8

LOCA (IN GERMAN)** #PRELIMINARY 149396 P 19

#ATTITUDES ON QUESTIONS PERTAINING TO NUCLEAR 149198 P 11

OF THE ADDITIONAL FISSION PRODUCT RELEASE PHENOMENA (IN 148872 P 16

PRESSURE AND DENSITY AS THE INDEPENDENT VARIABLES (IN 148205 P 20

II EXPERIMENTS FOR COLD LEG BREAK RUNS 413 AND 312) 149196 P 16

FAILURE BEHAVIOR (EMBELEMENT PARAMETER TEST IN NSRR) (IN 149196 P 16

IN LARGEST COLD LEG BREAKS (MUSK 302, 413, 422) (IN 149196 P 16

IN GERMAN)** #SUPER CANON EXPERIMENTS (IN 149196 P 16

#RISK OF HAZARD PROCESSES FOR RELIABILITY PROBLEMS (IN 149196 P 16

#PATRIC, A COMPUTER CODE FOR FAULT TREE CALCULATIONS (IN 149196 P 16

#THE SAFETY OF NUCLEAR REACTORS IN FRANCE (IN 149196 P 16

#EXPERIMENTAL STUDY OF CRITICAL TWO-PHASE FLOW (IN 149196 P 16

#STUDY OF POLLUTANT DISPERSION IN WATER AND AIR (IN 149196 P 16

#THE PRACTICE OF QUALITY ASSURANCE BY FRACTURE (IN 149196 P 16

#REFLEXIONS ANALYSIS OF THE RELOAD EXPERIMENTS (RS 62) (IN 149196 P 16

#ANALYSIS OF BOILING-WATER REACTOR STEAM CHUGGING (IN 149196 P 16

#FAULT TREE ANALYSIS WITH MULTISTATE COMPONENTS (IN 149196 P 16

#PROBABILITY EVALUATION OF A LWR PRESSURE VESSEL (IN 149196 P 16

OF EVENTS FOR A PROBABILISTIC EVALUATION OF PWR SAFETY (IN 149196 P 16

WIND SHIELD MODEL FOR THE GERMAN REACTOR SAFETY STUDY (IN 149196 P 16

OF LOFT L1-2 EXPERIMENT BY ALARM-PI COMPUTER CODE (IN 149196 P 16

AT NON-STEADY STRESS AND TEMPERATURE LOADING CONDITIONS (IN 149196 P 16

TRANSIENT DOWNDUT IN AN ANNULAR TEST SECTION (IN JAPANESE 149196 P 16

PROGRAM FOR PRESSURIZED WATER REACTOR BLOWDOWN ANALYSIS (IN 149196 P 16

OF FLOW SODIUM BOILING EXPERIMENT IN A 19-PIN BUNDLE (IN 149196 P 16

OF THE NON-NUCLEAR LOFT TEST L1-5 (IN GERMAN AND 149196 P 16

EARTHQUAKES, FIRES, EXPLOSIONS AND AIRCRAFT CRASHES (IN 149138 P 5

EAUX FROIDES PLANT ACCORDING TO INFLUENTIAL PHENOMENA** # 149139 P 6

ECCS INJECTION AND PUMP CIRCULATION ON LOCA PHENOMENA IN 149391 P 27

EDUCATIONAL SYSTEM USING COMBINATION OF MEDIA FOR THE 149768 P 41

EFFECT OF HEAT GENERATION PROFILE IN PELLET ON FUEL 149374 P 28

EFFECT OF THERMAL CYCLING (IN ENGLISH)** #ANALYSIS 149595 P 5

EFFECTIVE WATER HEAT DURING A RELOAD PHASE OF PWR LOCA I 149391 P 27

EFFECTS OF ECCS INJECTION AND PUMP CIRCULATION ON LOCA 149374 P 28

EFFICIENCY FILTERS AT AEE WENFITH* 149577 P 11

EFFICIENCY NOT 100%, AND 1 OUT OF 2 STANDBY SYSTEM (IN 149296 P 1

EFFICIENT COMPUTATION OF THREE-DIMENSIONAL FLUID-STRUCTURE 149804 P 40

ELECTRICAL HEATING METHOD (ZRIC)2- H2O SYSTEM (IN 148844 P 4

ELECTRICAL HEATING POWER (IN GERMAN)** 148894 P 19

ELECTRONIC CONTRIBUTION TO THE THERMODYNAMICS OF MELTEN 148845 P 23

ELECTRONIC CONTRIBUTION TO THE THERMODYNAMICS OF MELTEN 147752 P 40

ELECTRONIC TRANSITIONS CONTRIBUTE TO THE THERMODYNAMICS OF 148846 P 19

ELEMENT BEHAVIOR DURING A LOSS-OF-COOLANT ACCIDENT AND 149396 P 14

ELEMENT METHOD (IN FRENCH)** #CALCULATIONAL 149375 P 4

ELEMENTS FROM LIGHT WATER REACTORS (IN FRENCH)THE SAFETY 147103 P 3

ELLIPSOIDAL VESSEL HEADS UNDER INTERNAL PRESSURE (IN 149870 P 4

EMERGENCY COOLING DEPRESSURIZATION RESEARCH, BUCREC 148296 P 8

EMERGENCY CORE COOLING (IN GERMAN)** #FUEL ELEMENT BEHAVIOR 149396 P 19

EMISSON ANALYSIS (SEA) (IN GERMAN)** 149198 P 11

EMISSON ANALYSIS: EXPANDED INSTRUMENTATION AND EVALUATION 148872 P 16

EMISSON MEASUREMENTS IN FRACTURE MECHANICS RESEARCH ON 148205 P 20

EMISSON-IMPULSES IN THICK WALLED STRUCTURES (IN GERMAN)** 149196 P 16

EMPIRICAL DESCRIPTION OF THE FUEL ROD BEHAVIOR DURING 149196 P 16

ENERGY: PLUTONIUM (IN GERMAN)** 149196 P 16

ENGLISH & JAPANESE** #AN ANALYSIS 149523 P 27

ENGLISH & JAPANESE** FOR GENERATING STEAM TABLES HAVING 149527 P 28

ENGLISH & JAPANESE** WITH RELAP5 CODE (ANALYSIS OF RUSA- 149534 P 27

ENGLISH & JAPANESE** GENERATION PROFILE IN PELLET ON FUEL 149374 P 28

ENGLISH & JAPANESE** PUMP CIRCULATION ON LOCA PHENOMENA 149391 P 27

ENGLISH** 149391 P 2

ENGLISH** 149377 P 3

ENGLISH** 149378 P 4

ENGLISH** 149103 P 6

ENGLISH** 149390 P 7

ENGLISH** 149801 P 6

ENGLISH** 148665 P 9

ENGLISH** 149393 P 14

ENGLISH** 149805 P 17

ENGLISH** 149806 P 20

ENGLISH** #FRACTURE 149870 P 5

ENGLISH** #SELECTION 149175 P 5

ENGLISH** #A PROPOSED 149806 P 23

ENGLISH** #AN ANALYSIS 149526 P 27

ENGLISH** #CREEP RUPTURE 149390 P 19

ENGLISH** #FLOW REDUCTION 149844 P 25

ENGLISH** #ALARM-PI: A COMPUTER 149702 P 20

ENGLISH** #ACOUSTIC NOISES WITH LOSS- 149776 P 26

ENGLISH** #POST-EXPERIMENT CALCULATION 149674 P 16

IN THE SAFETY EVALUATION OF NUCLEAR INSTALLATIONS (IN ENGLISH)	ENGLISH	#USE OF PROBABILISTIC METHODS	147001 P 0
IN 37-PIN SODIUM COOLED BUNDLES WITH GRID SPACERS (IN ENGLISH)	ENGLISH	#LOCAL FLOW BLOCKAGE EXPERIMENTS	143907 P 25
AND CONFINED SPHERICAL STRATIFIED GASEOUS MIXTURES (IN ENGLISH)	ENGLISH	#FLAME PROPAGATION THROUGH LACUNATED	143078 P 8
FROM A DEFECTED FUEL ROD - EFFECT OF THERMAL CEILING (IN ENGLISH)	ENGLISH	#ANALYSIS OF THE FISSION PRODUCT RELEASE	144050 P 8
DETONATION OF SPHERICAL AIR-HYDROGEN GASEOUS MIXTURES (IN ENGLISH)	ENGLISH	#STUDY OF THE OVERPRESSURE GENERATED BY THE	143138 P 5
EARTHQUAKES, FIRES, EXPLOSIONS AND AIRCRAFT CRASHES (IN ENGLISH)	ENGLISH	#POWER PLANTS (NPPS) AGAINST EXTERNAL EVENTS	143138 P 5
THERMO-HYDRAULIC BEHAVIOR IN A PWR CORE DURING A LOCA (IN ENGLISH)	ENGLISH	#ASUT-1: A COMPUTER PROGRAM FOR ANALYZING THE	147409 P 27
AND TECHNOLOGY OF THE FEDERAL REPUBLIC OF GERMANY, 1978 (IN ENGLISH)	ENGLISH	#PROJECTS SPONSORED BY THE MINISTRY FOR RESEARCH	143775 P 12
THE PROTECTION SYSTEM OF THE FESSENHEIM 1 PWR REACTOR (IN ENGLISH)	ENGLISH	#EVENT PROBLEM BY THE STUDY OF THE RELIABILITY OF	143775 P 12
CORROSION ON THE MECHANICAL PROPERTIES OF Zr-Y-TURBINE (IN ENGLISH)	ENGLISH	#IRRADIATED WASTE FUEL: INFLUENCE OF THE INTERNAL	144073 P 14
OPERATING SYSTEM MVS (WITH FORTRAN-H-EXTENDED COMPILER) (IN ENGLISH)	ENGLISH	#CASE-CODE TRAC-PI (VERSION 16.3) TO IOM STANDARD	143956 P 14
REPORT PERIOD OCTOBER 1 - DECEMBER 31, 1978 (IN GERMAN)	GERMAN	#REACTOR SAFETY RESEARCH PROGRAMS OF DMFT, USARC,	144004 P 10
HEAT GENERATION PROFILE IN PELLET ON FUEL FAILURE BEHAVIOR (IN ENGLISH)	ENGLISH	#RESEARCH PROGRAMS OF DMFT, USARC, EPRI, AND JSA,	144613 P 17
WITH A PWR FOLLOWING A LOCA FOR PURPOSES OF STUDYING THE (IN ENGLISH)	ENGLISH	#ENRICHMENT PARAMETER TEST IN NSRR) (IN ENGLISH & JAPANESE)	143974 P 18
HYDROGEN-AIR MIXTURES AND PROPAGATION OF SHOCK WAVES IN THE (IN ENGLISH)	ENGLISH	#ENRICHMENT DECONTAMINATION AND TRANSPORT (IN GERMAN)	144795 P 24
FROM THE REACTOR SAFETY RESEARCH PROGRAMS OF DMFT, USARC, (IN ENGLISH)	ENGLISH	#ENVIRONMENT (IN GERMAN)	144280 P 12
A CALCULATIONAL PROGRAM FOR THE SOLUTION OF THE NEUTRONIC (IN ENGLISH)	ENGLISH	#INITIATION OF DECONTAMINATION OF	144004 P 16
CHARACTERISTICS AND RESOLUTIONS OF THE SYSTEM OF (IN ENGLISH)	ENGLISH	#EPRI AND JSA, REPORT PERIOD OCTOBER 1 - DECEMBER 31,	144013 P 17
COMPONENT, 2D-PHASE SYSTEMS - THE (IN ENGLISH)	ENGLISH	#EQUATION OF A MULTIDIMENSIONAL HGR MODEL (IN GERMAN)	144072 P 17
BUBBLE (IN ENGLISH)	ENGLISH	#EQUATIONS DERIVED FROM PARTIAL HYPERBOLIC (IN FRENCH)	147104 P 7
GERMAN)	GERMAN	#EQUATIONS FOR HYDRODYNAMIC ANALYSIS (IN FRENCH)	147104 P 7
FINAL REPORT (IN GERMAN)	GERMAN	#EQUILIBRIUM CRITICAL DISCHARGE MODEL APPLIED TO MULTI-	144043 P 21
IN GERMAN)	GERMAN	#EQUILIBRIUM FLOW OF WATER IN SIMPLE PIPE SYSTEMS	144043 P 21
#DISTURBANCE AND	GERMAN	#ERROR ANALYSIS OF THE AMPLITUDE CURVE, FINAL REPORT (IN	144208 P 13
#THE	GERMAN	#ERROR RECONSTRUCTION WITH HELP OF TIME-OF-FLIGHT DATA,	144208 P 13
#FOR THE ATTAINMENT OF INTERPRETATIONAL MODELS FOR THE (IN ENGLISH)	ENGLISH	#ESSENTIAL SAFETY ASPECTS OF A CONFINED NUCLEAR FUEL CIRC.	147102 P 13
SPECKLE PHOTOGRAPHY METHODS - ON DEFECT DETECTION AND STRAIN (IN ENGLISH)	ENGLISH	#ESTIMATES OF PRESSURES AND YIELDS	144043 P 21
#FRACTURE PROBABILITY	ENGLISH	#ESTIMATION OF FLOW USING ULTRASONIC TESTS ON NUCLEAR	144046 P 13
#USE OF PROBABILISTIC METHODS IN THE SAFETY (IN ENGLISH)	ENGLISH	#EVALUATION IN PRESSURE VESSELS	144043 P 21
#SELECTION OF EVENTS FOR A PROBABILISTIC (IN ENGLISH)	ENGLISH	#EVALUATION OF A LWR PRESSURE VESSEL (IN ENGLISH)	143070 P 5
CALCULATIONAL PROGRAM (IN GERMAN)	GERMAN	#EVALUATION OF NUCLEAR INSTALLATIONS (IN ENGLISH)	147043 P 0
USING SONIC EMISSION ANALYSIS; EXPANDED INSTRUMENTATION AND (IN ENGLISH)	ENGLISH	#EVALUATION OF PWR SAFETY (IN ENGLISH)	147170 P 14
PROTECTION SYSTEM OF THE (IN ENGLISH)	ENGLISH	#EVALUATION OF THE 20-MGD BUNDLE TEST (RS-32C) WITH THE	143902 P 14
CORE MELTDOWN ACCIDENT: CALCULATION TO (IN ENGLISH)	ENGLISH	#EVALUATION PROGRAM (IN GERMAN)	144072 P 17
PROTECTION OF NUCLEAR POWER PLANTS (NPPS) AGAINST EXTERNAL (IN ENGLISH)	ENGLISH	#ON LEAKAGE MONITORING	143775 P 7
REACTORS, PHASE II, WATER INGRESS, AIR INGRESS, REACTIVITY (IN ENGLISH)	ENGLISH	#EVENT PROBLEM BY THE STUDY OF THE RELIABILITY OF THE	147170 P 14
ON LEAKAGE MONITORING USING SONIC EMISSION ANALYSIS (IN ENGLISH)	ENGLISH	#EVENTS FOR A PROBABILISTIC EVALUATION OF PWR SAFETY (IN	143775 P 7
#RESEARCH PROGRAM ON REACTOR SAFETY, 3D (IN ENGLISH)	ENGLISH	#EVENTS IN THE CONCRETE - FAILURE PHASE OF THE HYDROLYTICAL	143775 P 7
PRESSURE TRANSIENT OF A NUCLEAR POWER PLANT IN THE LARGE (IN ENGLISH)	ENGLISH	#FACETS: EARTHQUAKES, FIRES, EXPLOSIONS AND AIRCRAFT	143138 P 5
N2O SYSTEM) (IN JAPANESE)	JAPANESE	#EXAMINATION OF 3 HSE PLATES RUPTURED IN AIR (IN GERMAN)	144795 P 24
IN GERMAN AND ENGLISH)	GERMAN	#EXCURSIONS (IN GERMAN)	144070 P 10
COOLING CIRCUIT IN WATER-COOLED REACTORS, QUICK LOOK REPORT (IN ENGLISH)	ENGLISH	#EXPANDED INSTRUMENTATION AND EVALUATION PROGRAM (IN GERMAN)	144072 P 10
ACOUSTIC NOISES WITH LOSS-OF-FLOW SODIUM BOILING (IN ENGLISH)	ENGLISH	#EXPERIENCE SUPER-CANON (IN FRENCH)	144072 P 10
CODE "MORVAN" (IN GERMAN)	GERMAN	#EXPERIMENT (IN GERMAN)	144072 P 10
REFLOOD PHASE OF PWR LOCA (IN JAPANESE)	JAPANESE	#EXPERIMENT (IN GERMAN)	144072 P 10
ANALYSIS; EXPANDED INSTRUMENTATION AND (IN ENGLISH)	ENGLISH	#EXPERIMENT AREA OF THE GKSS (IN GERMAN)	144072 P 10
POWER (IN GERMAN)	GERMAN	#EXPERIMENT BY MARK-PI COMPUTER CODE (IN ENGLISH)	144072 P 10
HYDRAULIC BEHAVIOR IN THE INITIAL BLOWDOWN PHASE, PARTS A, (IN ENGLISH)	ENGLISH	#EXPERIMENT BY DIRECT ELECTRICAL HEATING METHOD (ZK1022-	144072 P 10
SECONDARY SIDE OF A STEAM GENERATOR (IN FRENCH)	FRENCH	#EXPERIMENT CALCULATION OF THE NON-NUCLEAR LOFT TEST L1-5 (144072 P 10
EFFICIENT IN THE CONTAINMENT DURING A COOLING SYSTEM (IN ENGLISH)	ENGLISH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
A LIGHT WATER REACTOR CONTAINMENT FOLLOWING A LOSS-OF- (IN ENGLISH)	ENGLISH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
A LIGHT WATER REACTOR CONTAINMENT FOLLOWING A LOSS-OF- (IN ENGLISH)	ENGLISH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
PRIMARY SYSTEM OF PRESSURIZED WATER REACTORS (IN GERMAN)	GERMAN	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
UNDER ACCIDENT (STATUS AND RESULTS OF THE THEORETICAL AND (IN ENGLISH)	ENGLISH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
IN THE PRESSURE TRANSIENT OF A NUCLEAR POWER PLANT IN THE (IN ENGLISH)	ENGLISH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
DETONATION OF SPHERICAL AIR-HYDROGEN GASEOUS MIXTURES (IN ENGLISH)	ENGLISH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
ON REINFORCED CONCRETE, CALCULATION AND COMPARISON WITH (IN ENGLISH)	ENGLISH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
UNDER INTERNAL PRESSURE (IN FRENCH)	FRENCH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
ROOM TEMPERATURE (IN FRENCH)	FRENCH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
#SUPER CANON	GERMAN	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
#REFLEXIONS ANALYSIS OF THE REFLOOD (IN ENGLISH)	ENGLISH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
#THE CONTAINMENT TEST FACILITY (IN ENGLISH)	ENGLISH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
IN WATER-COOLED REACTORS, CONDENSATION IN CONTAINMENT BY (IN ENGLISH)	ENGLISH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
OF LOCA EXPERIMENTS WITH RELAP-4 CODE (ANALYSIS OF ROSA-II (IN ENGLISH)	ENGLISH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
SPACERS (IN ENGLISH)	ENGLISH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
PART 2 (IN GERMAN)	GERMAN	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
EXPERIMENTS FOR COLD LEG BREAK RUNS 413 (IN ENGLISH)	ENGLISH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
PLANTS (NPPS) AGAINST EXTERNAL EVENTS: EARTHQUAKES, FIRES, (IN ENGLISH)	ENGLISH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
16.3) TO IOM STANDARD OPERATING SYSTEM MVS (WITH FORTRAN-H- (IN ENGLISH)	ENGLISH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
AIRCRAFT #PROTECTION OF NUCLEAR POWER PLANTS (NPPS) AGAINST (IN ENGLISH)	ENGLISH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
EXTERNAL EVENTS: EARTHQUAKES, FIRES, EXPLOSIONS AND	ENGLISH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
#THE CONTAINMENT TEST FACILITY (EXPERIMENTS C AND D) (IN GERMAN)	GERMAN	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
FAILURE (BLOWDOWN) COMPARISON OF SEVERAL MODELS (IN FRENCH)	FRENCH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
FAILURE BEHAVIOR (ENRICHMENT PARAMETER TEST IN NSRR) (IN ENGLISH)	ENGLISH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10
FAILURE OF THE CORE SUPPORT STRUCTURE DUE TO THE FORMATION	ENGLISH	#EXPERIMENT DTD (IN GERMAN)	144072 P 10

AND INTERACTION WITH THE EMERGENCY COOLING IN GERMANIA#	FUEL ELEMENT BEHAVIOR DURING A LOSS-OF-COOLANT ACCIDENT	142940 P 19
OF FRENCH INSTALLATIONS FOR THE STORAGE OF IRRADIATED	FUEL ELEMENTS FROM LIGHT WATER REACTORS (IN FRENCH) SAFETY	143160 P 2
IN ENGLISH. EFFECT OF HEAT GENERATION PROFILE IN PELLET ON	FUEL FAILURE BEHAVIOR ENRICHMENT PARAMETER TEST IN NGRM1	143174 P 20
ANALYSIS OF THE FISSION PRODUCT RELEASE FROM A DEFECTED	FUEL FLO - EFFECT OF THERMAL STAINING (IN ENGLISH)#	144090 P 5
THE THEORETICAL AND EXPERIMENTAL INVESTIGATIONS ON THE LWR	FUEL ROD BEHAVIOR UNDER REDUCED CONDITIONS (IN GERMANIA)#	142740 P 19
PRELIMINARY EXPERIMENTAL DESCRIPTION OF THE	FUEL ROD BEHAVIOR DURING LOCA (IN GERMANIA)	145150 P 22
DESCRIPTIONS THE THERMAL AND MECHANICAL BEHAVIOR OF A PWR	FUEL ROD DURING A LOCA (IN FRENCH)#	145071 P 1
OPERATING POWER RAMP#	FUEL ROD IRRADIATION TESTS 1576/77 (GERMANIA)	145070 P 23
OF A PRESSURIZED WATER REACTOR; PRIMARY CIRCUIT BY	FUEL RODS SHOWING MANUFACTURING FAULTS (IN FRENCH)#	141380 P 8
SIMULATION OF THE CHEMICAL STATE OF IRRADIATED URANIUM	FUEL INFLUENCE OF THE INTERNAL CORROSION ON THE	145070 P 23
CODE FOR THE DETERMINATION OF PRESSURE TRANSIENTS IN	FULL-PRESSURE CONTAINMENTS OF WATER-COOLED NUCLEAR POWER	145030 P 10
FRENCH)#		
#STUDY OF THE RANGE OF VELOCITY OF	GAS INSIDE A BUBBLE RISING THROUGH A LIQUID COOLANT (IN	143210 P 2
REDUPRENSIVE DAMAGE ANALYSIS FOR HIGH TEMPERATURE	WATER REACTORS, PHASE II, WATER INGRESS, AIR INGRESS,	145070 P 10
THROUGH UNCONFINED AND CONFINED HEMI-SPHERICAL STRATIFIED	GASELS MIXTURED (IN ENGLISH)#	143070 P 0
GENERATED BY THE DETONATION OF SPHERICAL AIR-HYDROGEN	GASELS MIXTURED (IN ENGLISH)#	144410 P 0
CHARGES MIXTURES I. EXPERIMENTAL STUDY OF THE OVERPRESSURE	GENERATED BY THE DETONATION OF SPHERICAL AIR-HYDROGEN	144410 P 0
INDEPENDENT VARIABLES (IN SPANISH) A COMPUTER SUBROUTINE FOR	GENERATING STEAM TABLES HAVING PRESSURE AND DENSITY AS THE	144527 P 20
ENRICHMENT PARAMETER TEST IN NGRM1 (IN	GENERATION PROFILE IN PELLET ON FUEL FAILURE BEHAVIOR (143974 P 20
EFFECT OF HEAT	GENERATOR (IN FRENCH)#	148204 P 0
STUDY OF THE BLOWDOWN OF THE SECONDARY SIDE OF A STEAM	GERMANY (IN GERMANIA)#	148256 P 23
DEPRESSURIZATION RESEARCH, BLOCKED COOLING CHANNELS WITH NGR	GERMANY 5 (ENGLISH)#	145011 P 17
AND DATA, REPORT PERIOD OCTOBER 1 - DECEMBER 31, 1979 (IN	GERMANIA & ENGLISH)#	144504 P 10
GRM1, USM1, SPM1 AND JSM1, JULY 1-SEPTEMBER 30, 1979 (IN	GERMANIA AND ENGLISH)#	145074 P 10
CALCULATION OF THE NON-NUCLEAR LOFT TEST LE-5 (IN	GERMANIA REACTOR SAFETY STUDY (IN ENGLISH)#	145000 P 21
IN A PROPOSED WIND SHEFT MODEL FOR THE	GERMANIA)	143800 P 10
WINE CONTAINMENT TEST FACILITY (EXPERIMENTS C AND D) (IN	GERMANIA)	146759 P 13
EXAMINATION OF 1 INSE PLATES RUPTURED IN AIR (IN	GERMANIA)	146763 P 13
MAGNETIC FLUX METHOD, FINAL REPORT (IN	GERMANIA)	148370 P 10
BERNHAR ANALYSIS OF THE AMPLITUDE CURVE, FINAL REPORT (IN	GERMANIA)	148369 P 17
BWR'S ANNUAL PROGRESS REPORT-1978 (IN	GERMANIA)	147754 P 19
SAFETY CONTAINMENT OF NUCLEAR POWER PLANTS (IN	GERMANIA)	148071 P 21
NUCLEAR SAFETY PROJECT FIRST SEMI-ANNUAL REPORT 1978 (IN	GERMANIA)	143083 P 22
RESEARCH PROGRAM ON REACTOR SAFETY, 3D EXPERIMENT (IN	GERMANIA)	144158 P 22
ACTIVATED CORROSION PRODUCTS IN LWR LOOPS (IN	GERMANIA)	145070 P 23
WASTE/POWER RAMP# FUEL ROD IRRADIATION TESTS 1976/77 (IN	GERMANIA)	143904 P 24
PHASE SEPARATION (IN	GERMANIA)	145063 P 10
ON QUESTIONS PERTAINING TO NUCLEONIC ENERGY FLUCTUATION (IN	GERMANIA)	144020 P 23
OF A SECTION SYSTEM FOR INSTALLATIONS AND FITTINGS (IN	GERMANIA)	145020 P 24
ASSESSMENT OF THE SECONDARY COOLANT ALIGNMENT OF A PWR (IN	GERMANIA)	145162 P 10
SAFETY ASPECTS OF A CONFINED NUCLEAR FUEL CYCLE (IN	GERMANIA)	145790 P 13
SYSTEM FOR THE GAS-MULTIPLE TUBE RESEARCH PROGRAM (IN	GERMANIA)	144158 P 11
MEASUREMENT TECHNIQUES OF SONIC EMISSION ANALYSIS (SERIAL	GERMANIA)	143952 P 14
RS FOR BURKE TEST (RS-37C) WITH THE CALCULATIONAL PROGRAM (IN	GERMANIA)	143054 P 10
ROD FOR (LOR) CONTROL OF THE ELECTRICAL HEATING POWER (IN	GERMANIA)	144158 P 22
DESCRIPTION OF THE FUEL ROD BEHAVIOR DURING LOCA (IN	GERMANIA)	145150 P 10
SONIC EMISSION IMPULSES IN THICK WALLED STRUCTURES (IN	GERMANIA)	143211 P 19
DIMENSIONAL CALCULATION OF A WATER-VAPOR NOZZLE FLOW (IN	GERMANIA)	145256 P 23
RESEARCH, BLOCKED COOLING CHANNELS WITH NGR GEOMETRY (IN	GERMANIA)	146756 P 11
OF RESEARCH PROJECT RS 50 (MODEL CONTAINMENT) PART 2 (IN	GERMANIA)	144157 P 25
FOR TRANSIENT END PHASE STATE USING ATOMIC RESONANCE (IN	GERMANIA)	145068 P 10
WITH HELP OF TIME-OF-FLIGHT DATA, FINAL REPORT (IN	GERMANIA)	145155 P 22
POST RUN CALCULATIONS USING THE COMPUTER CODE "SURFMAN" (IN	GERMANIA)	143900 P 15
SAFETY DEVICE PROTECTION SYSTEM FOR REACTOR COMPONENTS (IN	GERMANIA)	145030 P 10
LOFT-TESTS LE-4 (PWR) AND POST TEST CALCULATIONS (IN	GERMANIA)	145075 P 10
CONTAINMENTS OF WATER-COOLED NUCLEAR POWER PLANTS (IN	GERMANIA)	147473 P 10
A WATER COOLED REACTOR - INTERIM RESEARCH REPORT C 13 (IN	GERMANIA)	147473 P 10
SAFETY REPORTING PERIOD OCTOBER-DECEMBER 31, 1978 (IN	GERMANIA)	147017 P 23
LIMITING THE H ₂ -CONCENTRATION IN THE DWR CONTAINMENT (IN	GERMANIA)	147003 P 13
AND PROJECTS OF DIFFERENT MASS AND VELOCITY (IN	GERMANIA)	147017 P 23
LOFT COMPONENTS OF COMMISSIONED NUCLEAR POWER PLANTS (IN	GERMANIA)	147003 P 20
ON MATERIALS USED IN FAST SODIUM-COOLED REACTORS (IN	GERMANIA)	145074 P 12
TRANSIENT IN LIGHT-WATER AND HIGH TEMPERATURE REACTORS (IN	GERMANIA)	145038 P 21
CALCULATION AND COMPARISON WITH EXPERIMENTAL TESTS (IN	GERMANIA)	145160 P 10
THE AREA OF REACTOR SAFETY, JULY 1-SEPTEMBER 30, 1978 (IN	GERMANIA)	145046 P 11
OF FLOW USING ULTRASONIC TESTS ON NUCLEAR REACTORS (IN	GERMANIA)	148272 P 12
THE RESEARCH PROGRAM RS 50 (MODEL CONTAINMENT) PART 1 (IN	GERMANIA)	148297 P 10
IN CONTAINMENT BY EXPERIMENTS C04 AND C1 TO C16 (IN	GERMANIA)	144582 P 11
IN THE CONTAINMENT DURING A COOLING SYSTEM BLOWDOWN (IN	GERMANIA)	144586 P 12
A LOSS-OF-COOLANT ACCIDENT, QUICK LOOK REPORT 1 (IN	GERMANIA)	145070 P 17
A LOSS-OF-COOLANT ACCIDENT, QUICK LOOK REPORT 2 (IN	GERMANIA)	146791 P 24
THE NEUTRONIC EQUATION OF A MULTIDIMENSIONAL HTGR MODEL (IN	GERMANIA)	145040 P 10
OF STUDYING THE ENSURING DECONTAMINATION AND TRANSPORT (IN	GERMANIA)	145040 P 10
WATER-COOLED REACTORS, QUICK LOOK REPORT EXPERIMENT 015 (IN	GERMANIA)	145040 P 10
OF A COMPRESSIBLE MEDIUM IN COUPLED RECTANGULAR AREAS (IN	GERMANIA)	145040 P 10
REACTORS, SUPPLEMENTAL RESEARCH DOCUMENTATION DIS (IN	GERMANIA)	145040 P 11
TO IDENTIFY THE INFLUENCE OF VARIOUS PARAMETERS (IN	GERMANIA)	143379 P 15
OF OPERATING PERSONNEL AT NUCLEAR POWER PLANTS (IN	GERMANIA)	143768 P 21
IN THE PRIMARY SYSTEM OF PRESSURIZED WATER REACTORS (IN	GERMANIA)	144596 P 18
II, WATER INGRESS, AIR INGRESS, REACTIVITY EXCURSIONS (IN	GERMANIA)	148070 P 18
BEHAVIOR IN THE INITIAL BLOWDOWN PHASE, PARTS A, B, & C (IN	GERMANIA)	143901 P 24
IN WATER-COOLED REACTORS WITH REFRIGERATED CONDENSER (IN	GERMANIA)	143325 P 10
EXPANDED INSTRUMENTATION AND EVALUATION PROGRAM (IN	GERMANIA)	145074 P 10
DURING BLOWDOWN OF A PRESSURIZED WATER REACTOR-FLUX (IN	GERMANIA)	145040 P 20

- SWELL IN THE CONDENSATION CHAMBER OF A REACTOR SYSTEM (IN
AND PROPAGATION OF SHOCK WAVES IN THE ENVIRONMENT (IN
PLANTS AND THE COUPLING OF MELTIN-1 AND DELANZ-1), PART I (IN
AND INTERACTION WITH THE EMERGENCY CORE COOLING (IN
OF MELT DUE TO PRESSURE VESSEL FAILURE, PART II (IN
INFLUENCE OF THESE OCCURRENCES IN FAULT TREE ANALYSIS (IN
ON THE LOW FUEL ROD BEHAVIOR UNDER ACCIDENT CONDITIONS (IN
POWER PLANT IN THE LARGE EXPERIMENT AREA OF THE GKSS (IN
FOR RESEARCH AND TECHNOLOGY OF THE FEDERAL REPUBLIC OF
A NUCLEAR POWER PLANT IN THE LARGE EXPERIMENT AREA OF THE
BUCKLE EXPERIMENTS IN 17-PIN SODIUM-LOADED BUNDLES WITH
ENGLISH)* #REFLUX* #GRS ANNUAL PROGRESS REPORT-1978 (IN GERMAN)*
- 16.32 TO IBM STANDARD OPERATING SYSTEM MVS (WITH FORTRAN-
#MAPOL) A COMPUTER SUBROUTINE FOR GENERATING STEAM TABLES
#EXPERIMENTAL TESTS ON BUCKLING OF ELLIPTICAL VESSEL
#EXPERIMENT OF THE DOWNCOMER EFFECTIVE WATER
ENRICHMENT PARAMETER TEST IN NSRR) (IN ENGLISH)* #EFFECT OF
COOLING SYSTEM BLOWDOWN (EXPERIMENTAL DETERMINATION OF THE
REFLOODING PHASE OF A PWR'S CORE (IN FRENCH)* #FLOW AND
BLOWDOWN OF A PART OF THE LOOP OMEGA INCLUDING A 30 DIRECT
#FUEL-COOLANT INTERACTION EXPERIMENT BY DIRECT ELECTRICAL
- EXPERIMENT NS 109 (LODI) CONTROL OF THE ELECTRICAL
#DISTURBANCE AND ERROR RECONSTRUCTION WITH
#FLAME PROPAGATION THROUGH UNCONFINED AND CONFINED
#IN-SITU TESTING OF
AIR INGRESS, REACTIVITY #COMPREHENSIVE DAMAGE ANALYSIS FOR
AND THE SUBSEQUENT THERMAL TRANSIENT IN LIGHT-WATER AND
DETECTION AND STRAIN EVALUATION IN PRESSURE VESSELS* #LASER
TO MULTI-COMPONENT, TWO-PHASE SYSTEMS - THE #A SIMPLE
#EXAMINATION OF 3
SOLUTION OF THE NEUTRONIC EQUATION OF A MULTIDIMENSIONAL
NON-EQUILIBRIUM FLOW OF WATER IN SIMPLE PIPE SYSTEMS* #
OR MC III AT JULY 1976: A PROGRAM FOR TRANSIENT THERMAL-
#ASCOT-1: A COMPUTER PROGRAM FOR ANALYZING THE THERMO-
#S #EXPERIMENTAL AND THEORETICAL RESEARCH ON THE THERMAL
CODE FOR PWR (IN FRENCH)* #FRENCH THERMO-
OF VARIOUS SYSTEMS AND DESCRIPTION OF EQUATIONS FOR
FOLLOWING A LOSS-OF-# #EXPERIMENTAL INVESTIGATION OF THE
FOLLOWING A LOSS-OF-# #EXPERIMENTAL INVESTIGATION OF THE
OVERPRESSURE GENERATED BY THE DETONATION OF SPHERICAL AIR-
THE ENVIRONMENT (IN GERMAN)* #INITIATION OF DETONATION OF
RESOLUTIONS OF THE SYSTEM OF EQUATIONS DERIVED FROM PARTIAL
THE COURSE OF EVENTS IN THE CONCRETE - FAILURE PHASE OF THE
#INVESTIGATION AND DEVELOPMENT OF SYSTEMS LIMITING THE
EXPERIMENT BY DIRECT ELECTRICAL HEATING METHOD (2R1032-
#EXTENDED COMPILER) (IN ENGLISH)* #CODE TRAC-PI (VERSION
HAVING PRESSURE AND DENSITY AS THE INDEPENDENT VARIABLES (IN
HEADS UNDER INTERNAL PRESSURE (IN FRENCH)*
HEAT DURING A REFLOOD PHASE OF PWR LUCA (IN JAPANESE)*
HEAT GENERATION PROFILE IN PELLET ON FUEL FAILURE BEHAVIOR
HEAT TRANSFER COEFFICIENT IN THE CONTAINMENT DURING A
HEAT TRANSFER THERMOHYDRAULIC MODELISATION DURING THE
HEATED ROD BUNDLE (IN FRENCH)*
HEATING METHOD (2R1032- H2O SYSTEM) (IN JAPANESE)*
HEATING POWER (IN GERMAN)* #BLOWDOWN
HELP OF TIME-OF-FLIGHT DATA, FINAL REPORT (IN GERMAN)*
HEMISPHERICAL STRATIFIED GASEOUS MIXTURES (IN ENGLISH)*
HIGH EFFICIENCY FILTERS AT AEE WINDRIITH*
HIGH TEMPERATURE GAS REACTORS, PHASE II, WATER INGRESS*
HIGH TEMPERATURE REACTORS (IN GERMAN)* #BLOWDOWN ACCIDENT
HOLOGRAPHIC AND SPECKLE PHOTOGRAPHY METHODS FOR DEFECT
HOMOGENEOUS EQUILIBRIUM CRITICAL EXCHANGE MODEL APPLIED
MSST PLATES RUPTURED IN AIR (IN GERMAN)*
HTGR MODEL (IN GERMAN)* #A CALCULATIONAL PROGRAM FOR THE
#BUBLE BOUNCE III: A COMPUTER PROGRAM TO DESCRIBE INHERAL
HYDRAULIC ANALYSIS* #THE STATUS OF RELAP-
HYDRAULIC BEHAVIOR IN A PWR CORE DURING A LOCA (IN ENGLISH)*
HYDRAULIC BEHAVIOR IN THE INITIAL BLOWDOWN PHASE, PARTS A,
HYDRAULIC STUDIES FOR THE DEVELOPMENT OF SAFETY ADVANCED
HYDRODYNAMIC ANALYSIS (IN FRENCH)* #STUDY OF THE STABILITY
HYDROGEN DISTRIBUTION IN A LIGHT WATER REACTOR CONTAINMENT
HYDROGEN DISTRIBUTION IN A LIGHT WATER REACTOR CONTAINMENT
HYDROGEN GASEOUS MIXTURES (IN ENGLISH)* #STUDY OF THE
HYDROGEN-AIR MIXTURES AND PROPAGATION OF SHOCK WAVES IN
HYPERGOLICS (IN FRENCH)* #CHARACTERISTICS AND
HYPOTHETICAL CORE MELTDOWN ACCIDENTS: CALCULATION TO
H2-CONCENTRATION IN THE BWR CONTAINMENT (IN GERMAN)*
H2O SYSTEM) (IN JAPANESE)* #FUEL COOLANT INTERACTION
- 145757 P 10
144280 P 12
145408 P 25
143906 P 19
145750 P 20
143779 P 24
143745 P 19
140488 P 14
147407 P 18
140488 P 14
143907 P 20
143903 P 14
145309 P 17
- 16.32 TO IBM STANDARD OPERATING SYSTEM MVS (WITH FORTRAN-
#MAPOL) A COMPUTER SUBROUTINE FOR GENERATING STEAM TABLES
#EXPERIMENTAL TESTS ON BUCKLING OF ELLIPTICAL VESSEL
#EXPERIMENT OF THE DOWNCOMER EFFECTIVE WATER
ENRICHMENT PARAMETER TEST IN NSRR) (IN ENGLISH)* #EFFECT OF
COOLING SYSTEM BLOWDOWN (EXPERIMENTAL DETERMINATION OF THE
REFLOODING PHASE OF A PWR'S CORE (IN FRENCH)* #FLOW AND
BLOWDOWN OF A PART OF THE LOOP OMEGA INCLUDING A 30 DIRECT
#FUEL-COOLANT INTERACTION EXPERIMENT BY DIRECT ELECTRICAL
- EXPERIMENT NS 109 (LODI) CONTROL OF THE ELECTRICAL
#DISTURBANCE AND ERROR RECONSTRUCTION WITH
#FLAME PROPAGATION THROUGH UNCONFINED AND CONFINED
#IN-SITU TESTING OF
AIR INGRESS, REACTIVITY #COMPREHENSIVE DAMAGE ANALYSIS FOR
AND THE SUBSEQUENT THERMAL TRANSIENT IN LIGHT-WATER AND
DETECTION AND STRAIN EVALUATION IN PRESSURE VESSELS* #LASER
TO MULTI-COMPONENT, TWO-PHASE SYSTEMS - THE #A SIMPLE
#EXAMINATION OF 3
SOLUTION OF THE NEUTRONIC EQUATION OF A MULTIDIMENSIONAL
NON-EQUILIBRIUM FLOW OF WATER IN SIMPLE PIPE SYSTEMS* #
OR MC III AT JULY 1976: A PROGRAM FOR TRANSIENT THERMAL-
#ASCOT-1: A COMPUTER PROGRAM FOR ANALYZING THE THERMO-
#S #EXPERIMENTAL AND THEORETICAL RESEARCH ON THE THERMAL
CODE FOR PWR (IN FRENCH)* #FRENCH THERMO-
OF VARIOUS SYSTEMS AND DESCRIPTION OF EQUATIONS FOR
FOLLOWING A LOSS-OF-# #EXPERIMENTAL INVESTIGATION OF THE
FOLLOWING A LOSS-OF-# #EXPERIMENTAL INVESTIGATION OF THE
OVERPRESSURE GENERATED BY THE DETONATION OF SPHERICAL AIR-
THE ENVIRONMENT (IN GERMAN)* #INITIATION OF DETONATION OF
RESOLUTIONS OF THE SYSTEM OF EQUATIONS DERIVED FROM PARTIAL
THE COURSE OF EVENTS IN THE CONCRETE - FAILURE PHASE OF THE
#INVESTIGATION AND DEVELOPMENT OF SYSTEMS LIMITING THE
EXPERIMENT BY DIRECT ELECTRICAL HEATING METHOD (2R1032-
#EXTENDED COMPILER) (IN ENGLISH)* #CODE TRAC-PI (VERSION
HAVING PRESSURE AND DENSITY AS THE INDEPENDENT VARIABLES (IN
HEADS UNDER INTERNAL PRESSURE (IN FRENCH)*
HEAT DURING A REFLOOD PHASE OF PWR LUCA (IN JAPANESE)*
HEAT GENERATION PROFILE IN PELLET ON FUEL FAILURE BEHAVIOR
HEAT TRANSFER COEFFICIENT IN THE CONTAINMENT DURING A
HEAT TRANSFER THERMOHYDRAULIC MODELISATION DURING THE
HEATED ROD BUNDLE (IN FRENCH)*
HEATING METHOD (2R1032- H2O SYSTEM) (IN JAPANESE)*
HEATING POWER (IN GERMAN)* #BLOWDOWN
HELP OF TIME-OF-FLIGHT DATA, FINAL REPORT (IN GERMAN)*
HEMISPHERICAL STRATIFIED GASEOUS MIXTURES (IN ENGLISH)*
HIGH EFFICIENCY FILTERS AT AEE WINDRIITH*
HIGH TEMPERATURE GAS REACTORS, PHASE II, WATER INGRESS*
HIGH TEMPERATURE REACTORS (IN GERMAN)* #BLOWDOWN ACCIDENT
HOLOGRAPHIC AND SPECKLE PHOTOGRAPHY METHODS FOR DEFECT
HOMOGENEOUS EQUILIBRIUM CRITICAL EXCHANGE MODEL APPLIED
MSST PLATES RUPTURED IN AIR (IN GERMAN)*
HTGR MODEL (IN GERMAN)* #A CALCULATIONAL PROGRAM FOR THE
#BUBLE BOUNCE III: A COMPUTER PROGRAM TO DESCRIBE INHERAL
HYDRAULIC ANALYSIS* #THE STATUS OF RELAP-
HYDRAULIC BEHAVIOR IN A PWR CORE DURING A LOCA (IN ENGLISH)*
HYDRAULIC BEHAVIOR IN THE INITIAL BLOWDOWN PHASE, PARTS A,
HYDRAULIC STUDIES FOR THE DEVELOPMENT OF SAFETY ADVANCED
HYDRODYNAMIC ANALYSIS (IN FRENCH)* #STUDY OF THE STABILITY
HYDROGEN DISTRIBUTION IN A LIGHT WATER REACTOR CONTAINMENT
HYDROGEN DISTRIBUTION IN A LIGHT WATER REACTOR CONTAINMENT
HYDROGEN GASEOUS MIXTURES (IN ENGLISH)* #STUDY OF THE
HYDROGEN-AIR MIXTURES AND PROPAGATION OF SHOCK WAVES IN
HYPERGOLICS (IN FRENCH)* #CHARACTERISTICS AND
HYPOTHETICAL CORE MELTDOWN ACCIDENTS: CALCULATION TO
H2-CONCENTRATION IN THE BWR CONTAINMENT (IN GERMAN)*
H2O SYSTEM) (IN JAPANESE)* #FUEL COOLANT INTERACTION
- 143906 P 14
144027 P 20
146070 P 8
143093 P 20
143974 P 20
146270 P 10
143228 P 1
143929 P 4
148042 P 29
143094 P 10
148068 P 13
143076 P 8
146077 P 21
146070 P 18
146074 P 12
120343 P 21
146483 P 23
146799 P 13
146072 P 17
140447 P 22
146040 P 21
144529 P 27
143901 P 24
143758 P 6
143750 P 2
144520 P 11
144520 P 12
144280 P 12
144410 P 6
144280 P 12
147101 P 7
143779 P 15
143773 P 23
148042 P 29
- THE CONVERSION OF THE LASL-CODE TRAC-PI (VERSION 16.3) TO
OF THE HYPOTHETICAL CORE MELTDOWN ACCIDENTS: CALCULATION TO
DAMAGE ANALYSIS FOR HIGH TEMPERATURE GAS REACTORS, PHASE
FLOW OF WATER IN SIMPLE PIPE SYSTEMS* #BUBLE BOUNCE
OF SPECIFIC REACTOR MATERIALS AND COMPONENT PARTS AT
CALCULATION AND COMPARISON WITH EXPERIMENTAL TESTS (IN
EMISSION ANALYSIS (SEA) (IN GERMAN)*
#PROPAGATION OF DEMILATED SONIC EMISSION
#BLOWDOWN OF A PART OF THE LOOP OMEGA
GENERATING STEAM TABLES HAVING PRESSURE AND DENSITY AS THE
#SIMULATION OF THE CHEMICAL STATE OF IRRADIATED OXIDE FUELS
OF THE FAILURE RATE OF COMPONENTS AND THE APPARENT
CORE MELTDOWN ACCIDENT: CALCULATION TO IDENTIFY THE
THE VALUES OF ST. LAURENT DES EAUX POWER PLANT ACCORDING TO
ANALYSIS FOR HIGH TEMPERATURE GAS REACTORS, PHASE II, WATER
HIGH TEMPERATURE GAS REACTORS, PHASE II, WATER INGRESS, AIR
CLEARANCE PATTERNS* #ASSESSMENTS OF RISK FOLLOWING THE
RESEARCH ON THE THERMAL HYDRAULIC BEHAVIOR IN THE
PROPAGATION OF SHOCK WAVES IN THE ENVIRONMENT (IN GERMAN)* #
-LARGEST COLD #MO5A-II TEST DATA REPORT 12 EFFECTS OF ECCS
FRENCH)* #STUDY OF THE RANGE OF VELOCITY OF GAS
PROBABILISTIC METHODS IN THE SAFETY EVALUATION OF NUCLEAR
DEVELOPMENT OF A SUCTION SYSTEM FOR
FROM LIGHT WATER REACTORS (IN FRENCH)*#THE SAFETY OF FRENCH
USING COMBINATION OF MEDIA FOR THE INTENSIVE TRAINING AND
LEAKAGE MONITORING USING SONIC EMISSION ANALYSIS: EXPANDED
PROGRAM (IN GERMAN)* #
OF AN EDUCATIONAL SYSTEM USING COMBINATION OF MEDIA FOR THE
2R1032- H2O SYSTEM) (IN JAPANESE)* #FUEL COOLANT
FUEL ELEMENT BEHAVIOR DURING A LOSS-OF-COOLANT ACCIDENT AND
#THERMAL
#THERMAL
- IBM STANDARD OPERATING SYSTEM MVS (WITH FORTRAN-#EXTENDED
IDENTIFY THE INFLUENCE OF VARIOUS PARAMETERS (IN GERMAN)*
II, WATER INGRESS, AIR INGRESS, REACTIVITY EXCURSIONS (IN
III: A COMPUTER PROGRAM TO DESCRIBE THERMAL NON-EQUILIBRIUM
IMPACT OF FRAGMENTS AND PROJECTILES OF DIFFERENT MASS AND
IMPACT OF STEEL PROJECTILES ON REINFORCED CONCRETE*
IMPROVEMENT IN THE MEASUREMENT TECHNIQUES OF SONIC-
IMPULSES IN THICK WALLED STRUCTURES (IN GERMAN)*
IN-SITU TESTING OF HIGH EFFICIENCY FILTERS AT AEE WINDRIITH*
INCLUDING A 30 DIRECT HEATED ROD BUNDLE (IN FRENCH)*
INDEPENDENT VARIABLES (IN ENGLISH & JAPANESE)* #FOR
INFLUENCE OF THE INTERNAL CORROSION ON THE MECHANICAL
INFLUENCE OF THESE OCCURRENCES IN FAULT TREE ANALYSIS (IN
INFLUENCE OF VARIOUS PARAMETERS (IN GERMAN)* #HYPOTHETICAL
INFLUENTIAL PARAMETERS* #MODEL OF THE FAILURE RATES OF
INGRESS, AIR INGRESS, REACTIVITY EXCURSIONS (IN GERMAN)*
INGRESS, REACTIVITY EXCURSIONS (IN GERMAN)* #ANALYSIS FOR
INHALATION OF PLUTONIUM OXIDE USING OBSERVED LUNG
INITIAL BLOWDOWN PHASE, PARTS A, B, & C (IN GERMAN)*
INITIATION OF DETONATION OF HYDROGEN-AIR MIXTURES AND
INJECTION AND PUMP CIRCULATION ON LUCA PHENOMENA IN
INSIDE A BUBBLE RISING THROUGH A LIQUID CODE CABU I (IN
INSTALLATIONS (IN ENGLISH)* #USE OF
INSTALLATIONS AND FITTINGS (IN GERMAN)*
INSTALLATIONS FOR THE STORAGE OF IRRADIATED FUEL ELEMENTS
INSTRUCTION OF OPERATING PERSONNEL AT NUCLEAR POWER PLANTS
INSTRUMENTATION AND EVALUATION PROGRAM (IN GERMAN)* #ON
INSTRUMENTATION SYSTEM FOR THE GAS-MULTIPLE TUBE RESEARCH
INTENSIVE TRAINING AND INSTRUCTION OF OPERATING PERSONNEL
INTERACTION EXPERIMENT BY DIRECT ELECTRICAL HEATING METHOD
INTERACTION WITH THE EMERGENCY CORE COOLING (IN GERMAN)* #
INTERACTIONS BETWEEN CERNUBOND AND WATER*
INTERACTIONS BETWEEN CERNUTRU AND WATER*
- 143906 P 14
143379 P 15
146070 P 18
140447 P 22
143803 P 13
145003 P 21
144198 P 11
144198 P 10
146077 P 21
143929 P 4
144027 P 20
146073 P 20
143779 P 24
143379 P 10
143339 P 6
146070 P 12
146070 P 12
145281 P 25
143901 P 24
144280 P 12
143891 P 27
143210 P 2
147021 P 5
144828 P 23
147103 P 3
143768 P 21
146072 P 10
146796 P 13
143768 P 21
148042 P 29
143906 P 19
148048 P 24
143910 P 32

<p>#ACUSTIC NOISES WITH LOG-UP-BLOW DOWN DURING EXPERIMENT IN A NON-EQUILIBRIUM FLOW OF WATER IN SIMPLE PIPE SYSTEM #POST-EXPERIMENT CALCULATION OF THE BURST BURST AT WHICH ARE UNCONNECTED AND WAITING FOR PERIODIC TESTING #RELATIONSHIPS ANALYSIS OF THE TESTING-NONNEGLECTIBLE TEST DURATION, TEST EFFICIENCY A ONE- AND TWO-DIMENSIONAL CALCULATION OF A WATER-VAPOR EXPLOSION AND #PROTECTION OF NUCLEAR POWER PLANTS IN ON FUEL FAILURE BEHAVIOR ENRICHMENT PARAMETER TEST IN A PROBABILISTIC STUDY OF VESSEL BURST IN LIGHT WATER REACTORS ON QUESTIONS PERTAINING TO THE ESSENTIAL SAFETY ASPECTS OF A CONFINED BLOW-UP PROBABILISTIC METHODS IN THE SAFETY EVALUATION OF #POST-EXPERIMENT CALCULATION OF THE NON- AND MULTIPLE TUBE ARRAYS IN THE PRESSURE TRANSIENT OF A PURPOSES OF STUDYING THE #SPECIFICATION OF CONDITIONS OF A #SAFETY CONTAINMENT OF OF RADIOACTIVE PRIMARY LOOP COMPONENTS OF DECOMMISSIONED TRAINING AND INSTRUCTION OF OPERATING PERSONNEL AT TRANSCIENTS IN FULL-PRESSURE CONTAINMENTS OF WATER-COOLED REACTORS, FIRES, EXPLOSIONS AND AIRCRAFT #PROTECTION OF MODELS FOR THE ESTIMATION OF FLOW USING ULTRASONIC TESTS ON #THE SAFETY OF</p>	<p>142776 P 22 142777 P 23 142778 P 24 142779 P 25 142780 P 26 142781 P 27 142782 P 28 142783 P 29 142784 P 30 142785 P 31 142786 P 32 142787 P 33 142788 P 34 142789 P 35 142790 P 36 142791 P 37 142792 P 38 142793 P 39 142794 P 40 142795 P 41 142796 P 42 142797 P 43 142798 P 44 142799 P 45 142800 P 46 142801 P 47 142802 P 48 142803 P 49 142804 P 50 142805 P 51 142806 P 52 142807 P 53 142808 P 54 142809 P 55 142810 P 56 142811 P 57 142812 P 58 142813 P 59 142814 P 60 142815 P 61 142816 P 62 142817 P 63 142818 P 64 142819 P 65 142820 P 66 142821 P 67 142822 P 68 142823 P 69 142824 P 70 142825 P 71 142826 P 72 142827 P 73 142828 P 74 142829 P 75 142830 P 76 142831 P 77 142832 P 78 142833 P 79 142834 P 80 142835 P 81 142836 P 82 142837 P 83 142838 P 84 142839 P 85 142840 P 86 142841 P 87 142842 P 88 142843 P 89 142844 P 90 142845 P 91 142846 P 92 142847 P 93 142848 P 94 142849 P 95 142850 P 96 142851 P 97 142852 P 98 142853 P 99 142854 P 100 142855 P 101 142856 P 102 142857 P 103 142858 P 104 142859 P 105 142860 P 106 142861 P 107 142862 P 108 142863 P 109 142864 P 110 142865 P 111 142866 P 112 142867 P 113 142868 P 114 142869 P 115 142870 P 116 142871 P 117 142872 P 118 142873 P 119 142874 P 120 142875 P 121 142876 P 122 142877 P 123 142878 P 124 142879 P 125 142880 P 126 142881 P 127 142882 P 128 142883 P 129 142884 P 130 142885 P 131 142886 P 132 142887 P 133 142888 P 134 142889 P 135 142890 P 136 142891 P 137 142892 P 138 142893 P 139 142894 P 140 142895 P 141 142896 P 142 142897 P 143 142898 P 144 142899 P 145 142900 P 146 142901 P 147 142902 P 148 142903 P 149 142904 P 150 142905 P 151 142906 P 152 142907 P 153 142908 P 154 142909 P 155 142910 P 156 142911 P 157 142912 P 158 142913 P 159 142914 P 160 142915 P 161 142916 P 162 142917 P 163 142918 P 164 142919 P 165 142920 P 166 142921 P 167 142922 P 168 142923 P 169 142924 P 170 142925 P 171 142926 P 172 142927 P 173 142928 P 174 142929 P 175 142930 P 176 142931 P 177 142932 P 178 142933 P 179 142934 P 180 142935 P 181 142936 P 182 142937 P 183 142938 P 184 142939 P 185 142940 P 186 142941 P 187 142942 P 188 142943 P 189 142944 P 190 142945 P 191 142946 P 192 142947 P 193 142948 P 194 142949 P 195 142950 P 196 142951 P 197 142952 P 198 142953 P 199 142954 P 200 142955 P 201 142956 P 202 142957 P 203 142958 P 204 142959 P 205 142960 P 206 142961 P 207 142962 P 208 142963 P 209 142964 P 210 142965 P 211 142966 P 212 142967 P 213 142968 P 214 142969 P 215 142970 P 216 142971 P 217 142972 P 218 142973 P 219 142974 P 220 142975 P 221 142976 P 222 142977 P 223 142978 P 224 142979 P 225 142980 P 226 142981 P 227 142982 P 228 142983 P 229 142984 P 230 142985 P 231 142986 P 232 142987 P 233 142988 P 234 142989 P 235 142990 P 236 142991 P 237 142992 P 238 142993 P 239 142994 P 240 142995 P 241 142996 P 242 142997 P 243 142998 P 244 142999 P 245 143000 P 246</p>
---	---

- AFTER RUPTURE OF THE PRIMARY COOLING #INVESTIGATION OF THE AFTER RUPTURE OF THE PRIMARY COOLING #INVESTIGATION OF THE CRITICAL FLOW AND FLOW BLOCKAGE
- EVALUATION IN PRESSURE #BLASER HOLOGRAPHIC AND SPECKLE #MIXED FLOW BLOCKAGE EXPERIMENTS IN A 19- #TUBULAR FLOW BLOCKAGE EXPERIMENTS IN 37- #TUBES WITH LOSS-OF-FLOW TUBES BOILING EXPERIMENTS IN 19- #TUBES
- TO MEASURE THERMAL NON-EQUILIBRIUM FLOW OF WATER IN SIMPLE FAILURE RATES OF THE VALVES OF ST. LAURENT DES EAUX POWER TUBE ARRAYS IN THE PRESSURE TRANSIENT OF A NUCLEAR POWER THE ENSURING #SPECIFICATION OF CONDITIONS OF A NUCLEAR POWER MELT ACCIDENT WITH MELTING IN A PWR AND A PWR STANDARD #SAFETY CONTAINMENT OF NUCLEAR POWER
- PRIMARY LOOP COMPONENTS OF DECOMMISSIONED NUCLEAR POWER AND INSTRUCTION OF OPERATING PERSONNEL AT NUCLEAR POWER IN FULL-PRESSURE CONTAINMENTS OF WATER-COOLED NUCLEAR POWER EXPLOSIONS AND AIRCRAFT #PROTECTION OF NUCLEAR POWER
- #CALCULATING #EXAMINATION OF 3 HSTT #SATISFACTION ON QUESTIONS PERTAINING TO NUCLEAR ENERGY #ASSESSMENTS OF RISK FOLLOWING THE INHALATION OF #STUDY OF
- MODEL FOR THE TWO-DIMENSIONAL CALCULATION OF THE WATER GERMANIA #RESULTS OF THE LOFT EXPERIMENT LI-4 #RESULTS ANALYSIS OF THE NONNUCLEAR LOFT-TESTS LI-4 (PRE AND CE-5) (IN GERMAN AND ENGLISH) #
- EXPERIMENT RS 104 (LOFT) CONTROL OF THE ELECTRICAL HEATING OF THE FAILURE RATES OF THE VALVES OF ST. LAURENT DES EAUX MULTIPLE TUBE ARRAYS IN THE PRESSURE TRANSIENT OF A NUCLEAR STUDYING THE #SPECIFICATION OF CONDITIONS OF A NUCLEAR #SAFETY CONTAINMENT OF NUCLEAR
- IN FULL-PRESSURE CONTAINMENTS OF WATER-COOLED NUCLEAR PRIMARY LOOP COMPONENTS OF DECOMMISSIONED NUCLEAR TRAINING AND INSTRUCTION OF OPERATING PERSONNEL AT NUCLEAR FIRES, EXPLOSIONS AND AIRCRAFT #PROTECTION OF NUCLEAR #FAKAWU #THE
- #RELAP-RZURS ANALYSIS OF THE NONNUCLEAR LOFT-TESTS LI-4 (BEHAVIOUR DURING LOCA (IN GERMAN) #
- MISSION ANALYSIS EXPANDED INSTRUMENTATION AND EVALUATION ON BUCKLING OF ELLIPSOIDAL VESSEL HEADS UNDER INTERNAL A COMPUTER SIMULATING FOR GENERATING STEAM TABLES HAVING CODE FOR THE DETERMINATION OF PRESSURE TRANSIENTS IN FULL- OF THE PROCESSES IN A MULTIPLE COMPARTMENT CONTAINMENT BY FROM THE COOLANT BLOWDOWN OF A WATER- #INVESTIGATION OF THE RESEARCH ON SINGLE AND MULTIPLE TUBE ARRAYS IN THE COOLED #LOU FLOW - A COMPUTER CODE FOR THE DETERMINATION OF #FRACTURE PROBABILITY EVALUATION OF A LWR SUPPORT STRUCTURE DUE TO THE FORMATION OF MELT OR DUE TO METHODS FOR DEFECT DETECTION AND STRAIN EVALUATION IN #WIND/WATER VIBRATION EXPLOSIONS - ESTIMATES OF #OSLAIM-III A COMPUTER PROGRAM FOR #FLUID-STRUCTURE INTERACTIONS DURING BLOWDOWN OF A SPOKING MANUFACTURING FAULTS (IN FRENCH) #CONTAMINATION OF A OF THE RADIOACTIVITY IN THE PRIMARY SYSTEM OF IN FRENCH) #CONTAMINATION OF A PRESSURIZED WATER REACTOR'S WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE FINAL REPORT CONTROLLED-BLASTING DEMOLITION OF RADIOACTIVE #EXPERIMENTAL INVESTIGATIONS OF THE RADIOACTIVITY IN THE #SELECTION OF EVENTS FOR A INSTALLATIONS (IN ENGLISH) # #USE OF IN FRENCH) #A #FRACTURE #SHORT ANALYSIS OF A PROGRESSIVE DISTORTION SYSTEM OF THE #A FIRST APPROACH OF THE RARE EVENT #USE OF MARKOV PROCESSES FOR RELIABILITY SOLUTIONS TO REACTOR BENCHMARK DWR LATTICE CELL #USE OF MARKOV
- PRESSURE IN WATER-COOLED REACTORS #INVESTIGATION OF THE THERMAL CYCLING (IN ENGLISH) #ANALYSIS OF THE FISSION #AN ANALYSIS OF THE ADDITIONAL FISSION #ACTIVATED CORROSION
- PARAMETER TEST IN N89W (IN #EFFECT OF HEAT GENERATION TO MULTICOMPARTMENT, TWO-PHASE SYSTEMS - THE COMPUTER 30, 1978 (LIST OF REPORTS FROM THE REACTOR SAFETY RESEARCH OCTOBER 1 (LIST OF REPORTS FROM THE REACTOR SAFETY RESEARCH (IN FRENCH) # #SHORT ANALYSIS OF A #NUCLEAR SAFETY #ANALYSIS OF THE D SERIES COMPONENTS OF RESEARCH MATERIALS AND COMPONENT PARTS AT IMPACT OF FRAGMENTS AND COMPARISON WITH EXPERIMENTAL TESTS (IN #IMPACT OF STEEL #MINISTRY FOR RESEARCH AND TECHNOLOGY CONCERNING RESEARCH TECHNOLOGY OF THE ANNUAL REPORT ON REACTOR SAFETY RESEARCH #INITIATION OF DETONATION OF HYDROGEN-AIR MIXTURES AND WALLED STRUCTURES (IN GERMAN) #
- PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT PHENOMENA FOR A TWO-PHASE FLOW (IN FRENCH) #
- PHOTOGRAPHY METHODS FOR DEFECT DETECTION AND STRAIN PIN BUNDLE (IN ENGLISH) # #ACOUSTIC PIN SODIUM COOLED BUNDLES WITH GRID SPACERS (IN ENGLISH) #
- PIPE SYSTEMS # #BUBBLE BUBBLE III: A COMPUTER PROGRAM PLANT ACCORDING TO INFLUENTIAL PARAMETERS #MODEL OF THE PLANT IN THE LARGE EXPERIMENT AREA OF THE GRS (IN GERMAN) #
- PLANT WITH A PWR FOLLOWING A LOCA FOR PURPOSES OF STUDYING PLANT, AND THE COUPLING OF MELDIN-1 AND BILANZ-14 PART I I PLANTS (IN GERMAN) #
- PLANTS (IN GERMAN) # #BLASTING DEMOLITION OF RADIOACTIVE PLANTS (IN GERMAN) # OF MEDIA FOR THE INTENSIVE TRAINING PLANTS (IN GERMAN) # DETERMINATION OF PRESSURE TRANSIENTS PLANTS (INPP) AGAINST EXTERNAL EVENTS (EARTHQUAKES, FIRES) PLASTIC DEFORMATION OF STRUCTURES (IN FRENCH) #
- PLATES FRACTURED IN AIR (IN GERMAN) # PLUTONIUM (IN GERMAN) #
- PLUTONIUM GASE USING OBSERVED LUNG CLEARANCE PATTERNS # POLLUTANT DISPERSION IN WATER AND AIR (IN ENGLISH) #
- POOL-SWELL IN THE CONDENSATION CHAMBER OF A NUCLEAR SYSTEM POST TEST CALCULATIONS USING THE COMPUTER CODE PURIFAN (IN POST TEST CALCULATIONS) (IN GERMAN) # #RELAP- POWER (IN GERMAN) # #RELAP- POWER (IN GERMAN) #
- POWER PLANT ACCORDING TO INFLUENTIAL PARAMETERS # BRIDEL POWER PLANT IN THE LARGE EXPERIMENT AREA OF THE GRS (IN POWER PLANT WITH A PWR FOLLOWING A LOCA FOR PURPOSES OF POWER PLANTS (IN GERMAN) #
- POWER PLANTS (IN GERMAN) # OF PRESSURE TRANSIENTS POWER PLANTS (IN GERMAN) # OF MEDIA FOR THE INTENSIVE POWER PLANTS (IN GERMAN) # OF MEDIA FOR THE INTENSIVE POWER PLANTS (INPP) AGAINST EXTERNAL EVENTS (EARTHQUAKES, FIRES) POWER RAMP FUEL ROD IRRADIATION TESTS 1576777 (GERMAN) #
- PRACTICE OF QUALITY ASSURANCE BY FRANGTUM (IN ENGLISH) # PRE AND POST TEST CALCULATIONS) (IN GERMAN) #
- PREDICTION OF ROSA-III EXPERIMENT RUN 702 (IN JAPANESE) # PRELIMINARY EMPIRICAL DESCRIPTION OF THE FUEL ROD PRELIMINARY EXPERIMENT ON LEAKAGE MONITORING USING SONIC PRESSURE (IN FRENCH) # #EXPERIMENTAL TESTS PRESSURE AND DENSITY AS THE INDEPENDENT VARIABLES (IN PRESSURE CONTAINMENTS OF WATER-COOLED NUCLEAR POWER PLANTS PRESSURE IN WATER-COOLED REACTORS WITH REFRIGERATED PRESSURE TRANSIENT IN A MULTICOMPARTMENT CONTAINMENT PRESSURE TRANSIENT OF A NUCLEAR POWER PLANT IN THE LARGE PRESSURE TRANSIENTS IN FULL-PRESSURE CONTAINMENTS OF WATER- PRESSURE VESSEL (IN ENGLISH) #
- PRESSURE VESSEL FAILURE, PART II (IN GERMAN) # OF THE CORE PRESSURE VESSEL#BLASER HOLOGRAPHIC AND SPECKLE PHOTOGRAPHY PRESSURES AND YIELDS #
- PRESSURIZED WATER REACTOR BLOWDOWN ANALYSIS (IN ENGLISH) # PRESSURIZED WATER REACTOR-PLUS (IN GERMAN) # DIMENSIONAL PRESSURIZED WATER REACTOR'S PRIMARY CIRCUIT BY FUEL RODS PRESSURIZED WATER REACTORS (IN GERMAN) # INVESTIGATIONS PRIMARY CIRCUIT BY FUEL RODS SHOWING MANUFACTURING FAULTS (PRIMARY COOLING CIRCUIT IN WATER-COOLED #FACTORS) # PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS # QUICK PRIMARY COOLING CIRCUITS IN WATER-COOLED REACTORS #
- PRIMARY LOOP COMPONENTS OF DECOMMISSIONED NUCLEAR POWER PRIMARY SYSTEM OF PRESSURIZED WATER REACTORS (IN GERMAN) #
- PROBABILISTIC EVALUATION OF PWR SAFETY (IN ENGLISH) # PROBABILISTIC METHODS IN THE SAFETY EVALUATION OF NUCLEAR PROBABILISTIC STUDY OF VESSEL BURST IN LIGHT WATER NSSS I PROBABILISTIC EVALUATION OF A LWR PRESSURE VESSEL (IN PROBLEM (TENSION AND CYCLE TORSION) (IN FRENCH) #
- PROBLEM (THE STUDY OF THE RELIABILITY OF THE PROTECTION PROBLEMS (IN ENGLISH) # #A REVIEW OF INTERNATIONAL PROCESSES FOR RELIABILITY PROBLEMS (IN ENGLISH) #
- PROCESSES IN A MULTIPLE COMPARTMENT CONTAINMENT BY PRODUCT RELEASE FROM A DEFECTED FUEL ROD - EFFECT OF PRODUCT RELEASE PHENOMENA (IN ENGLISH & JAPANESE) #
- PRODUCTS IN LWR LOOPS (IN GERMAN) # PROFILE IN PELLETS ON FUEL FAILURE BEHAVIOR ENRICHMENT PROGRAMS CRIS AND CRITER # DISCHARGE MODEL APPLIED PROGRAMS OF DMFT, USNRC, EPRI AND JSA, JULY 1-SEPTEMBER PROGRAMS OF DMFT, USNRC, EPRI, AND JSA, REPORT PERIOD PROGRESSIVE DISTORTION PROBLEM (TENSION AND CYCLE TORSION) PROJECT FIRST SEMI-ANNUAL REPORT 1578 (IN GERMAN) #
- PROJECT RS 90 (MODEL CONTAINMENT) PART 2 (IN GERMAN) # PROJECTS OF DIFFERENT MASS AND VELOCITY (IN GERMAN) #
- PROJECTILES ON REINFORCED CONCRETE, CALCULATION AND PROJECTS IN THE AREA OF REACTOR SAFETY REPORTING PERIOD PROJECTS SPONSORED BY THE MINISTRY FOR RESEARCH AND PROPAGATION OF SHOCK WAVES IN THE ENVIRONMENT (IN GERMAN) #
- PROPAGATION OF SIMULATED SONIC EMISSION-IMPULSES IN THICK 145847 P 11
146273 P 12
143913 P 1
120343 P 23
143776 P 21
143907 P 26
140447 P 24
143335 P 6
146488 P 14
146755 P 14
145458 P 25
147176 P 17
147617 P 23
143768 P 21
145636 P 16
143138 P 5
142870 P 3
146759 P 13
146363 P 18
146281 P 20
146901 P 8
145757 P 10
146068 P 16
143905 P 10
146074 P 10
143854 P 15
143334 P 6
146488 P 14
146755 P 14
147176 P 17
145636 P 16
147617 P 23
143768 P 21
143138 P 5
146675 P 23
146065 P 9
143905 P 10
147004 P 12
145194 P 22
145872 P 10
146870 P 8
144527 P 28
145636 P 16
143325 P 15
146875 P 10
146488 P 14
145636 P 16
143876 P 5
145756 P 15
120342 P 21
146842 P 23
147502 P 29
146804 P 20
143383 P 4
144596 P 18
143383 P 4
146273 P 12
145846 P 10
145847 P 11
147617 P 23
144596 P 18
147175 P 9
147021 P 5
146865 P 8
143876 P 5
146871 P 8
143775 P 7
143777 P 3
147755 P 24
143777 P 3
143329 P 16
144596 P 5
144523 P 27
143083 P 22
143974 P 48
146482 P 23
144854 P 18
146813 P 17
146871 P 8
147794 P 19
146758 P 14
143803 P 13
145636 P 16
147463 P 18
147463 P 18
144260 P 14
144596 P 10

STRATIFIED GASEOUS MIXTURES (IN ENGLISH)* #FLARE FUEL INFLUENCE OF THE INTERNAL CORROSION ON THE MECHANICAL STRESS (IN ENGLISH)* #A EVENTS: EARTHQUAKES, FIRES, EXPLOSIONS AND AIRCRAFT PROGRAM FOR THE TESTING OF A FRACTURE SAFETY DEVICE RARE EVENT PROBLEM BY THE STUDY OF THE RELIABILITY OF THE #BORA-II TEST DATA REPORT 12 EFFECTS OF ECCS INJECTION AND OF A NUCLEAR POWER PLANT WITH A PWR FOLLOWING A LOCA FOR STUDIES FOR THE DEVELOPMENT OF SAFETY ADVANCED CODE FOR #RELIABILITY ASSESSMENT OF THE SECONDARY CONTAINMENT OF A FIRST PHASE OF THE CORE MELT ACCIDENT WITH WELSH-1 IN A PROGRAM FOR ANALYZING THE THERMO-HYDRAULIC BEHAVIOR IN A SPECIFICATION OF CONDITIONS OF A NUCLEAR POWER PLANT WITH A A LORE DESCRIBING THE THERMAL AND MECHANICAL BEHAVIOR OF A DOWNCOMER EFFECTIVE WATER HEAT DURING A REPLEDGE PHASE OF RELIABILITY OF THE PROTECTION SYSTEM OF THE FESSENHEIM 1 #SELECTION OF EVENTS FOR A PROBABILISTIC EVALUATION OF MODELIZATION DURING THE REPLEDGE PHASE OF A #REPORT ON THE CONVERSION OF THE LSW-CODE TRAIL- #AN ANALYSIS OF LOST LI-Z EXPERIMENT BY ALAHM- BLOWDOWN ANALYSIS (IN ENGLISH)* #ALAHM	PROPAGATION THROUGH UNCONFINED AND CONFINED HEMISPHERICAL PROPERTIES OF 2H2+4 TORJENE (IN ENGLISH)* IRRADIATED DRAIN PROPOSED WIND SHIFT MODEL FOR THE GERMAN REACTOR SAFETY PROTECTION OF NUCLEAR POWER PLANTS (NPPs) AGAINST EXTERNAL PROTECTION SYSTEM FOR REACTOR COMPONENTS (IN GERMAN)* PROTECTION SYSTEM OF THE FESSENHEIM 1 PWR REACTOR (IN PUMP CIRCULATION IN LOCA PHENOMENA IN LARGEST COOL-LEG PURPOSES OF STUDYING THE ENSUING DECONTAMINATION AND PWR (IN FRENCH)* #FRENCH THERMO-HYDRAULIC PWR (IN GERMAN)* PWR AND A BWR STANDARD PLANT, AND THE COUPLING OF WELSH-1 PWR CORE DURING A LOCA (IN ENGLISH)* #ASCLT-1: A COMPUTER PWR FOLLOWING A LOCA FOR PURPOSES OF STUDYING THE ENSUING PWR FUEL ROD DURING A LOCA (IN FRENCH)* #COUPLING PWR LCCA (IN JAPANESE)* #EXPERIMENT OF THE PWR REACTOR (IN ENGLISH)* RARE EVENT PROBLEM BY THE STUDY OF THE PWR SAFETY (IN ENGLISH)* PWR'S CORE (IN FRENCH)* AND HEAT TRANSFER THERMOHYDRAULIC PI (VERSION 16-3) TO TEM STANDARD OPERATING SYSTEM MVD (IN PI COMPUTER CODE (IN ENGLISH)* PI: A COMPUTER PROGRAM FOR PRESSURIZED WATER REACTOR	143078 P 0 144073 P 20 144086 P 21 144138 P 0 144150 P 22 144279 P 2 144386 P 27 144755 P 24 144758 P 6 144840 P 24 144840 P 25 144849 P 27 144795 P 24 144871 P 1 144893 P 28 144779 P 7 144775 P 9 144940 P 1 144966 P 18 144970 P 27 147592 P 29
GERMAN)* #THE PRACTICE OF #ATTITUDES ON OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS, REACTOR CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT, REACTOR CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT,	QUALITY ASSURANCE BY FRANGIBLE (IN ENGLISH)* QUESTIONS PERTAINING TO NUCLEAR ENERGY PLUTONIUM (IN QUICK LOCK REPORT EXPERIMENT D18 (IN GERMAN)* #FATER RUPTURE QUICK LOCK REPORT 1 (IN GERMAN)* IN A LIGHT WATER QUICK LOCK REPORT 2 (IN GERMAN)* IN A LIGHT WATER	148669 P 9 148863 P 18 148846 P 10 148862 P 11 148860 P 12
#POST TEST CALCULATIONS (IN GERMAN)* #RELAP- #RS236 - FINAL REPORT CONTROLLED-BLASTING DEMOLITION OF REACTORS (IN GERMAN)* #EXPERIMENTAL INVESTIGATIONS OF THE #KFA/KNU POWER LIQUID CODE CABO I (IN FRENCH)* #MODEL FOR THE CALCULATION OF THE RATE OF VOIDING DURING A PROTECTION SYSTEM OF THE #A FIRST APPROACH OF THE FRENCH)* #EXPERIMENTAL TESTS ON OCCURRENCES IN FAULT TREE #UNCERTAINTY OF THE FAILURE COMPARISON OF SEVERAL #MODEL FOR THE CALCULATION OF THE ACCORDING TO INFLUENTIAL PARAMETERS* #MODEL OF THE FAILURE #AN APPRAISAL OF SUBCOOLED BOILING AND SLIP GAS REACTORS, PHASE II, WATER INGRESS, AIR INGRESS, OF THE PROTECTION SYSTEM OF THE FESSENHEIM 1 PWR CONTAINMENT FROM THE COOLANT BLOWDOWN OF A WATER-COOLED #ALAHM-PI: A COMPUTER PROGRAM FOR PRESSURIZED WATER TESTING OF A PWR - RE SAFETY DEVICE PROTECTION SYSTEM FOR INVESTIGATION OF THE HYDROGEN DISTRIBUTION IN A LIGHT WATER INVESTIGATION OF THE HYDROGEN DISTRIBUTION IN A LIGHT WATER #PROGRESS REPORT ON FAST BREEDER FRAGMENTS AND PROJECTILES OF #BEHAVIOR OF SPECIFIC AND TECHNOLOGY CONCERNING RESEARCH PROJECTS IN THE AREA OF JSTA, JULY 1-SEPTEMBER 30, 1978 (#LIST OF REPORTS FROM THE JSTA, REPORT PERIOD OCTOBER 1 - #LIST OF REPORTS FROM THE FOR RESEARCH AND TECHNOLOGY OF THE #ANNUAL REPORT ON #A PROPOSED WIND SHIFT MODEL FOR THE GERMAN ON THE RESEARCH PROGRAM SPONSORED BY DMPT IN THE AREA OF #RESEARCH PROGRAM ON #ANALYSIS OF BOILING-WATER OF THE WATER POOL-SWELL IN THE CONDENSATION CHAMBER OF A INTERACTIONS DURING BLOWDOWN OF A PRESSURIZED WATER MANUFACTURING FAULTS (CONTAMINATION OF A PRESSURIZED WATER THE STORAGE OF IRRADIATED FUEL ELEMENTS FROM LIGHT WATER THE ESTIMATION OF FLOW USING ULTRASONIC TESTS ON NUCLEAR MECHANICS RESEARCH ON MATERIALS USED IN FAST SODIUM-COOLED RADIOACTIVITY IN THE PRIMARY SYSTEM OF PRESSURIZED WATER THERMAL TRANSIENT IN LIGHT-WATER AND HIGH TEMPERATURE #THE SAFETY OF NUCLEAR COMPARTMENT CONTAINMENT BY PRESSURE IN WATER-COOLED RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED #COMPREHENSIVE DAMAGE ANALYSIS FOR HIGH TEMPERATURE GAS RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED RUPTURE OF THE PRIMARY COOLING CIRCUITS IN WATER-COOLED REPORT (IN GERMAN)* #DISTURBANCE AND ERROR TWO-DIMENSIONAL FLOW OF A COMPRESSIBLE MEDIUM IN COUPLED #DUOSE TUBULAR TEST SECTION ON OMEGA LOOP (IN FRENCH)* #DATA JAPANESE & ENGLISH)* #FLOW #REFLUX-GRS ANALYSIS OF THE #EXPERIMENT OF THE DOWNCOMER EFFECTIVE WATER HEAT DURING A AND HEAT TRANSFER THROUGH A TUBULAR MODIFICATION DURING THE ENGLISH)* CONTAINMENT BY PRESSURE IN WATER-COOLED REACTORS WITH	R/GRS ANALYSIS OF THE NONNUCLEAR LOFT-TESTS LB-4 (PRE AND RADIOACTIVE PRIMARY LOOP COMPONENTS OF DECOMMISSIONED RADIOACTIVITY IN THE PRIMARY SYSTEM OF PRESSURIZED WATER RAMPS FUEL ROD IRRADIATION TESTS (1976/77 (GERMAN)* RANGE OF VELOCITY OF GAS INJURY A BUBBLE RISING THROUGH A RAPID FAILURE (BLOWDOWN) COMPARISON OF SEVERAL MODELS (IN RARE EVENT PROBLEM BY THE STUDY OF THE RELIABILITY OF THE WATCHET OF 304 AUSTENITIC STEEL, AT HIGH TEMPERATURE (IN RATE OF COMPONENTS AND THE APPARENT INFLUENCE OF THESE RATE OF VOIDING DURING A RAPID FAILURE (BLOWDOWN) RATES OF THE VALVES OF ST. LAURENT DES EAUX POWER PLANT RATIC FROM MEASUREMENTS MADE IN LINGEN GWR* REACTIVITY EXCURSIONS (IN GERMAN)* FOR HIGH TEMPERATURE REACTOR (IN ENGLISH)* BY THE STUDY OF THE RELIABILITY OF THE REACTOR - INTERIM RESEARCH REPORT C 13 (IN GERMAN)* REACTOR BLOWDOWN ANALYSIS (IN ENGLISH)* REACTOR COMPONENTS (IN GERMAN)* PROGRAM FOR THE REACTOR CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT, REACTOR CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT, REACTOR DEVELOPMENT IN JAPAN, APRIL-JUNE 1978* REACTOR MATERIALS AND COMPONENT PARTS AT IMPACT OF REACTOR SAFETY REPORTING PERIOD OCTOBER-DECEMBER 31, 1978 REACTOR SAFETY RESEARCH PROGRAMS OF DMPT, LSNC, EPHI AND REACTOR SAFETY RESEARCH PROGRAMS OF DMPT, LSNC, EPHI, AND REACTOR SAFETY RESEARCH PROJECTS SPONSORED BY THE MINISTRY REACTOR SAFETY STUDY (IN ENGLISH)* REACTOR SAFETY, JULY 1-SEPTEMBER 30, 1978 (IN GERMAN)* REACTOR SAFETY, 30 EXPERIMENT (IN GERMAN)* REACTOR STEAM CHUGGING (IN ENGLISH)* REACTOR SYSTEM (IN GERMAN)*THE TWO-DIMENSIONAL CALCULATION REACTOR-FLUX (IN GERMAN)*THREE-DIMENSIONAL FLOID-STRUCTURE REACTORS PRIMARY CIRCUIT BY FUEL ROD SHAKING REACTORS (IN FRENCH)* SAFETY OF FRENCH INSTALLATIONS FOR REACTORS (IN GERMAN)* OF INTERPRETATIONAL MODELS FOR REACTORS (IN GERMAN)* EMISSION MEASUREMENTS IN FRACTURE REACTORS (IN GERMAN)* #EXPERIMENTAL INVESTIGATIONS OF THE REACTORS (IN GERMAN)* BLOWDOWN ACCIDENT AND THE SUBSEQUENT REACTORS IN FRANCE (IN ENGLISH)* REACTORS WITH REFRIGERATED CONDENSER (IN GERMAN)* MULTIPLE REACTORS, CONDENSATION IN CONTAINMENT BY EXPERIMENTS CGA REACTORS, PHASE II, WATER INGRESS, AIR INGRESS, REACTIVITY REACTORS, QUICK LOCK REPORT EXPERIMENT D18 (IN GERMAN)* REACTORS, SUPPLEMENTAL RESEARCH DOCUMENTATION D18 (IN RECONSTRUCTION WITH HELP OF TIME-OF-FLIGHT DATA, FINAL RECTANGULAR AREA (IN GERMAN)* FOR THE CALCULATION OF THE REDUCTION (IN GERMAN)* REDUCTION OF THE FIRST TEST SERIES OF BLOWDOWN ON A REDUCTION TRANSIENT BURNOUT IN AN ANNULAR TEST SECTION (IN REFLOED EXPERIMENTS (RS 62) (IN ENGLISH)* REFLOED PHASE OF PWR LCCA (IN JAPANESE)* REFLOED PHASE OF A PWR'S CORE (IN FRENCH)* #FLUX #REFLUX-GRS ANALYSIS OF THE REPLEDGE EXPERIMENTS (RS 62) (IN REFRIGERATED CONDENSER (IN GERMAN)* A MULTIPLE COMPARTMENT	143995 P 10 147817 P 21 148596 P 18 148675 P 23 148610 P 2 143751 P 2 143779 P 7 148672 P 4 147779 P 24 143751 P 6 143239 P 2 148847 P 31 148670 P 18 143779 P 7 148876 P 10 148792 P 17 147592 P 29 145155 P 22 148882 P 13 148286 P 14 143343 P 20 143803 P 13 147483 P 18 144854 P 16 148813 P 17 147407 P 18 148806 P 21 145165 P 16 148671 P 21 148825 P 17 145757 P 10 148804 P 20 143383 P 4 147193 P 3 146486 P 13 148805 P 26 148596 P 18 148874 P 12 143103 P 6 143329 P 10 148733 P 12 148670 P 13 145846 P 10 145847 P 11 148368 P 13 148750 P 20 144158 P 22 143755 P 3 148844 P 29 143903 P 14 143893 P 28 143928 P 1 143903 P 14 143329 P 10

EXPERIMENTAL TESTS (EN)	#EFFECT OF STEEL PROJECTILES ON PWR AND MUST TEST CALCULATIONS FOR GERMAN*	145036 P 21
FORMAL-HYDRAULIC ANALYSIS	#THE STATUS OF DEVELOPMENT OF THE BUBBLE RISE MODEL IN BREAK RUNS 413 AND 3121 (ANALYSIS OF LOCA EXPERIMENTS WITH CYCLING (EN ENGLISH)*	143905 P 10
PWR (EN GERMAN)*	#ANALYSIS OF THE FISSION PRODUCT #AN ANALYSIS OF THE ADDITIONAL FISSION PRODUCT	145040 P 31
APPROACH OF THE MARK EVENT PROBLEM BY THE STUDY OF THE RISE OF MARKER PROCESSES FOR BWRs ANNUAL PROGRESS	CONCERNING RESEARCH PROJECTS IN THE AREA OF REACTOR SAFETY USNR, EPRI, AND JSTA, JULY 1-SEPTEMBER 30, 1978 #LIST OF USNR, EPRI, AND JSTA, REPORT PERIOD OCTO 1 - 1 #LIST OF BY THE MINISTRY FOR RESEARCH AND TECHNOLOGY OF THE FEDERAL AREA OF REACTOR #REPORT OF THE FEDERAL MINISTER FOR SAFETY RESEARCH PROJECTS SPONSORED BY THE MINISTRY FOR COOLING CIRCUITS IN WATER-COOLED REACTORS, SUPPLEMENTAL IN #INVESTIGATION MEASUREMENTS IN FRACTURE MECHANICS (PRELIMINARY TESTS OF A NUCLEAR POWER PLANT #EXPERIMENTAL BREAKDOWN PHASES, PARTS A, B, C #EXPERIMENTAL AND THEORETICAL INSTRUMENTATION SYSTEM FOR THE GAS-MULTIPLE TUBE GERMAN*	145041 P 31
SUMMARY OF THE THEORETICAL STUDIES ON THE D-SERIES OF THE SAFETY, JULY 1-SEPTEMBER 30, 1978 (EN GERMAN)*#REPORT ON THE SEPTEMBER 30, 1978 #LIST OF REPORTS FROM THE REACTOR SAFETY PERIOD OCTOBER 1 - #LIST OF REPORTS FROM THE REACTOR SAFETY GERMAN*	#ANNUAL REPORT ON REACTOR SAFETY THE COOLANT BLOWDOWN OF A WATER-COOLED REACTOR - INTERIM #EMERGENCY COOLING DEPRESSURIZATION PARTIAL HYPERBOLETS (EN FRENCH)* #CHARACTERISTICS AND DENSITY METHOD FOR TRANSCIENT TWO PHASE STATIC ATOMIC #THE COMPUTER CODE "DURFAN" (EN GERMAN)* #ON THE LWR FUEL ROD BEHAVIOR UNDER ACCIDENT #STATUS AND LATTICE CELL PROBLEMS*	145042 P 17
CONTRIBUTE TO THE THERMODYNAMICS OF CONDENSED UO ₂ #A #DEVELOPMENT OF THE BUBBLE #STUDY OF THE RANGE OF VELOCITY OF GAS INSIDE A BUBBLE OBSERVED LUNA CLEARANCE PATTERNS #ASSESSMENTS OF #OF THE FISSION PRODUCT RELEASE FROM A DEFECTED FUEL THEORETICAL AND EXPERIMENTAL INVESTIGATIONS ON THE LWR FUEL #PRELIMINARY EMPIRICAL DESCRIPTION OF THE FUEL #OF A PART OF THE LWR OMEGA INCLUDING A 30 DIRECT HEATED IN GERMAN*	#REVALUATION OF THE 25- THE THERMAL AND MECHANICAL BEHAVIOR OF A PWR FUEL #KFA/KWU POWER RAMPS FUEL #A PRESSURIZED WATER REACTOR'S PRIMARY CIRCUIT BY FUEL #EXPERIMENTAL TESTS ON BATCHES OF 304 AUSTENITIC STEEL AT ANALYSIS OF LOCA EXPERIMENTS WITH RELAP4 CODE (ANALYSIS OF PUMP CIRCULATION ON LOCA PHENOMENA IN LARGEST COOL LEG #PREDICTION OF #SHUTDOWN - EXPERIMENT THEORETICAL STUDIES ON THE D-SERIES OF THE RESEARCH PROGRAM #ANALYSIS OF THE D-SERIES EXPERIMENTS OF RESEARCH PROJECT #REFLECTIONS ANALYSIS OF THE REFLOOD EXPERIMENTS (#REVALUATION OF THE 25-RD DUNLE TEST (#RADIOACTIVE PRIMARY LOOP COMPONENTS OF DECOMMISSIONED GERMAN)* #RESULTS OF THE LFT EXPERIMENT LI-42 POST #PREDICTION OF WOSA-111 EXPERIMENT CIRCULATION ON LOCA PHENOMENA IN LARGEST COOL LEG BREAKS (#CODE ANALYSIS OF WOSA-11 EXPERIMENTS FOR COOL LEG BREAK CONDITIONS (EN ENGLISH)* #CREEP OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER #REGENERATION OF 3 HST PLATES	145043 P 12
REINFORCED CONCRETE, CALCULATION AND COMPARISON WITH RELAP-4FORS ANALYSIS OF THE NONNUCLEAR LOCA-TESTS LI-4 1 RELAP-4FORS 111 AT JULY 1976 - A PROGRAM FOR TRANSIENT RELAP-4FORS	RELAP4 CODE (ANALYSIS OF WOSA-11 EXPERIMENTS FOR COOL LEG RELEASE FROM A DEFECTED FUEL ROD - EFFECT OF THERMAL RELEASE PHENOMENA (EN ENGLISH & JAPANESE)*	145044 P 27
RELIABILITY ASSESSMENT OF THE SECONDARY CONTAINMENT OF A RELIABILITY OF THE PROTECTION SYSTEM OF THE FESSENHEIM 1 RELIABILITY PROBLEMS (EN ENGLISH)*	REPORT-1978 (EN GERMAN)*	145045 P 5
REPORTING PERIOD OCTOBER-DECEMBER 31, 1978 (EN GERMAN)*	REPORTS FROM THE REACTOR SAFETY RESEARCH PROGRAMS OF DMFT, REPUBLIC OF GERMANY, 1978 (EN ENGLISH)* PROJECTS SPONSORED RESEARCH AND TECHNOLOGY CONCERNING RESEARCH PROJECTS IN RESEARCH DOCUMENTATION DIS (EN GERMAN)* #OF THE PRIMARY RESEARCH ON MATERIALS USED IN FAST SODIUM-COOLED REACTORS (RESEARCH ON SINGLE AND MULTIPLE FLOW ARRAYS IN THE RESEARCH ON THE THERMAL HYDRAULIC BEHAVIOR IN THE INITIAL RESEARCH PROGRAM (EN GERMAN)*	145046 P 24
RESEARCH PROGRAM ON REACTOR SAFETY, 30 EXPERIMENT (EN RESEARCH PROGRAM RD 50 (MODEL CONTAINMENT) PART 1 (EN RESEARCH PROGRAM SPONSORED BY DMFT IN THE AREA OF REACTOR RESEARCH PROGRAMS OF DMFT, USNR, EPRI, AND JSTA, JULY 1- RESEARCH PROGRAMS OF DMFT, USNR, EPRI, AND JSTA, REPORT RESEARCH PROJECT RD 50 (MODEL CONTAINMENT) PART 2 (EN RESEARCH PROJECTS IN THE AREA OF REACTOR SAFETY REPORTING RESEARCH PROJECTS SPONSORED BY THE MINISTRY FOR RESEARCH RESEARCH REPORT C 13 (EN GERMAN)* #CONTAINMENT FROM RESEARCH, BLOCKED COOLING CHANNELS WITH BWR GELMREY (EN RESOLUTIONS OF THE SYSTEM OF EQUATIONS DERIVED FROM RESEARCH (EN GERMAN)* #DEVELOPMENT OF A MASS- RESULTS OF THE LFT EXPERIMENT LI-4: POST RUN CALCULATIONS RESULTS OF THE THEORETICAL AND EXPERIMENTAL INVESTIGATIONS REVIEW OF INTERNATIONAL SOLUTIONS TO NEACP BENCHMARK BWR #VIEW OF THE ARGUMENTS* #ODI ELECTRONIC TRANSITIONS WISE #MODEL IN RELAP-4FORS*	145047 P 12	
RISE THROUGH A LIQUID CODE CABU 1 (EN FRENCH)*	RISK FOLLOWING THE INHALATION OF PLUTONIUM OXIDE USING ROD - EFFECT OF THERMAL CYCLING (EN ENGLISH)* #ANALYSIS ROD BEHAVIOR UNDER ACCIDENT CONDITIONS (EN GERMAN)*# OF THE ROD BEHAVIOR DURING LOCA (EN GERMAN)*	145048 P 18
ROD BURST TEST (RS-37C) WITH THE CALCULATIONAL PROGRAM L ROD DURING A LOCA (EN FRENCH)* #CUPIDONE A CODE DESCRIBING ROD IRRADIATION TESTS 1976/77 (GERMAN)*	RODS SHOWING MANUFACTURING FAULTS (EN FRENCH)*	145049 P 4
ROD TEMPERATURE (EN FRENCH)*	ROSA-11 EXPERIMENTS FOR COOL LEG BREAK RUNS 413 AND 3121 (ROSA-11 TEST DATA REPORT 12 EFFECT OF UO ₂ INJECTION AND ROSA-111 EXPERIMENT RUN 702 (EN JAPANESE)*	145050 P 27
ROSA-11 EXPERIMENT RUN 702 (EN JAPANESE)*	RS 109 (LOU) CONTROL OF THE ELECTRICAL HEATING POWER (EN RS 50 (MODEL CONTAINMENT) PART 1 (EN GERMAN)* #OF THE RS 50 (MODEL CONTAINMENT) PART 2 (EN GERMAN)*	145051 P 15
RS 622 (EN ENGLISH)*	RS-37C) WITH THE CALCULATIONAL PROGRAM (EN GERMAN)*	145052 P 14
RS-37C) WITH THE CALCULATIONAL PROGRAM (EN GERMAN)*	RS236 - FINAL REPORT CONTROLLED-BLASTING DEMOLITION OF RUN CALCULATIONS USING THE COMPUTER CODE "DURFAN" (EN RUN 702 (EN JAPANESE)*	145053 P 23
RUNS 332, 413, 4252 (EN ENGLISH & JAPANESE)* #AND PUMP RUNS 413 AND 3121 (EN ENGLISH & JAPANESE)* #WITH RELAP4J RUPTURE AT NON-STEADY STRESS AND TEMPERATURE LOADING RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED RUPTURE OF THE PRIMARY COOLING CIRCUITS IN WATER-COOLED RUPTURED IN AIR (EN GERMAN)*	145054 P 27	
RUPTURED IN AIR (EN GERMAN)*		145055 P 2
		145056 P 17
		145057 P 12
		145058 P 10
		145059 P 31
		145060 P 27
		145061 P 5
		145062 P 24
		145063 P 7
		145064 P 2
		145065 P 17
		145066 P 12
		145067 P 10
		145068 P 18
		145069 P 11
		145070 P 25
		145071 P 11
		145072 P 26
		145073 P 18
		145074 P 24
		145075 P 13
		145076 P 21
		145077 P 11
		145078 P 16
		145079 P 10
		145080 P 17
		145081 P 12
		145082 P 18
		145083 P 10
		145084 P 16
		145085 P 17
		145086 P 18
		145087 P 11
		145088 P 23
		145089 P 21
		145090 P 5
		145091 P 6
		145092 P 17
		145093 P 17
		145094 P 22
		145095 P 5
		145096 P 2
		145097 P 6
		145098 P 18
		145099 P 16
		145100 P 17
		145101 P 12
		145102 P 10
		145103 P 16
		145104 P 2
		145105 P 17
		145106 P 18
		145107 P 11
		145108 P 23
		145109 P 21
		145110 P 5
		145111 P 6
		145112 P 17
		145113 P 17
		145114 P 18
		145115 P 10
		145116 P 16
		145117 P 17
		145118 P 18
		145119 P 11
		145120 P 23
		145121 P 21

RESEARCH PROGRAM SPONSORED BY DMPT IN THE AREA OF REACTOR SAFETY, 30 EXPERIMENT (IN GERMAN)*	ON THE	145160 P 10
IN THE MEASUREMENT TECHNIQUE OF SONIC EMISSION ANALYSIS (SEAI) (IN GERMAN)*		145171 P 11
RELIABILITY ASSESSMENT OF THE EXPERIMENTAL AND THEORETICAL STUDY OF THE BLOWDOWN OF THE IN TRANSIENT CONDITIONS, ON A STAINLESS STEEL TEST SECTION (IN FRENCH)*		145180 P 11
FLOW REDUCTION TRANSIENT BURNOUT IN AN ANNULAR TEST OF THE FIRST TEST SERIES OF BLOWDOWN ON A TUBULAR TEST SAFETY (IN ENGLISH)*		145190 P 12
PROBLEMS SAFETY PROJECT FIRST PHASE		145200 P 13
CONTAINMENTS PART 2 (IN GERMAN)*		145210 P 14
ANALYSIS OF THE D (IN FRENCH)*		145220 P 15
COMPREHENSIVE SUMMARY OF THE THEORETICAL STUDIES IN THE OF VOIDING DURING A RAPID FAILURE (BLOWDOWN) COMPARISON OF ENGINE* *A PROPOSED WIND OF DETONATION OF HYDROGEN-AIR MIXTURES AND PROPAGATION OF TENSION AND CYCLE (FORSION) (IN FRENCH)*		145230 P 16
A PRESSURIZED WATER REACTOR'S PRIMARY CIRCUIT BY FUEL RODS AND THEORETICAL STUDY OF THE BLOWDOWN OF THE SECONDARY APPLIED TO MULTI-COMPONENT, TWO-PHASE SYSTEMS - THE TO DESCRIBE THERMAL NON-EQUILIBRIUM FLOW OF WATER IN STRUCTURES (IN GERMAN)*		145240 P 17
PROPAGATION OF INFLUENCE OF THE INTERNAL CORROSION ON THE MECHANICAL OF A NUCLEAR POWER PLANT IN THE EXPERIMENTAL RESEARCH ON (IN GERMAN)*		145250 P 18
APPRAISAL OF SUBCOOLED BOILING AND ACOUSTIC NOISES WITH LOSS-OF-FLOW LOCAL FLOW BLOCKAGE EXPERIMENTS IN 37-PIN IN FRACTURE MECHANICS RESEARCH ON MATERIALS USED IN FAST HIGH MODEL DEVELOPMENT OF A CALCULATIONAL PROGRAM FOR THE (A REVIEW OF INTERNATIONAL PRELIMINARY EXPERIMENT ON LEAKAGE MONITORING USING PROPAGATION OF SIMULATED IMPROVEMENT IN THE MEASUREMENT TECHNIQUES OF ON MATERIALS USED IN FAST SODIUM-COOLED REACTORS (IN EXPERIMENTS IN 37-PIN SODIUM COOLED BUNDLES WITH GRID HAVING PRESSURE AND DENSITY AS THE INDEPENDENT VARIABLES (OF FRAGMENTS AND PROJECTILES OF DIFFERENT MASS BEHAVIOR OF A PWR FOLLOWING A LOCA FOR PURPOSES OF STUDYING THE STRAIN EVALUATION IN PRESSURE VESSELS LASER HOLOGRAPHIC AND STUDY OF THE OVERPRESSURE GENERATED BY THE DETONATION OF SEPTEMBER 30, 1978 (IN REPORT ON THE RESEARCH PROGRAM THE ANNUAL REPORT ON REACTOR SAFETY RESEARCH PROJECTS PARAMETERS MODEL OF THE FAILURE RATES OF THE VALVES OF FOR HYDRODYNAMIC ANALYSIS (IN FRENCH)*		145260 P 19
STUDY OF THE OF AIR AND STEAM MIXTURES, IN TRANSIENT CONDITIONS, ON A CONVERSION OF THE LASH-CODE ERAC-PI (VERSION 16.3) TO IBM OF THE CORE MELT ACCIDENT WITH MELT-1 IN A PWR AND A BWR TEST DURATION, TEST EFFICIENCY NOT 100%, AND 1 CFT OF 2 CORROSION OF THE MECHANICAL SIMULATION OF THE CHEMICAL OF A MASS-DENSITY METHOD FOR TRANSIENT TWO PHASE INVESTIGATIONS ON THE LOW FUEL ROD BEHAVIOR UNDER ACCIDENT TRANSIENT THERMAL-HYDRAULIC ANALYSIS* (THE CREEP RUPTURE AT NON-ANALYSIS OF BOILING-WATER REACTOR STUDY OF THE BLOWDOWN OF THE SECONDARY SIDE OF A STEEL TEST SECTION (STUDY ON THE CONDENSATION OF AIR AND INDEPENDENT SPADE) A COMPUTER SUBROUTINE FOR GENERATING COMPARISON WITH EXPERIMENTAL TESTS (IN GERMAN)* IMPACT OF AND STEAM MIXTURES IN TRANSIENT CONDITIONS, ON A STAINLESS EXPERIMENTAL TESTS ON MATCHES OF 304 AUSTENITIC REACTORS (IN THE SAFETY OF FRENCH INSTALLATIONS FOR THE AND SPECKLE PHOTOGRAPHY METHODS FOR DEFECT DETECTION AND PROPAGATION THROUGH UNCONFINED AND CONFINED HEMISPHERICAL CREEP RUPTURE AT NON-STEADY MELT ACCIDENT AFTER THE AFTER FAILURE OF THE CORE SUPPORT WATER EFFICIENT COMPUTATION OF THREE-DIMENSIONAL FLUID-CALCULATING PLASTIC DEFORMATION OF OF SIMULATED SONIC EMISSION-IMPULSES IN THICK WALLED PWR (IN FRENCH)* FRENCH THERMO-HYDRAULIC MODEL COMPREHENSIVE SUMMARY OF THE THEORETICAL SECTION 4 - A COMPUTER PROGRAM FOR USE IN NUCLEAR SAFETY (A PROPOSED WIND SHIFT MODEL FOR THE GERMAN REACTOR SAFETY EXPERIMENTAL GENERATOR (IN FRENCH)* (ANA EXPERIMENTAL AND THEORETICAL SPHERICAL AIR-HYDROGEN GASEOUS MIXTURES (IN EXPERIMENTAL RESURG THROUGH A LIQUID CORE CABLE (IN FRENCH)* (A FIRST APPROACH OF THE HARE EVENT PROBLEM BY THE OF EQUATIONS FOR HYDRODYNAMIC ANALYSIS (IN FRENCH)* (A PROBABILISTIC TRANSIENT CONDITIONS, ON A STAINLESS STEEL TEST SECTION (POWER PLANT WITH A PWR FOLLOWING A LOCA FOR PURPOSES OF LINGEN BWR)* (AN APPRAISAL OF DENSITY AS THE INDEPENDENT VARIABLES (IN SPADE) A COMPUTER OF A COOKING SYSTEM BLOWDOWN ACCIDENT AND THE SAFETY, JULY 1-SEPTEMBER 30, 1978 (IN GERMAN)*		145270 P 20
ON THE SAFETY, 30 EXPERIMENT (IN GERMAN)*		145280 P 21
SEAI (IN GERMAN)*		145290 P 22
REINFORCEMENT		145300 P 23
SECONDARY CONTAINMENT OF A PWR (IN GERMAN)*		145310 P 24
SECONDARY SIDE OF A STEAM GENERATOR (IN FRENCH)*		145320 P 25
(ANA SECTION (IN FRENCH)*		145330 P 26
OF AIR AND STEAM MIXTURES, (IN GERMAN)*		145340 P 27
DATA REDUCTION		145350 P 28
SELECTION OF EVENTS FOR A PROBABILISTIC EVALUATION OF PWR SEMIANNUAL REPORT 1978 (IN GERMAN)*		145360 P 29
SEPARATION (IN GERMAN)*		145370 P 30
RESEARCH PROJECT RD 50 (MUEL SERIES OF BLOWDOWN ON A TUBULAR TEST SECTION (IN GERMAN)*		145380 P 31
SEVERAL MODELS (IN FRENCH)*		145390 P 32
FOR THE CALCULATION OF THE RATE SHIFT MODEL FOR THE GERMAN REACTOR SAFETY STUDY (IN SHOCK WAVES IN THE ENVIRONMENT (IN GERMAN)*		145400 P 33
IDENTIFICATION SHORT ANALYSIS OF A PROGRESSIVE DESTRUCTION PROBLEM (SHOWING MANUFACTURING FAULTS (IN FRENCH)*		145410 P 34
UNSTABILIZATION OF SIDE OF A STEAM GENERATOR (IN FRENCH)*		145420 P 35
(ANA EXPERIMENTAL SIMPLE HOMOGENEOUS EQUILIBRIUM CRITICAL DISCHARGE MODEL (IN GERMAN)*		145430 P 36
BUBBLE DOUBLE (IN GERMAN)*		145440 P 37
A COMPUTER PROGRAM SIMULATED SONIC EMISSION-IMPULSES IN THICK WALLED STRUCTURES (IN GERMAN)*		145450 P 38
SIMULATION OF THE CHEMICAL STATE OF IRRADIATED UREO FUEL SINGLE AND MULTIPLE TUBE ARRAYS IN THE PRESSURE TRANSIENT SITU TESTING OF HIGH EFFICIENCY FILTERS AT AEE (IN ENGLISH)*		145460 P 39
SLEEP RATIO FROM MEASUREMENTS MADE IN LINGEN BWR)*		145470 P 40
SODIUM BOILING EXPERIMENT IN A 19-PIN BUNDLE (IN ENGLISH)*		145480 P 41
SODIUM COOLED BUNDLES WITH GRID SPACERS (IN ENGLISH)*		145490 P 42
SODIUM-COOLED REACTORS (IN GERMAN)*		145500 P 43
EMISSION MEASUREMENT SOLUTION OF THE NEUTRONIC EVOLUTION OF A MULTIDIMENSIONAL SOLUTIONS TO NEACP BENCHMARK BWR LATTICE CELL PROBLEMS* SONIC EMISSION ANALYSIS: EXPANDED INSTRUMENTATION AND SONIC EMISSION-IMPULSES IN THICK WALLED STRUCTURES (IN SONIC EMISSION ANALYSIS (SEA) (IN GERMAN)*		145510 P 44
SONIC EMISSION MEASUREMENTS IN FRACTURE MECHANICS RESEARCH SPACERS (IN ENGLISH)*		145520 P 45
LOCAL FLOW BLOCKAGE		145530 P 46
SPADE) A COMPUTER SUBROUTINE FOR GENERATING STEAM TABLES SPECIFIC REACTOR MATERIALS AND COMPONENT PARTS AT IMPACT SPECIFICATION OF CONDITIONS OF A NUCLEAR POWER PLANT WITH SPECIALLY PHOTOGRAPHY METHODS FOR DEFECT DETECTION AND SPHERICAL AIR-HYDROGEN GASEOUS MIXTURES (IN ENGLISH)*		145540 P 47
SPONSORED BY DMPT IN THE AREA OF REACTOR SAFETY, JULY 1- SEPTEMBER 30, 1978 (IN REPORT ON THE RESEARCH PROGRAM THE ANNUAL REPORT ON REACTOR SAFETY RESEARCH PROJECTS PARAMETERS MODEL OF THE FAILURE RATES OF THE VALVES OF FOR HYDRODYNAMIC ANALYSIS (IN FRENCH)*		145550 P 48
STUDY OF THE OF AIR AND STEAM MIXTURES, IN TRANSIENT CONDITIONS, ON A CONVERSION OF THE LASH-CODE ERAC-PI (VERSION 16.3) TO IBM OF THE CORE MELT ACCIDENT WITH MELT-1 IN A PWR AND A BWR TEST DURATION, TEST EFFICIENCY NOT 100%, AND 1 CFT OF 2 CORROSION OF THE MECHANICAL SIMULATION OF THE CHEMICAL OF A MASS-DENSITY METHOD FOR TRANSIENT TWO PHASE INVESTIGATIONS ON THE LOW FUEL ROD BEHAVIOR UNDER ACCIDENT TRANSIENT THERMAL-HYDRAULIC ANALYSIS* (THE CREEP RUPTURE AT NON-ANALYSIS OF BOILING-WATER REACTOR STUDY OF THE BLOWDOWN OF THE SECONDARY SIDE OF A STEEL TEST SECTION (STUDY ON THE CONDENSATION OF AIR AND INDEPENDENT SPADE) A COMPUTER SUBROUTINE FOR GENERATING COMPARISON WITH EXPERIMENTAL TESTS (IN GERMAN)* IMPACT OF AND STEAM MIXTURES IN TRANSIENT CONDITIONS, ON A STAINLESS EXPERIMENTAL TESTS ON MATCHES OF 304 AUSTENITIC REACTORS (IN THE SAFETY OF FRENCH INSTALLATIONS FOR THE AND SPECKLE PHOTOGRAPHY METHODS FOR DEFECT DETECTION AND PROPAGATION THROUGH UNCONFINED AND CONFINED HEMISPHERICAL CREEP RUPTURE AT NON-STEADY MELT ACCIDENT AFTER THE AFTER FAILURE OF THE CORE SUPPORT WATER EFFICIENT COMPUTATION OF THREE-DIMENSIONAL FLUID-CALCULATING PLASTIC DEFORMATION OF OF SIMULATED SONIC EMISSION-IMPULSES IN THICK WALLED PWR (IN FRENCH)* FRENCH THERMO-HYDRAULIC MODEL COMPREHENSIVE SUMMARY OF THE THEORETICAL SECTION 4 - A COMPUTER PROGRAM FOR USE IN NUCLEAR SAFETY (A PROPOSED WIND SHIFT MODEL FOR THE GERMAN REACTOR SAFETY EXPERIMENTAL GENERATOR (IN FRENCH)* (ANA EXPERIMENTAL AND THEORETICAL SPHERICAL AIR-HYDROGEN GASEOUS MIXTURES (IN EXPERIMENTAL RESURG THROUGH A LIQUID CORE CABLE (IN FRENCH)* (A FIRST APPROACH OF THE HARE EVENT PROBLEM BY THE OF EQUATIONS FOR HYDRODYNAMIC ANALYSIS (IN FRENCH)* (A PROBABILISTIC TRANSIENT CONDITIONS, ON A STAINLESS STEEL TEST SECTION (POWER PLANT WITH A PWR FOLLOWING A LOCA FOR PURPOSES OF LINGEN BWR)* (AN APPRAISAL OF DENSITY AS THE INDEPENDENT VARIABLES (IN SPADE) A COMPUTER OF A COOKING SYSTEM BLOWDOWN ACCIDENT AND THE		145560 P 49
		145570 P 50
		145580 P 51
		145590 P 52
		145600 P 53
		145610 P 54
		145620 P 55
		145630 P 56
		145640 P 57
		145650 P 58
		145660 P 59
		145670 P 60
		145680 P 61
		145690 P 62
		145700 P 63
		145710 P 64
		145720 P 65
		145730 P 66
		145740 P 67
		145750 P 68
		145760 P 69
		145770 P 70
		145780 P 71
		145790 P 72
		145800 P 73
		145810 P 74
		145820 P 75
		145830 P 76
		145840 P 77
		145850 P 78
		145860 P 79
		145870 P 80
		145880 P 81
		145890 P 82
		145900 P 83
		145910 P 84
		145920 P 85
		145930 P 86
		145940 P 87
		145950 P 88
		145960 P 89
		145970 P 90
		145980 P 91
		145990 P 92
		146000 P 93
		146010 P 94
		146020 P 95
		146030 P 96
		146040 P 97
		146050 P 98
		146060 P 99
		146070 P 100

#DEVELOPMENT OF A RESEARCH PROGRAM TO MODEL CONTAINMENT	#COMPREHENSIVE	SUCTION SYSTEM FOR INSTALLATIONS AND FITTINGS (IN GERMAN)*	144020 P 43
#EXPERIENCE OF THE PRIMARY COOLING CIRCUITS IN WATER-COOLED REACTORS, THE CORE MELT ACCIDENT AFTER THE AFTER FAILURE OF THE CORE MODEL FOR THE TWO-DIMENSIONAL CALCULATION OF THE WATER POOL MEDIA FOR THE INTENSIVE TRAINING AND	#DEVELOPMENT AND DURATION, TEST EFFICIENCY NOT 100%, AND 1 OUT OF 2 STANDBY WATER POOL SWELL IN THE CONDENSATION CHAMBER OF A REACTOR TRANSFER COEFFICIENT IN THE CONTAINMENT DURING A COOLING TRANSIENT IN LIGHT-WATER	#COMPARATIVE INVESTIGATIONS OF A COOLING #DEVELOPMENT OF A SUCTION SYSTEM FOR REACTOR COMPONENTS (IN GERMAN)*	145155 P 22
FOR THE TESTING OF A FRACTURE SAFETY DEVICE PROTECTION	#INSTRUMENTATION	SYSTEM FOR THE DAB-MULTIPLE TUBE RESEARCH PROGRAM (IN GERMAN)*	145790 P 13
LASL-CODE TRACE (PI EVERSION 10.3) TO IOM STANDARD OPERATING	#CHARACTERISTICS AND RESULTS OF THE INVESTIGATIONS OF THE RADIOACTIVITY IN THE PRIMARY PROBLEM BY THE STUDY OF THE RELIABILITY OF THE PROTECTION TRAINING AND	SYSTEM MVS (WITH FURTHER-EXTENDED COMPILER) (IN ENGLISH)*	143960 P 19
#DEVELOPMENT AND SYNTHESIS OF AN EDUCATIONAL EXPERIMENT BY DIRECT ELECTRICAL HEATING METHOD (ZRE012- H2O DISCHARGE MODEL APPLIED TO MULTI-COMPONENT, TWO-PHASE ANALYSIS (IN FRENCH)*	#STUDY OF THE STABILITY OF VARIOUS CONTAINMENT (IN GERMAN)*	SYSTEM OF EQUATIONS DERIVED FROM PARTIAL HYPERBOLICS (IN GERMAN)*	147101 P 7
#INVESTIGATION AND DEVELOPMENT OF TESTING-NONNEGLECTIBLE TEST DURATION, TEST EFFICIENCY NOT THERMAL NON-EQUILIBRIUM FLOW OF WATER IN SIMPLE PIPE		SYSTEM OF PRESSURIZED WATER REACTORS (IN GERMAN)*	144590 P 10
		SYSTEM OF THE FESSENHEIM 1 PWR REACTOR (IN ENGLISH)*	143779 P 7
		SYSTEM USING COMBINATION OF MEDIA FOR THE INTENSIVE SYSTEMS (IN JAPANESE)*	143760 P 21
		SYSTEMS - THE COMPUTER PROGRAMS CRITS AND CRITICALLY CRITICAL SYSTEMS AND DESCRIPTION OF EQUATIONS FOR HYDRODYNAMIC SYSTEMS LIMITING THE RE-CONCENTRATION IN THE DRX SYSTEMS WHICH ARE UNCONNECTED AND WAITING FOR PERIODIC SYSTEMS* #HUBLE BUBBLE III: A COMPUTER PROGRAM TO DESCRIBE	140842 P 29 140843 P 23 143700 P 2 147473 P 2 145290 P 1 140447 P 32
#PAVED A COMPUTER SUBROUTINE FOR GENERATING STEAM	#IMPROVEMENT IN THE MEASUREMENT	TECHNIQUES OF SONIC-EMISSION ANALYSIS (SEA) (IN GERMAN)*	144027 P 20
REACTOR	#REPORT OF THE FEDERAL MINISTER FOR RESEARCH AND PROJECTS SPONSORED BY THE MINISTRY FOR RESEARCH AND TESTS ON BATCHET OF 304 AUSTENITIC STEEL, AT ROOM	TECHNOLOGY CONCERNING RESEARCH PROJECTS IN THE AREA OF TECHNOLOGY OF THE FEDERAL REPUBLIC OF GERMANY, 1978 (IN TEMPERATURE (IN FRENCH)*	147483 P 10 147467 P 16 146872 P 4
ENGRSS, REACTIVITY	#COMPREHENSIVE DAMAGE ANALYSIS FOR HIGH	TEMPERATURE GAS REACTORS, PHASE II, WATER INGRESS, AIR	146070 P 14
#CRACK RUPTURE AT NON-STEADY STRESS AND THE SUBSEQUENT THERMAL TRANSIENT IN LIGHT-WATER AND HIGH	#SHORT ANALYSIS OF A PROGRESSIVE DISTORTION PROBLEM	TEMPERATURE LOADING CONDITIONS (IN ENGLISH)*	143089 P 19
	#EVALUATION OF THE 25-REC BUNDLE ANALYSIS OF THE NONNUCLEAR LOFT-TESTS LI-4 (PRE AND POST	TEMPERATURE REACTORS (IN GERMAN)*	145074 P 12
	CIRCULATION ON LOCA PHENOMENA IN LARGEST COOL LEG #MOSA-II UNCONNECTED AND WAITING FOR PERIODIC TESTING-NONNEGLECTIBLE	TENSILE AND CYCLE TORSION (IN FRENCH)*	146871 P 8
	WAITING FOR PERIODIC TESTING-NONNEGLECTIBLE TEST DURATION, #THE CONTAINMENT	TEST (RD-37C) WITH THE CALCULATIONAL PROGRAM (IN GERMAN)*	143902 P 14
	IN PELLETS ON FUEL FAILURE BEHAVIOR (ENRICHMENT PARAMETER	TEST CALCULATIONS (IN GERMAN)*	143905 P 15
	#POST-EXPERIMENT CALCULATION OF THE NON-NUCLEAR LOFT	TEST DATA REPORT 14 EFFECTS OF ECCS INJECTION AND PUMP	143091 P 27
	MIXTURES, IN TRANSIENT CONDITIONS, ON A STAINLESS STEEL	TEST DURATION, TEST EFFICIENCY NOT 100%, AND 1 OUT OF 2	145290 P 1
	#FLOW REDUCTION TRANSIENT BURST IN AN ANNULAR	TEST EFFICIENCY NOT 100%, AND 1 OUT OF 2 STANDBY SYSTEM 1	145290 P 1
	REDUCTION OF THE FIRST TEST SERIES OF BLOWDOWN ON A TUBULAR	TEST FACILITY (EXPERIMENTS C AND D) (IN GERMAN)*	143800 P 12
	LOOP (IN FRENCH)*	TEST IN NSRR (IN ENGLISH & JAPANESE)*	143974 P 20
	#DATA REDUCTION OF THE FIRST	TEST LI-5 (IN GERMAN AND ENGLISH)*	146074 P 16
	REACTOR COMPONENTS (IN	TEST SECTION (IN FRENCH)*	144000 P 7
	#IN-SITU	TEST SECTION (IN JAPANESE & ENGLISH)*	140844 P 29
	#SYSTEMS WHICH ARE UNCONNECTED AND WAITING FOR PERIODIC	TEST SECTION ON OREGA LOOP (IN FRENCH)*	143750 P 3
	CONCRETE, CALCULATION AND COMPARISON WITH EXPERIMENTAL	TEST SERIES OF BLOWDOWN ON A TUBULAR TEST SECTION ON OREGA	143750 P 3
	#RELAP-RGVS ANALYSIS OF THE NONNUCLEAR LOFT-	TESTING OF A FRACTURE SAFETY DEVICE PROTECTION SYSTEM FOR	145155 P 22
	INTERNAL PRESSURE (IN FRENCH)*	TESTING OF HIGH EFFICIENCY FILTERS AT ABB #INFRIT*	140877 P 31
	#MODELS FOR THE ESTIMATION OF FLOW USING ULTRASONIC	TESTING-NONNEGLECTIBLE TEST DURATION, TEST EFFICIENCY NOT	145290 P 1
	TEMPERATURE (IN FRENCH)*	TESTS (IN GERMAN)*	145036 P 21
	#AFZK/NU POWER RAMPS FUEL ROD IRRADIATION	TESTS LI-4 (PRE AND POST TEST CALCULATIONS) (IN GERMAN)*	143905 P 10
	FUEL ROD BEHAVIOR UNDER ACCIDENT #STATUS AND RESULTS OF THE	TESTS ON BUCKLING OF ELLIPSOIDAL VESSEL HEADS UNDER	146870 P 6
	THE INITIAL BLOWDOWN PHASE, PARTS A, B, & #EXPERIMENTAL AND	TESTS ON NUCLEAR REACTORS (IN GERMAN)*	146486 P 12
	PROGRAM HS 50 (MODEL	TESTS ON BATCHET OF 304 AUSTENITIC STEEL, AT ROOM	146872 P 4
	#COMPREHENSIVE SUMMARY OF THE	TESTS 1976/77 (GERMAN)*	146075 P 23
	A STEAM GENERATOR (IN FRENCH)*	THEORETICAL AND EXPERIMENTAL INVESTIGATIONS ON THE LWR	143740 P 19
	LOCA (IN FRENCH)*	THEORETICAL RESEARCH ON THE THERMAL HYDRAULIC BEHAVIOR IN	143901 P 24
	#CUPIDEN: A CODE DESCRIBING THE	THEORETICAL STUDIES ON THE D-SERIES OF THE RESEARCH	146272 P 11
	PRODUCT RELEASE FROM A DEFECTED FUEL ROD - EFFECT OF	THEORETICAL STUDY OF THE BLOWDOWN OF THE SECONDARY SIDE OF	148734 P 5
	PARTS A, B, & #EXPERIMENTAL AND THEORETICAL RESEARCH ON THE	THERMAL AND MECHANICAL BEHAVIOR OF A PWR FUEL ROD DURING A	146873 P 1
		THERMAL CYCLING (IN ENGLISH)*	144095 P 5
		THERMAL HYDRAULIC BEHAVIOR IN THE INITIAL BLOWDOWN PHASE,	143901 P 24
		THERMAL INTERACTIONS BETWEEN CARBOND AND WATER*	146870 P 22
		THERMAL INTERACTIONS BETWEEN CARBON AND WATER*	146870 P 22
		THERMAL NON-EQUILIBRIUM FLOW OF WATER IN SIMPLE PIPE	140447 P 32
		THERMAL TRANSIENT IN LIGHT-WATER AND HIGH TEMPERATURE	146874 P 12
		THERMAL-HYDRAULIC ANALYSIS*	145040 P 31
		THERMO-HYDRAULIC BEHAVIOR IN A PWR CORE DURING A LOCA (IN	144249 P 27
		THERMO-HYDRAULIC STUDIES FOR THE DEVELOPMENT OF SAFETY	143750 P 6
		THERMODYNAMICS OF CONDENSED UO ₂ : A REVIEW OF THE	148840 P 34
		THERMODYNAMICS OF MOLTEN UO ₂ *	146845 P 33
		THERMODYNAMICS OF UO ₂ *	143773 P 33
		THERMOHYDRAULIC MODELISATION DURING THE REFOLDING PHASE	143928 P 1
		THESE OCCURRENCES IN FAULT TREE ANALYSIS (IN GERMAN)*	147779 P 24
		THICK WALLED STRUCTURES (IN GERMAN)*	144150 P 10
		THREE-DIMENSIONAL FLUID-STRUCTURE INTERACTIONS DURING	146804 P 20
		THROUGH A LIQUID COOLANT CABU 1 (IN FRENCH)*	143110 P 2
		THROUGH UNCONNECTED AND CONFINED HEMISPHERICAL STRATIFIED	143078 P 18
		TIME-OF-FLIGHT DATA, FINAL REPORT (IN GERMAN)*	148306 P 12
		TIRIX 4 - A COMPUTER PROGRAM FOR USE IN NUCLEAR SAFETY	147602 P 33
		TORSION (IN FRENCH)*	146871 P 8
		TRACE (PI EVERSION 10.3) TO IOM STANDARD OPERATING SYSTEM	143960 P 19

PROGRAM TO DESCRIBE THERMAL NON-EQUILIBRIUM FLOW OF ANALYSIS FOR HIGH TEMPERATURE GAS REACTORS, PHASE II.
 #A PROBABILISTIC STUDY OF VESSEL BURST IN LIGHT
 #COMPUTER MODEL FOR THE TWO-DIMENSIONAL CALCULATION OF THE #ALARM-PI: A COMPUTER PROGRAM FOR PRESSURIZED INVESTIGATION OF THE HYDROGEN DISTRIBUTION IN A LIGHT INVESTIGATION OF THE HYDROGEN DISTRIBUTION IN A LIGHT #ANALYSIS OF COILING-STRUCTURE INTERACTIONS DURING BLOWDOWN OF A PRESSURIZED MANUFACTURING FAULTS (IN #CONTAMINATION OF A PRESSURIZED FOR THE STORAGE OF IRRADIATED FUEL ELEMENTS FROM LIGHT OF THE RADIOACTIVITY IN THE #ARY SYSTEM OF PRESSURIZED #THERMAL INTERACTIONS BETWEEN CERAMIC AND #THERMAL INTERACTIONS BETWEEN CERAMIC AND OF PRESSURE TRANSIENTS IN FULL-PRESSURE CONTAINMENT OF CONTAINMENT FROM THE COOLANT BLOWDOWN OF A IN A MULTIPLE COMPARTMENT CONTAINMENT BY PRESSURE IN CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN BETWEEN A ONE- AND TWO-DIMENSIONAL CALCULATION OF A OF HYDROGEN-AIR MIXTURES AND PROPAGATION OF SHOCK NONNEGLECTABLE TEST DURATION; TEST EFFICIENCY NOT #SYSTEMS ENGLISH# #A PROPOSED #IN-SITU TESTING OF HIGH EFFICIENCY FILTERS AT AEE THE PRIMARY #INVESTIGATION OF THE PHENOMENA OCCURRING THE PRIMARY #INVESTIGATION OF THE PHENOMENA OCCURRING THE PRIMARY #INVESTIGATION OF THE PHENOMENA OCCURRING WATER IN SIMPLE PIPE SYSTEMS# #BUBBLE BUBBLE III: A COMPUTER 140447 P 22 WATER INGRESS, AIR INGRESS, REACTIVITY EXCURSIONS (IN WATER LOSS (IN FRENCH)# 148670 P 18 WATER FOUL-SWELL IN THE CONDENSATION CHAMBER OF A REACTOR 146865 P 6 WATER REACTOR BLOWDOWN ANALYSIS (IN ENGLISH)# 145757 P 15 WATER REACTOR CONTAINMENT FOLLOWING A LOSS-OF-COOLANT 147502 P 25 WATER REACTOR CONTAINMENT FOLLOWING A LOSS-OF-COOLANT 144582 P 11 WATER REACTOR STEAM CHUCCING (IN ENGLISH)# 144286 P 12 WATER REACTOR-FLUX (IN GERMAN)# OF THREE-DIMENSIONAL FLUID- 146800 P 17 WATER REACTOR'S PRIMARY CIRCUIT BY FUEL RODS SHAKING 146804 P 20 WATER REACTORS (IN FRENCH)# SAFETY OF FRENCH INSTALLATIONS 143383 P 4 WATER REACTORS (IN GERMAN)# EXPERIMENTAL INVESTIGATIONS 147103 P 3 WATER# 144596 P 18 WATER# 143510 P 22 WATER# 148048 P 22 WATER-COOLED NUCLEAR POWER PLANTS (IN GERMAN)# 145626 P 16 WATER-COOLED REACTOR - INTERIM RESEARCH REPORT C 13 (IN 145875 P 10 WATER-COOLED REACTORS WITH REFRIGERATED CONDENSER (IN 143329 P 10 WATER-COOLED REACTORS, CONDENSATION IN CONTAINMENT BY 146273 P 12 WATER-COOLED REACTORS, QUICK LOOK REPORT EXPERIMENT DIS (145846 P 10 WATER-COOLED REACTORS, SUPPLEMENTAL RESEARCH DOCUMENTATION 145847 P 11 WATER-VAPOR NOZZLE FLOW (IN GERMAN)# #COMPANION 143211 P 19 WAVES IN THE ENVIRONMENT (IN GERMAN)# OF DETONATION 144280 P 12 WHICH ARE UNCONNECTED AND WAITING FOR PERIODIC TESTING- 145296 P 1 WIND SHIFT MODEL FOR THE GERMAN REACTOR SAFETY STUDY (IN 146006 P 21 WINDRITH# 145877 P 31 WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF 145846 P 10 WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF 145847 P 11 WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF 146273 P 12

#NG/WATER VAPOUR EXPLOSIONS - ESTIMATES OF PRESSURES AND YIELDS# 148043 P 33

INTERACTION EXPERIMENT BY DIRECT ELECTRICAL HEATING METHOD (ZRE032- H2O SYSTEM) (IN JAPANESE)# #FUEL COOLANT 148842 P 29 OF THE INTERNAL CORROSION ON THE MECHANICAL PROPERTIES OF ZRY-4 TURBINE (IN ENGLISH)# CALDE FUEL; INFLUENCE 148673 P 20

NUREG/CR-1101
 ORNL/NUREG/NSIC-170
 (Vol. VII of TID-3362
 Dist. Category AE

INTERNAL DISTRIBUTION

- | | |
|---------------------|---|
| 1. Seymour Baron | 18. G. T. Mays |
| 2. J. R. Buchanan | 19. F. R. Mynatt |
| 3. C. A. Burchsted | 20. D. S. Queener |
| 4. W. R. Casto | 21. J. L. Rich |
| 5. R. O. Chester | 22. R. L. Scott |
| 6. W. B. Cottrell | 23. S. D. Swisher |
| 7. A. L. Crawford | 24. H. E. Trammell |
| 8. C. E. Davis | 25. D. B. Trauger |
| 9. R. B. Gallaher | 26. M. L. Winton |
| 10. C. T. Garten | 27-28. Central Research Library |
| 11. E. W. Hagen | 29. Y-12 Document Reference
Section (DRS) |
| 12. P. M. Haas | 30. Laboratory Records Department
Record Copy (LRD-RC) |
| 13. F. A. Heddleson | 31-47. Nuclear Safety Information
Center (NSIC) |
| 14. M. Heiskell | |
| 15. R. D. Hurt | |
| 16. W. H. Jordan | |
| 17. Milton Levenson | |

EXTERNAL DISTRIBUTION

48. Office of Assistant Manager, Energy Research and Development,
 Oak Ridge Operations Office, Oak Ridge, TN 37830
- 49-50. DOE Technical Information Center, P.O. Box 62, Oak Ridge, TN 37830
- 51-450. Given distribution as shown in Category AE (NTIS-10)

NUCLEAR SAFETY

A BIMONTHLY REVIEW JOURNAL PREPARED BY NSIC

Nuclear Safety covers significant developments in the field of nuclear safety.

The scope is limited to topics relevant to the analysis and control of hazards associated with nuclear reactors, operations involving fissionable materials, and the products of nuclear fission.

Primary emphasis is on safety in reactor design, construction, and operation; however, safety considerations in reactor fuel fabrication, spent-fuel processing, nuclear waste disposal, handling of radioisotopes, and related operations are also treated.

Qualified authors are invited to submit interpretive review articles, which will be reviewed for technical accuracy and pertinency. Authors will be advised as soon as possible of acceptance or suggested changes. Send inquiries or 3 copies of manuscripts (with the draftsman's original line drawing plus 2 copies, and with continuous-tone glossy prints of photographs plus 2 copies) to Wm. B. Cottrell, Oak Ridge National Laboratory, P.O. Box Y, Oak Ridge, Tennessee 37830.

Nuclear Safety is prepared by the Nuclear Safety Information Center at Oak Ridge National Laboratory for the U.S. Nuclear Regulatory Commission and the U.S. Department of Energy. For subscriptions, address Superintendent of Documents, U.S. Government Printing Office, Washington, D.C. 20402. The subscription rate is \$14.00 per year. Below is an order blank for your convenience.

U.S. GOVERNMENT PRINTING OFFICE
DIVISION OF PUBLIC DOCUMENTS
WASHINGTON, D.C. 20402

OFFICIAL BUSINESS

RETURN AFTER 5 DAYS

POSTAGE AND FEES PAID
U.S. GOVERNMENT PRINTING OFFICE

Name _____
Street _____
City _____ State _____ ZIP _____

To Ensure Prompt, Accurate Shipment, Please Print or Type Address on Mailing Label Above

MAIL ORDER FORM TO:

Superintendent of Documents, U.S. Government Printing Office, Washington, D.C., 20402

Enclosed find \$ _____ (check, money order, or Superintendent of Documents coupons)

Please send me _____ subscriptions to *Nuclear Safety* at
\$14.00 per subscription. (Single issues are sold at \$2.40 per issue.)

Please charge this order _____
to my Deposit Account _____
No. _____ Name _____
Street _____
City _____ State _____ ZIP _____

FOR USE OF SUPT. DOCS.

___ Enclosed ___
___ To be mailed ___
___ later ___
___ Subscription ___
___ Refund ___
___ Coupon refund ___
___ Postage ___

POSTAGE AND REMITTANCE: Postpaid within the United States, Canada, Mexico, and all Central and South American countries except Argentina, Brazil, Guyana, French Guiana, Surinam, and British Honduras. For these and all other countries, add \$3.50 for each annual subscription; for a single issue, add one-fourth of the single-issue price. Payment should be by check, money order, or document coupons, and MUST accompany order. Remittances from foreign countries should be made by international money order or draft on an American bank payable to the Superintendent of Documents or by UNESCO book coupons.

AVAILABILITY OF NSIC DOCUMENTS (Continued)

ORNL/ NSIC	Title	Price*
122	Annotated Bibliography of Safety-Related Occurrences in Nuclear Power Plants as Reported in 1974, R. L. Scott and R. B. Gallaher, May 1975	\$15.00
123	Nuclear Power: Accident Probability, Risks, and Benefits: A Bibliography, NSIC Staff, Feb. 1976	\$ 6.00
118	Siting of Nuclear Facilities, Selections from <i>Nuclear Safety</i> , J. R. Buchanan, July 1976	\$ 9.75
125	LMFBR Safety, 1. Review of Current Issues and Bibliography of Literature (1960-1969), J. R. Buchanan and G. W. Keilholtz, Sept. 1976	\$12.75
126	Annotated Bibliography of Safety-Related Occurrences in Boiling-Water Nuclear Power Plants as Reported in 1975, R. L. Scott and R. B. Gallaher, July 1976	\$11.00
127	Annotated Bibliography of Safety-Related Occurrences in Pressurized-Water Nuclear Power Plants as Reported in 1975, R. L. Scott and R. B. Gallaher, July 1976	\$10.75
128	HTGR Safety, 1. Review of Current Issues and Bibliography of Literature (1960-1977), J. R. Buchanan and G. W. Keilholtz, July 1978	\$ 6.50
129	LMFBR Safety, 2. Review of Current Issues and Bibliography of Literature (1970-1972), J. R. Buchanan and G. W. Keilholtz, Dec. 1976.	\$13.00
130	Bibliography of Reports on Research Sponsored by the NRC Office of Nuclear Regulatory Research, November 1975-June 1976, J. R. Buchanan, Oct. 1976.	\$ 5.50
131	LMFBR Safety, 3. Review of Current Issues and Bibliography of Literature (1972-1974), J. R. Buchanan and G. W. Keilholtz, April 1977	\$12.75
132	LMFBR Safety, 4. Review of Current Issues and Bibliography of Literature (1974-1975), J. R. Buchanan and G. W. Keilholtz, April 1977	\$13.00
133	Index to <i>Nuclear Safety</i> , A Technical Progress Review by Chronology, Permuted Title, and Author, Vol. 11, No. 1 Through Vol. 17, No. 6, Wm. B. Cottrell and Ann Klein, April 1977	\$ 5.50
134	Reports Distributed Under the NRC Light-Water Reactor Safety Technical Exchange, Wm. B. Cottrell and D. S. Sharp, April 1977	\$ 4.00
135	Bibliography of Reports on Research Sponsored by the NRC Office of Nuclear Regulatory Research, J. R. Buchanan, March 1977	\$ 5.50
136	Design Data and Safety Features of Commercial Nuclear Power Plants, Vol. VI (Sixth Volume of ORNL/NSIC-55), F. A. Heddleson, June 1977	\$ 5.50
137	Annotated Bibliography of Safety-Related Occurrences in Boiling-Water Nuclear Power Plants as Reported in 1976, R. L. Scott and R. B. Gallaher, Sept. 1977	\$11.75
138	Annotated Bibliography of Safety-Related Occurrences in Pressurized-Water Nuclear Power Plants as Reported in 1976, R. L. Scott and R. B. Gallaher, Aug. 1977	\$12.00
139	LMFBR Safety, 5. Review of Current Issues and Bibliography of Literature: Vol. 5, 1975-1976, J. R. Buchanan and G. W. Keilholtz, July 1977	\$13.75
140	Structural Integrity of Materials in Nuclear Service: A Bibliography, F. A. Heddleson, July 1977	\$ 9.25
141	Summary Data for U.S. Commercial Nuclear Power Plants, F. A. Heddleson	
142	Reports Distributed Under the NRC Light-Water Reactor Safety Technical Exchange, Vol. III (Jan.-June 1977), D. S. Sharp and Wm. B. Cottrell, Oct. 1977	\$ 4.00
143	Bibliography of Reports on Research Sponsored by the NRC Office of Nuclear Regulatory Research - Jan.-June 1977, J. R. Buchanan, Oct. 1977	\$ 5.50
144	Reactor Operating Experiences 1975-1977, U.S. Nuclear Regulatory Commission	
145	Bibliography of Reports on Research Sponsored by the NRC Office of Nuclear Regulatory Research - July-Dec. 1977, J. R. Buchanan	
146	Reports Distributed in 1977 Under the NRC Light-Water Reactor Safety Technical Exchange, Wm. B. Cottrell and D. S. Sharp	
147	Index to <i>Nuclear Safety</i> , A Technical Progress Review by Chronology, Permuted Title, and Author, Vol. 11, No. 1 Through Vol. 18, No. 6, Wm. B. Cottrell and Ann Klein	

*Due to additional mailing and handling costs, the foreign price per document is double that of the domestic price.

†Reports after this date are published as ORNL/NUREG/NSIC reports.