ornl

OAK
RIDGE
NATIONAL
LABORATORY

UNION
CARBIDE

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

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

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 Vol.I	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
55 Vol.II	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
55 Vol.III	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
55 Vol.IV	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
	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 *Nuclear Safety*, J. R. Buchanan, Mar. 1974	$15.00
101	Indexed Bibliography on Environmental Monitoring for Radioactivity, B. L. 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 *Nuclear Safety*, 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 *Nuclear Safety*, 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. Categ ry 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 Technolog 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 t e
DEPARTMENT C 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

CONTENTS

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

Report No.	Period Covered	Publication Date
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 mestic 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

[a]The number of proprietary documents received are listed in parentheses.

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

[a]Includes 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 Jaunary 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:

Page Range	Domestic[a,b] Price
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*

[a]Double the cost per copy for foreign price.

[b]These prices are subject to change by NTIS.

[c]Add $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

DOCUMENT TITLE

AUTHOR – AUTHORS ARE LISTED ALPHABETICALLY IN THE *AUTHOR INDEX* ALONG WITH ACCESSION NUMBERS

ACCESSION NUMBER – SERIAL NUMBER ASSIGNED TO THE DOCUMENT BY NSIC

BIBLIOGRAPHIC DATA

PUBLICATION DATE

137347
WEINBERG AM
OUTLINE FOR AN ACCEPTABLE NUCLEAR FUTURE
INST. FOR ENERGY ANALYSIS, OAK RIDGE, TENN.
ORAU/IEA(0)--77-17 +, 26 PP., JULY 1977

NUCLEAR ENERGY IS LIKELY TO DEVELOP IN TWO PHASES. PHASE I, BASED ON BURNER REACTORS, IS SELF-LIMITING BECAUSE THE RESERVE OF URANIUM IS LIMITED. PHASE II, BASED ON BREEDERS, MIGHT LAST FOR AN EXTREMELY LONG TIME. OPPOSITION TO PHASE I MIGHT BE REDUCED IF AN ACCEPTABLE PHASE II CAN BE CONSTRUCTED. ELEMENTS OF AN ACCEPTABLE PHASE II MIGHT INCLUDE ISOLATED AND COLLOCATED ENERGY CENTERS WITH RESIDENT IAEA INSPECTORS; HEAVIER SECURITY; PROFESSIONALIZATION OF THE NUCLEAR CADRE; IMMORTALITY OF THE OPERATING ENTITIES; AND SEPARATION OF GENERATION AND DISTRIBUTION. THE PROPOSED SITING POLICY, WITH IAEA RESIDENT INSPECTION, MIGHT BE MORE PROLIFERATION-RESISTANT THAN IS THE CURRENT DISPERSED SYSTEM.

ABSTRACT OF ORIGINAL DOCUMENT

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

POWER PLANT, NUCLEAR + URANIUM + REACTOR, LWR + REACTOR, BREEDER + IAEA + PROLIFERATION + ENERGY CENTER

CORPORATE AUTHOR

REPORT NUMBER

AVAILABILITY OF DOCUMENT

KEYWORDS – LISTED ALPHABETICALLY IN THE *KEYWORD INDEX* WITH THE ACCESSION AND PAGE NUMBERS

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
UNIDIMENSIONAL MODELS FOR DIPHASE FLOWS (IN FRENCH)
CEA DEPARTEMENT DES REACTEURS A EAU, FRANCE
DRE/STRE/LET/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-NONNEGLIGIBLE 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
CHAGROT 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 NUCLEIARE. 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
SERMA/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 THERMOHYDRAULIC 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 THERMOHYDRAULIC 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 THERMOHYDRAULIC 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 MODELISATION AND SOME CLASSICAL HEAT TRANSFER CORRELATIONS, TO DESCRIBE THE REFLOODING PHASE.

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

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

143756
BRUYERE 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 + FRRSR-156 +. 9 PPS, APRIL 13, 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 D
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 + FRRSR-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 CLASSED INTO FIVE CATEGORIES ACCORDING TO THE TYPE OF CORRELATION USED: (1) MARTINELLI-NELSONS'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
FRRSR-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. 20555(08 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 + FRRSR-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 ALGORYTHM, 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 BUBLE RISING THROUGH A LIQUID CODE CABU I (IN FRENCH)
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
S.E.S.T.R. 12 + FRRSR-152 +. 30 PPS, 7 FIGS, 13 REFS, JUNE 20, 1978

143210 *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 EXERTED 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
PUIT JC + LEFORT G
THE SAFETY OF FRENCH INSTALLATIONS FOR THE STORAGE OF IRRADIATED FUEL ELEMENTS FROM LIGHT WATER REACTORS (IN FRENCH)
CEA DEPARTEMENT OF SURETE NUCLEAIRE, FRANCE
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE DSN 235 + FRNSR-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
CAUMETTE P + CHEISSOUX JL + GARCIA JL
CALCULATING PLASTIC DEFORMATION OF STRUCTURES (IN FRENCH)
CEA CENTRE D'ETUDES NUCLEAIRES DE CADARACHE, FRANCE
DRE/STRE/LMA 78/154 + FRNSR-149 +. 54 PPS. FIGS. JULY 1978

THE METHODS FOR CALCULATING PLASTIC DEFORMATION OF STRUCTURES ARE PRESENTED, AS THEY ARE USED IN THE CASTEM SYSTEM, DEVELOPED BY THE DEPARTEMENT DES ETUDES MECANIQUES 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 + CARNINO A + GEORGIN JP + SIGNORET JP
USE OF MARKOV PROCESSES FOR RELIABILITY PROBLEMS (IN ENGLISH)
CEA DEPARTMENT DE SURETE NUCLEAIRE, FRANCE
DSN-234(E) + FRNSR-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 + FRNSR-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

143779
BLIN A + DUCHEMIN B + CARNINO A
PATREC, A COMPUTER CODE FOR FAULT TREE CALCULATIONS (IN ENGLISH)
CEA DEPARTMENT DE 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 PATERN RECOGNITION APPROACH AND PROGRAMMING TECHNIQUES FROM IBM PL/1 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)

VAILABILITY - 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 + RICQUE R + BOURGINE R
BLOWDOWN OF A PART OF THE LOOP OMEGA INCLUDING A 36 DIRECT HEATED ROD BUNDLE (IN FRENCH)
COMMISSARIAT A L'ENERGIE ATOMIQUIE, 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 + BROUARD 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 CONTROLED 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 98/78 + 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

148734 *CONTINUED*
SIRETA X
ANA EXPERIMENTAL AND THEORETICAL STUDY OF THE BLOWDOWN OF THE SECONDARY SIDE OF A STEAM GENERATOR (IN FRENCH)
FRAMATOME, FRANCE
TP/CT/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 EXPER MENTAL RESEARCH WORK PERFORMED AT FRAMATOME IN COLLABORATION WITH THE CEA IN ORDER _ STUDY THE BLOWDOWN OF THE SECONDARY SIDE OF A STEAM GENERATOR. IT SUMMARIZES THE WORK RELATED TO THE STUDY AND THE MODELING 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 + HARRER 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 FULL 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 HYPOTHESIS: 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
DUCO J + GOBERT T
PROTECTION OF NUCLEAR POWER PLANTS (NPPS) AGAINST EXTERNAL EVENTS: EARTHQUAKES, FIRES, EXPLOSIONS AND AIRCRAFT CRASHES (IN ENGLISH)
COMMISSARIAT A L'ENERGIA ATOMIC, FRANCE + ELECTRICITE DE FRANCE
FRRSR-154 +. 10 PPS, PAPER PRESENTED AT SESSION 04/7 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
DUFRESNE J + CARNINO A + QUERO J + LUCIZ AC
FRACTURE PROBABILITY EVALUATION OF A LWR PRESSURE VESSEL (IN ENGLISH)
CEA DEPARTEMENT DE SURETE NUCLEARE, FRANCE
FRRSR-158 +. 13 PPS, PAPER PRESENTED AT SESSION 04 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'; BASLE, 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 PROBABILISITIC 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, LMFBR

144415
BROSSARD J + DUCO J + GOBERT 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 THEORICAL 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 (LMFBR) + R AND D PROGRAM + SAFETY PROGRAM

143758
REOCREUX 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'ENERGIA 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 REFLOOD, 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 REFLOODING

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

143339 *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 + QUENEE R
A FIRST APPROACH OF THE RARE EVENT PROBLEM BY THE STUDY OF THE RELIABILITY OF THE PROTECTION SYSTEM OF THE FESSENHEIM I PWR REACTOR (IN ENGLISH)
CEA DEPARTMENT DE SURETE NUCLEAIRE, FRANCE + FRAMATOME, FRANCE
FRRSR-165 +. 15 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 (EXP -5) AND 10 (EXP -8) PER REACTOR YEAR AND WHICH COULD RESULT IN MORE OR LESS SERIOUS CONSEQUENCES ARE CONSIDERED. (EWH)

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

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

147101
BRUYERE M
CHARACTERISTICS AND RESOLUTIONS OF THE SYSTEM OF EQUATIONS DERIVED FROM PARTIAL HYPERBOLICS (IN FRENCH)
DEPARTEMENT DES REACTEURS A EAU, FRANCE
DRE/STRE/LMTA 78/173 + FRRSR-173 +. 21 PPS. REFS. NOV. 22, 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'S 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 + FRRSR-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 B + JEANDEY CH
EXPERIMENTAL STUDY OF CRITICAL TWO-PHASE FLOW (IN ENGLISH)
COMMISSARIATE A L'ENERGIE ATOMIQUE, FRANCE
FRRSR-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 + RABASSE C + LEYER JC
FLAME PROPAGATION THROUGH UNCONFINED AND CONFINED HEMISPERICAL STRATIFIED GASEOUS MIXTURES (IN ENGLISH)
UNIVERSITE DE POITIERS, 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 VAPOUR CLOUD EXPLOSIONS. AVAL: NTS

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

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

146801
PORTE R + MAIGNE JP
STUDY OF POLLUTANT DISPERSION IN WATER AND AIR (IN ENGLISH)
CEA DEPARTMENT DE SURETE NUCLEAIRE, FRANCE
DSN 243(E) + 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

146869
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 MERKLE. 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

146870
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

146870 *CONT UED*
AXISYMETR; 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 DEFLEXION RECORDINGS, STRAIN MEASUREMENTS AND VISUAL OBSERVATIONS OF THE GEOMETRY. (FAH)

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

BUCKLING + PRESSURE VESSELS + PRESSURE, INTERNAL + TESTING + STEEL + STEEL, STAINLESS + ALUMINUM + FRANCE

143735
RIEGEL B
EXPERIENCE SUPER-CANON (IN FRENCH)
CEA SERVICE DES TRANSFERTS THERMIQUES, FRANCE
TT/SETRE/79-2-B/3R + FRNSR-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

148669
VOIN R
THE PRACTICE OF QUALITY ASSURANCE BY FRAMATOME (IN ENGLISH)
FRAMATOME, FRANCE
FRNSR-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 P
SELECTION OF EVENTS FOR A PROBABILISTIC EVALUATION OF PWR SAFETY (IN ENGLISH)
FRAMATOME, FRANCE
FRNSR-181 *, 8 PPS, FROM HAMBOURG 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 KOEBERG 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. 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. GERMANY
BF RS 50-32-C15-1 + 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. GERMANY
BF-RS 50-32-C13-1 + GERRSR-359 +. APPROX. 200 PPS, FIGS, JULY 1976

***THERE WAS NC 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-INSTITUTE E.V., FRANKFURT AM MAIN, F.R. 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 + JOST H
PROPAGATION OF SIMULATED SONIC EMISSION-IMPULSES IN THICK WALLED STRUCTURES (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. 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. 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. GERMANY
BF-RS 50-32-D15 + GERRSR-341 +. APPROX. 150 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. GERMANY
BF-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

148272
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. GERMANY
BF-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 AM MODEL CONTAINMENT (V EQUAL 600 M 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

146798
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. GERMANY
BF-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. GERMANY
BF-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
BF-RS-50-21-1 + GERRSR-328 +. 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 M 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. 22161

GERMANY + COMPUTER PROGRAM + ACCIDENT, LOSS OF COOLANT + CONTAINMENT + *CONTAINMENT ANALYSIS + EXPERIMENT + R AND D PROGRAM

144286
EXPERIMENTAL INVESTIGATION OF THE HYDROGEN DI 'RIBUTION 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
BF-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. 22161

DIFFUSION + ANALYTICAL MODEL + GERMANY + HYDROGEN

148273
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
BF-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. 22161

REACTOR + REACTOR, SAFETY RESEARCH + CONTAINMENT + COMPUTER PROGRAM + HEAT TRANSFER + CONTAINMENT STRUCTURE + CONTAINMENT ANALYSIS

145874
DRESCHER HP + RODDER 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 INGENIEURGESELLSCHAFT 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. 22161

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. 22161

HYDROGEN + AIR + *EXPLOSION + *SHOCK WAVE + ENVIRONMENT + GERMANY

143803

143303 *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 E/6/78 (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

146485
LOTTERMOSER J + WALHKIES 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 780236-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 78-738-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

146793
DOBMANN G
MAGENTIC 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

148370
WALTE 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

148368
BARBIAN OA + 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 GKSS (IN GERMAN)
GESELLSCHAFT FUR KERNENERGIEVERWERTUNG, F.R. GERMANY
REPORT 73 03 AR B 59 + GERRSR-354 +. 32 PPS, 9 TABS, 13 FIGS, 4 REFS, DEC. 13, 1978

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 FUR REAKTORSICHERHEIT (GRS) MBH, F.R. 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 BMFT 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-BENGSTON 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-P1 (VERSION 16.3) TO IBM STANDARD OPERATING SYSTEM MVS (WITH FORTRAN-H-EXTENDED COMPILER) (IN ENGLISH)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GRS-A-206 + GERRSR-335 +. 137 PPS, 20 FIGS, SEPT, 1978

DESCRIBES THE CONVERSION OF TRAC P1 VERSION 16.3 FOR USE ON GRS'S AMDAHL 470 V/5 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
WAHBA AB
REFLUX-GRS ANALYSIS OF THE REFLOOD EXPERIMENTS (RS 62) (IN ENGLISH)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. 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 REFLOODING + HEAT TRANSFER EXPERIMENT + R AND D PROGRAM + HYDRAULIC EXPERIMENT + EMERGENCY COOLING

143894

143894 *CONTINUED*
POINTNER W + RINGER F
BLOWDOWN - EXPERIMENT RS 109 (LOBI) CONTROL OF THE ELECTRICAL HEATING POWER (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GRS-A-205 + GERRSR-337 +. 23 PPS. 15 FIGS. SEPT. 1978

IN THE LOBI 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 BMFT-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 + HEATERS + *DNB + *NUCLEATE BOILING

143905
ULLRICH R
RELAP-4/GRS ANALYSIS OF THE NONNUCLEAR LOFT-TESTS L1-4 (PRE AND POST TEST CALCULATIONS) (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHERT (GRS) MB-H, 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 200% COLD LEG BREAK WITH ECC INJECTION INTO THE INTACT LOOP COLD LEG /3, 24/. THE RESULTS OF BOTH THE RELAP-4/GRS-PRETEST AND POSTTEST 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 POSTTEST 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 K
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 FUR 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
WARNEMUNDE R + MAY H
THE ESSENTIAL SAFETY ASPECTS OF A CONFINED NUCLEAR FUEL CYCLE (IN GERMAN)
GESELLSCHAFT FUR 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 FUR 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 FUR 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 FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GRS-F-70 + GERRSR-344 +. 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 OBJECTIVE OF THIS PROGRAM IS TO INVESTIGATE IN GREATER DETAIL THE SAFETY MARGINS OF NUCLEAR ENERGY PLANTS AND THEIR SYSTEMS AND THE FURTHER DEVELOPMENT OF SAFETY TECHNOLOGY. 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

145636
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 FUR 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 FUR 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 FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GRS-A-243 + GERRSR-382 +. 48 PPS. FIGS. REFS. DEC. 1978

148668 *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 LOFT 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 + LOFT (S-RR) + ACCIDENT, LOSS OF COOLANT + REACTOR, PWR + GERMANY + PRESSURE TRANSIENT + THERMAL TRANSIENT + FLOW THEORY AND EXPERIMENTS

148369
GRS ANNUAL PROGRESS REPORT-1978 (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT, 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

146913
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 FUR 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

146805
PITTS JH
ANALYSIS OF BOILING-WATER REACTOR STEAM CHUGGING (IN ENGLISH)
GESELLSCHAFT FUR 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 GENER? CHARACTERISTICS OF STEAM CHUGGING, COMPARED WELL WITH TWO LARGE SCALE EXPERIMENTS, GKM II TE T 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 FUR 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 FUR 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 QUABOX-COARSE-MESH METHOD, BUT ITERATION METHODS ARE DIFFERENT FROM QUABOX 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 THTR 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, 1978 (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (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, 1978 (IN ENGLISH)
GESELLSCHAFT FUR REAKTORSICHERHEIT (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
ATTITUDES ON QUESTIONS PERTAINING TO NUCLEAR ENERGY: PLUTONIUM (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (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 PROCESS.

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

144396 *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 * OXYGEN

143936
ERBACHER F
FUEL ELEMENT BEHAVIOR DURING A LOSS-OF-COOLANT ACCIDENT AND INTERACTION WITH THE EMERGENCY CORE COOLING (IN GERMAN)
KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. 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
BOCEK M * CLASS G * ERBACHER F * FIEGE 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. GERMANY
KFK-EXT.28/78-1 * GEFRSR-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

143389
BOCEK M
CREEP RUPTURE AT NON-STEADY STRESS AND TEMPERATURE LOADING CONDITIONS (IN ENGLISH)
KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. 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
KEDZIUR F * MOSINGER H
COMPARISON BETWEEN A ONE- AND TWO-DIMENSIONAL CALCULATION OF A WATER-VAPOR NOZZLE FLOW (IN GERMAN)
KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. 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 DUESE (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 SEMIANNUAL 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 1/78 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

146834
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 CYLDY2. THE RELEVANT CONSERVATION EQUATIONS ARE APPROXIMATED BY AN IMPLICITY 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, PLENA, 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-5756 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

148673 *CONTINUED*
ZIRCALOY IS NOT COMPATIBLE WITH OXIDE FUEL NOR WITH SOME FISSION PRODUCT ELEMENTS. THEREFORE, CHEMICAL INTERACTION BETWEEN THE IRRADIATED OXIDE FUEL AND THE ZRY CLADDING MATERIAL 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 ZRY CLADDING MATERIAL HAVE ON THE MECHANICAL PROPERTIES OF ZRY-4 TUBING OUT-OF-PILE BURST EXPERIMENTS AND CREEP RUPTURE TESTS HAVE BEEN PERFORMED AT TEMPERATURES > OR = 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
ALDRICK DC + BAYER A + SCHUNCKLER M
A PROPOSED WIND SHIFT MODEL FOR THE GERMAN REACTOR SAFETY STUDY (IN ENGLISH)
KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. GERMANY
KFK-2791 + GERRSR-390 +. 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 RADIONUCLIDE 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 RADIONUCLIDE 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
NOWOTNY B + DAUBLEBSKY P
IMPACT OF STEEL PROJECTILES ON REINFORCED CONCRETE, CALCULATION AND COMPARISON WITH EXPERIMENTAL TESTS (IN GERMAN)
KERNTECHNIK, ENTWICKLUNG, DYNAMIK, F.R. 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 INVESTIGATIN 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
SCHWEICKERT A
RESEARCH PROGRAM ON REACTOR SAFETY, 3D EXPERIMENT (IN GERMAN)
KRAFTWERK UNION, ERLANGEN, F.R. GERMANY
REPORT RE 23/001/78 + BMFT 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 REFLOOD 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 TESTMATRIX 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 REFLOODING

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. GERMANY
BMFT 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

143758 *CONTINUED*
PERSONNEL OF NUCLEAR POWER PLANTS AFTER AN EVALUATION OF MEDIA. TAKING INTO ACCOUNT THE PRODUCTION AND REPRODUCTION CRITERIA FOR THE HARD*ARE AND SOFT*ARE TOGETHER *ITH TECHNICAL AND ECONOMIC ASPECTS, A STANDARD METHOD IS RECOMMENDED.

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

GERMANY + *PO*ER PLANT, NUCLEAR + OPERATOR ACTION + *LICENSED OPERATOR + *TRAINING

143083
SCHUSTER E + FUCHS A + KARNATH G
ACTIVATED CORROSION PRODUCTS IN S*R LOOPS (IN GERMAN)
KRAFT*ERK UNION, ERLANGEN, F.R. GERMANY
BMFT-RS-20 + RE 23/057/77 + GERRSR-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 PO*ER STATIONS *ERE EVALUATED. COMPILATIONS APPLIED HAVE DEMONSTRATED THAT THERE ARE SUFFICIENT DATA FOR P*R'S ALLO*ING THEIR COMPARISON. THE AVAILABLE DATA FOR B*R'S ARE NOT SUFFICIENT FOR SUCH AN ANALOGUS EVALUATION. THE COMPARISON *AS DONE *ITH ACTIVITY CONCENTRATIONS OF 58CO AND 60CO IN THE COOLANT OF FOUR P*R'S OPERATING AT FULL LOAD. FURTHER ON ANALYTICAL METHODS FOR THE DETERMINATION OF THE ELEMENTAL SPECIFIC ACTIVITIES OF 60CO AND 55FE IN SAMPLES FROM THE F'OLANT AND FROM DIFFERENT COMPONENTS OF THE PRIMARY CIRCUITS HAVE BEEN OPTIMIZED. (FAH)

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

GERMANY + REACTOR, P*R + CORROSION + RADIONUCLIDE + COBALT + IRON + MAIN COOLING SYSTEM

144150
ENGEL H
DOSE REDUCTION (IN GERMAN)
KRAFT*ERK UNION, ERLANGEN, F.R. GERMANY
BMFT RS 204 + RE 23/010/78 + GERRSR-317 +. 15 PPS, 1 FIG, OCT. 1978

TO IMPROVE THE HYDROGEN/OXYGEN MEASUREMENTS *ITHIN THE GASEOUS *ASTE PROCESSING SYSTEM AT P*RS, INVESTIGATIONS *ERE PERFORMED TO DETERMINE *HAT PARAMETERS INFLUENCED THE MEASUREMENTS. SUCH PARAMETERS AS GAS HUMIDITY, PRESSURE, FLO*, INFLUENCE OF OXYGEN, HELIUM, AND ARGON CONCENTRATIONS WERE CONSIDERED. (GYM)

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

GERMANY + HYDROGEN + OXYGEN + *MEASUREMENT + *OFF GAS + *ASTE TREATMENT, GAS + *REACTOR, P*R

145156
KNODLER D
PRELIMINARY EMPIRICAL DESCRIPTION OF THE FUEL ROD BEHAVIOUR DURING LOCA (IN GERMAN)
KRAFT*ERK UNION, ERLANGEN, F.R. GERMANY
RE 23/005/78 + BMFG RS 177 + GERRSR-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 LOCA'S. THE BURST STRAIN AT *HICH 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, *HICH IN CONTRAST TO REACTOR CONDITIONS EXHIBIT VERY HOMOGENEOUS TEMPERATURES. THIS LEADS TO PARTICULARLY HIGH BURST STRAINS. IT IS SHO*N HO* THESE MODELS CAN BE APPLIED TO CASES *ITH AXIMUTHAL AND AXIAL TEMPERATURE VARIATIONS. (FAH)

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

GERMANY + ZIRCALOY + FUEL ROD + ACCIDENT, LOSS OF COOLANT + ANALYTICAL MODEL

145155
DORNER H
INVESTIGATION PROGRAM FOR THE TESTING OF A FRACTURE SAFETY DEVICE P. OTECTION SYSTEM FOR REACTOR COMPONENTS (IN GERMAN)
KRAFT*ERK UNION, ERLANGEN, F.R. GERMANY
RE 23/021/78 BMFT RS 104 GERRSR-325 +. 150 PPS, TABS, FIGS, OCT. 1978

RESULTS OF INVESTIG S ON THE MATERIAL BEHAVIOUR OF INSULATION-CONCRETE SUBJECTED TO T*O-PHASE JET LOADS ARE DESCR. . FURTHERMORE THIS REPORT DEALS *ITH THE RESULTS OF THE BURST TESTS *ITH PIPES WHICH *ERE CAR IED OUT UNDER P*R CONDITIONS. TEST EQUIPMENT, INSTRUMENTATION, THE MEASURING TECHNIQUES AND THE TEST PROCEDURE ARE DESCRIBED. VARIOUS VOLUME INCREASES AND LEAKAGE AREAS OCCUR DURING THE PIPE FAILURE, *HICH INFLUENCE TO A GREAT EXTENT THE THERMOHYDRAULIC 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. 22161

GERMANY + REACTOR + COMPONENTS + CONCRETE + PIPES AND PIPE FITTINGS + REACTOR, P*R

144675
HILDENBRAND G
KFA/KWU POWER RAMPS FUEL ROD IRRADIATION TESTS 1976/77 (GERMAN)
KRAFTWERK UNION, ERLANGEN, F.R. GERMANY
BMFT RS 203 * RE 23/023/78 * GERRSR-345 *. 35 PPS, 2 TABS, 13 FIGS, 8 REFS, OCT. 1978

IRRADIATION EXPERIMENTS IN HFR PETTEN WERE CARRIED OUT TO DETERMINE THE OPERATIONAL BEHAVIOR OF FUEL RODS IN LIGHT WATER REACTORS DURING POWER RAMPS. 36 PWR TEST FUEL RODS, WHICH HAD BEEN PRE-IRRADIATED IN A NUCLEAR POWER STATION UP TO BURNUPS OF ABOUT 25 GWD/T (U), HAVE BEEN RAMPED IN A RELOADABLE "PRESSURE BOILING CAPSULE". ON ALL FUEL RODS WITH HIGH RAMP TERMINAL POWERS, PEAKS OF FISSION PRODUCTS AT PELLET INTERFACES AND CRACKS, INSIGNIFICANT RIDGES, 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
SCHWEICKERT H
EMERGENCY COOLING DEPRESSURIZATION RESEARCH, BLOCKED COOLING CHANELS WITH BWR GEOMETRY (IN GERMAN)
KRAFTWERK UNION, ERLANGEN, F.R. GERMANY
REPORT RE 23/024/78 * GERRSR-346 *. 165 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 BALLOONING 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

144828
ENGEL H
DEVELOPMENT OF A SUCTION SYSTEM FOR INSTALLATIONS AND FITTINGS (IN GERMAN)
KRAFTWERK UNION, ERLANGEN, F.R. GERMANY
RE 23/027/78 * BMFT RS 239 * GERRSR-323 *. 123 PPS, 1 DIAGS, 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 23/028/73 * GRRSR-364 *. 163 PPS, TABS, 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 HO
RS236 - FINAL REPORT CONTROLLED-BLASTING DEMOLITION OF RADIOACTIVE PRIMARY LOOP COMPONENTS OF DECOMMISSIONED NUCLEAR POWER PLANTS (IN GERMAN)
MESSERSCHMITT-BOLKOW-BLOHM GMBH, F.R. GERMANY
SOR-629 * RS236 * GERRSR-383 *. 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*
DECOMMISSIONINGS, 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 ENSUING DECONTAMINATION AND TRANSPORT (IN GERMAN)
NIS, NUKLEAR-INGUIEUR-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 CHAMGER AND A FUEL ASSEMBLY CLADDING TUBE DAMAGE OF 10% CORRESPONDING TO THE LICENSING GUIDELINES CURRENTLY VALID FOR THE RELASE OF IODINE. THE NUCLIDE-SPECIFIC DISTRIBUTION OF THE RADIOACTIVITY IN REFERENCE CHAMBERS IN THE CONTAINMENT IS DETERMINED WITH THE "CORRAL" 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)
TECHNISCHEN UNIVERSITAT BERLIN, F.R. GERMANY
BMFT-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. (EWH)

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

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

143?84
MAYINGER F * VIECENZ HJ
PHASE SEPARATIO" (IN GERMAN)
TECHNISCHE UNIV. HANNOVER, F.R. GERMANY
BMFT-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
BMFT-FB RS 163-03 * GERRSR-334 *. APPROX. 150 PPS, FIGS (NO DATE)

THREE AREAS ARE DISCUSSED: ENTRAINMENT INVESTIGATION AND POST DRYO /, 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
BMFT 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

146979 *CONTINUED*

CONCENTRATED EFFORTS: OF THE STEEL HULL 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

143458

KORBER H * WEIHELMANN R * 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 OFMELSIN-1 AND BILANZ-1, PART I (IN GERMAN)

UNIVERSITAT STUTTGART, F.R. GERMANY

BMFT-RS 211(PART I) * GERRSR-363 *. 106 PPS, FIGS, REFS, JUNE 1978

MELSIM-1 IS COUPLED WITH THE EN GY BALANCE CODE BILANZ-1 IN ORDER TO OBTAIN THE INFLUENCE OF MELSIN-MODELS ON THE ATMOSPHERIC CONDITIONS IN THE CONTAINMENT. OVERALL RESULTS AS MASS AND 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 - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 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

165756

BISANZ R * 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)

UNIVERSITAT STUTTGART, F.R. GERMANY

BMFT-RS 211 (PART II) * 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

KOSFELD R * PUHR-WESTERHEIDE P

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

BMFT RS 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 MEC-ANICS RESEARCH ON MATERIALS USED IN FAST SODIUM-COOLED REACTORS (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
BF-R-62.945-3 + GERRSR-330 +. 94 PPS. 2 TABS. 41 FIGS. 15 REFS. 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 CODAC-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

144532
MOCHIZUKI Y + SOBAJIMA M + SUZUK 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 . TUAL 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
KOBAYASHI K + SATO K
ASCOT-II: 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, 425) (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 REFLOODING + 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 + JPNRSR-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 PARITAL DERIVATIVE OF H WITH RESPECT TO RHO 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 PELLET ON FUEL FAILURE BEHAVIOR (ENRICHMENT PARAMETER TEST IN NSRR) (IN ENGLISH & JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO
JAERI-M-7990 + JPNRSR-201 +. 54 PPS, FIGS, REFS, NOV. 1978

THE EFFECT OF HEAT GENERATION PROFILE IN PELLET 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 INCEASING ENRICHMENT OF THE FUEL ROD; THE FAILURE THRESHOLD ENERGY DEPOSITIONS FOR 5%, 10%, AND 20% ENRICHED FUEL RODS ARE ABOUT 278, 265 AND 248 CAL/GUO(SUB 2), RESPECTIVELY FUEL FAILURE DETERIORATION OF THE CLADDING.DISP 143986 HEDR

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 + JPNRSR-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 80 SEC. AFTER BREAK.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, [illegible] .161

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

143893
SUDO Y + MURAO Y
EXPERIMENT OF THE DOWNCOMER EFFECTIVE WATER HEAT DURING A REFLOOD PHASE OF PWR LOCA (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7978 + JPNRSR-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 REFLOOD 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.

143393 *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
SATO K + SASAKI S + ARAYA F
ALARM-P1: A COMPUTER PROGRAM FOR PRESSURIZED WATER REACTOR BLOWDOWN ANALYSIS (IN ENGLISH)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-8034 + JPNRSR-205 *. 103 PPS, 20 FIGS, 35 REFS, DEC. 1978

ALARM-P1 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 REFLOOD 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

148342
TAKEDA T + HIRANO K
FUEL COOLANT INTERACTION EXPERIMENT BY DIRECT ELECTRICAL HEATING METHOD (ZR(O)2- H2O SYSTEM) (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-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 PELLET 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

148944
IWAMURA T + KUROYANAGI T
FLOW REDUCTION TRANSIENT BURNOUT IN AN ANNULAR TEST SECTION (IN JAPANESE & ENGLISH)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-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 BURNOUT 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), BURNOUT 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 BURNOUT 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, BURNOUT + HEAT FLUX, CRITICAL + FLOW, ANNULAR + REACTOR, LWR + TRANSIENT + BOILING

5. JA NESE FAST REACTOR SAFETY RESEARCH REPORTS RECEIVED BY NRC

THE FOLLO NG IS A LISTING OF REPORTS RECEIVED FROM JAPAN DURING THE IRST HALF 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 WAVEGUIDE 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 ACCUSTIC SIGNALS WERE TRANSMITTED ON THE PIPE AS SURFCE WAVES (RAYLEICH WAVES) OR LAMB WAVES. (EWH)

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 DIMENSIONALESS 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 DJV
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

145640
BRITTAIN I + 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 + SGHW/HTWG/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. 22161

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

147795
HALSALL MJ
A REVIEW OF INTERNATIONAL SOLUTIONS TO NEACRP BENCHMARK BWR LATTICE CELL PROBLEMS
UKAEA ATOMIC ENERGY ESTABLISHMENT, DORSET, U.K.
AEEW-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 HOMOGENISED 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. 22616

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 + RIMMER 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. 22161

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

148848
DULLFORCE TA + RIMMER 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. 22161

*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 22161

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

143773
MACINNES 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 UO(2) 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 CRITTER
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, CRITTER, 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
MACINNES DA
THE ELECTRONIC CONTRIBUTION TO THE THERMODYNAMICS OF MOLTEN UO(2)
UKAEA SAFETY & RELIABILITY DIRECTORATE, WARRINGTON, UK
SRD R 130 + UKRSR-230 +. 6 PPS. 3 TABS. 14 REFS. SEPT. 1978

THE SPECIFIC HEAT OF MOLTEN UO(2) WAS ANALYZED AT ITS MELTING POINT T(SUB M) AND SUGGEST THERE EXISTS A MECHANISM WHICH CAN ABSORB INTERNAL ENERGY AND WHICH IS PRESENT IN UO(2) BUT NOT IN OTHER IONIC AB(2) COMPOUNDS SUCH AS CAF(2). 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 UO(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 F + 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 METEROROLOGICAL 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 UO(2)? 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 UO(2) 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.

143781
RAMSDEN D
ASSESSMENTS OF RISK FOLLOWING THE INHALATION OF PLUTONIUM OXIDE USING OBSERVED LUNG CLEARANCE PATTERNS
UKAEA ATOMIC ENERGY ESTABLISHMENT, DORSET, U.K.
AEEW-R-1119 + UKWSR-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 AEE WINFRITH. THESE RISKS ARE COMPARED WITH PUBLISHED DATA ON RISKS ARISING FROM A LUNG CLEARANCE BASED ON THE ICRP LUNG MODEL. (GTM)

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

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.

AUTHOR INDEX

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

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. (# INDICATES BEGINNING OF TITLE. * INDICATES END OF TITLE.)

INTERNAL CORROSION ON THE MECHANICAL #SIMULATION OF THE	CHEMICAL STATE OF IRRADIATED OXIDE FUEL: INFLUENCE OF THE	148673 P 20
#ANALYSIS OF BOILING-WATER REACTOR STEAM	CHUGGING (IN ENGLISH)*	146805 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	148273 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 LUNG	CLEARANCE PATTERNS* #ASSESSMENTS OF RISK FOLLOWING	145281 P 35
PRESSURE TRANSIENTS IN FULL-PRESSURE CONTAINMENTS OF WATER-#	CO. FLO* - A COMPUTER CODE FOR THE DETERMINATION OF	145636 P 16
RUNS 413 AND #ANALYSIS OF LOCA EXPERIMENTS WITH RELAP4/J	CODE (ANALYSIS OF ROSA-II EXPERIMENTS FOR COLD LEG BREAK	144532 P 27
#AN ANALYSIS OF LOFT L1-2 EXPERIMENT BY ALARM-P1 COMPUTER	CODE (IN ENGLISH)*	144526 P 27
EXPERIMENT L1-4: POST RUN CALCULATIONS USING THE COMPUTER	CODE "DRUFAN" (IN GERMAN)* #RESULTS OF THE LOFT	148668 P 16
OF VELOCITY OF GAS INSIDE A BUBLE RISING THROUGH A LIQUID	CODE CABU 1 (IN FRENCH)* #STUDY OF THE RANGE	143210 P 2
PWR FUEL ROD DURING A LOCA (IN FRENCH)* #CUPIDON: A	CODE DESCRIBING THE THERMAL AND MECHANICAL BEHAVIOR OF A	145871 P 1
#PATREC, A COMPUTER	CODE FOR FAULT TREE CALCULATIONS (IN ENGLISH)*	143778 P 4
HYDRAULIC STUDIES FOR THE DEVELOPMENT OF SAFETY ADVANCED	CODE FOR PWR (IN FRENCH)* #FRENCH THERMO-	143758 P 6
PRESSURE CONTAINMENTS OF WATER- #CO. FLO* - A COMPUTER	CODE FOR THE DETERMINATION OF PRESSURE TRANSIENTS IN FULL-	145636 P 16
SYSTEM MVS (WITH #REPORT ON THE CONVERSION OF THE LASL-	CODE TRAC-P1 (VERSION 16.3) TO IBM STANDARD OPERATING	143966 P 14
BLOWDOWN (#EXPERIMENTAL DETERMINATION OF THE HEAT TRANSFER	COEFFICIENT IN THE CONTAINMENT DURING A COOLING SYSTEM	148297 P 16
WITH RELAP4J CODE (ANALYSIS OF ROSA-II EXPERIMENTS FOR	COLD LEG BREAK RUNS 413 AND 312) (IN ENGLISH & JAPANESE)*	144532 P 27
INJECTION AND PUMP CIRCULATION ON LOCA PHENOMENA IN LARGEST	COLD LEG BREAKS (RUNS 332, 413, 425) (IN ENGLISH &	143891 P 27
#DEVELOPMENT AND SYNTHESIS OF AN EDUCATIONAL SYSTEM USING	COMBINATION OF MEDIA FOR THE INTENSIVE TRAINING AND	143768 P 21
ACCIDENT AND THE SUBSEQUENT THERMAL TRANSIENT IN LIGHT- #	COMPARATIVE INVESTIGATIONS OF A COOLING SYSTEM BLOWDOWN	145874 P 12
OF A WATER-VAPOR NOZZLE FLOW (IN GERMAN)* #	COMPARISON BETWEEN A ONE- AND TWO-DIMENSIONAL CALCULATION	143211 P 19
OF THE RATE OF VOIDING DURING A RAPID FAILURE (BLOWDOWN)	COMPARISON OF SEVERAL MODELS (IN FRENCH)* THE CALCULATION	143751 P 2
STEEL PROJECTILES ON REINFORCED CONCRETE, CALCULATION AND	COMPARISON WITH EXPERIMENTAL TESTS (IN GERMAN)* #IMPACT OF	145638 P 21
#INVESTIGATION OF THE PHENOMEN OCCURRING WITHIN A MULTI-	COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY	145846 P 10
#INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-	COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY	145847 P 11
#INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-	COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY	148273 P 12
REACTORS WITH #INVESTIGATION OF THE PROCESSES IN A MULTIPLE	COMPARTMENT CONTAINMENT BY PRESSURE IN WATER-COOLED	143329 P 10
IBM STANDARD OPERATING SYSTEM MVS (WITH FORTRAN-H-EXTENDED	COMPILER) (IN ENGLISH)#LASL-CODE TRAC-P1 (VERSION 16.3) TO	143966 P 14
DIFFERENT MASS #BEHAVIOR OF SPECIFIC REACTOR MATERIALS AND	COMPONENT PARTS AT IMPACT OF FRAGMENTS AND PROJECTILES OF	143803 P 13
EQUILIBRIUM CRITICAL DISCHARGE MODEL APPLIED TO MULTI-	COMPONENT, TWO-PHASE SYSTEMS - THE COMPUTER PROGRAMS CRITS	146483 P 33
#FAULT TREE ANALYSIS WITH MULTISTATE	COMPONENTS (IN ENGLISH)*	147860 P 20
OF A FRACTURE SAFETY DEVICE PROTECTION SYSTEM FOR REACTOR	COMPONENTS (IN GERMAN)* PROGRAM FOR THE TESTING	145155 P 22
IN FAULT TREE ANALYSIS (#UNCERTAINTY OF THE FAILURE RATE OF	COMPONENTS AND THE APPARENT INFLUENCE OF THESE OCCURRENCES	147779 P 24
CONTROLLED-BLASTING DEMOLITION OF RADIOACTIVE PRIMARY LOOP	COMPONENTS OF DECOMMISSIONED NUCLEAR POWER PLANTS (IN	147817 P 23
REACTORS, PHASE II, WATER INGRESS, AIR INGRESS, REACTIVITY #	COMPREHENSIVE DAMAGE ANALYSIS FOR HIGH TEMPERATURE GAS	148670 P 18
SERIES OF THE RESEARCH PROGRAM RS 50 (MODEL CONTAINMENT) #	COMPREHENSIVE SUMMARY OF THE THEORETICAL STUDIES ON THE D-	148272 P 11
FOR THE CALCULATION OF THE TWO-DIMENSIONAL FLOW OF A	COMPRESSIBLE MEDIUM IN COUPLED RECTANGULAR AREAS (IN	146750 P 20
INTERACTIONS DURING BLOWDOWN OF A PRESSURIZED #EFFICIENT	COMPUTATION OF THREE-DIMENSIONAL FLUID-STRUCTURE	146804 P 20
#AN ANALYSIS OF LOFT L1-2 EXPERIMENT BY ALARM-P1	COMPUTER CODE (IN ENGLISH)*	144526 P 27
THE LOFT EXPERIMENT L1-4: POST RUN CALCULATIONS USING THE	COMPUTER CODE "DRUFAN" (IN GERMAN)* #RESULTS OF	148668 P 16
#PATREC, A	COMPUTER CODE FOR FAULT TREE CALCULATIONS (IN ENGLISH)*	143778 P 4
IN FULL-PRESSURE CONTAINMENTS OF WATER-COOLED #CO. FLOW - A	COMPUTER CODE FOR THE DETERMINATION OF PRESSURE TRANSIENTS	145636 P 16
WATER POOL-SWELL IN THE CONDENSATION CHAMBER OF A REACTOR #	COMPUTER MODEL FOR THE TWO-DIMENSIONAL CALCULATION OF THE	145757 P 15
BEHAVIOR IN A PWR CORE DURING A LOCA (IN #ASCOT-1: A	COMPUTER PROGRAM FOR ANALYZING THE THERMO-HYDRAULIC	144529 P 27
ANALYSIS (IN ENGLISH)* #ALARM-P1: A	COMPUTER PROGRAM FOR PRESSURIZED WATER REACTOR BLOWDOWN	147502 P 29
#TIRION 4 -A	COMPUTER PROGRAM FOR USE IN NUCLEAR SAFETY STUDIES*	147602 P 33
OF WATER IN SIMPLE PIPE SYSTEMS* #HUBBLE BUBBLE II: A	COMPUTER PROGRAM TO DESCRIBE THERMAL NON-EQUILIBRIUM FLOW	140447 P 32
MODEL APPLIED TO MULTI-COMPONENT, TWO-PHASE SYSTEMS - THE	COMPUTER PROGRAMS CRITS AND CRITTER* CRITICAL DISCHARGE	146483 P 33
PRESSURE AND DENSITY AS THE INDEPENDENT VARIABLES #SPADE: A	COMPUTER SUBROUTINE FOR GENERATING STEAM TABLES HAVING	144527 P 28
FLOW OF A COMPRESSIBLE MEDIUM IN COUPLED #FLUST-2D - A	COMPUTERCODE FOR THE CALCULATION OF THE TWO-DIMENSIONAL	146750 P 20
#INVESTIGATION AND DEVELOPMENT OF SYSTEMS LIMITING THE H2-	CONCENTRATION IN THE BWR CONTAINMENT (IN GERMAN)*	147473 P 23
#REPORT OF THE FEDERAL MINISTER FOR RESEARCH AND TECHNOLOGY	CONCERNING RESEARCH PROJECTS IN THE AREA OF REACTOR SAFETY	147483 P 18
ACCIDENT: CALCULATION TO #THE COURSE OF EVENTS IN THE	CONCRETE - FAILURE PHASE OF THE HYPOTHETICAL CORE MELTDOWN	143379 P 15
TESTS (IN GERMAN)#IMPACT OF STEEL PROJECTILES ON REINFORCED	CONCRETE, CALCULATION AND COMPARISON WITH EXPERIMENTAL	145638 P 21
TWO-DIMENSIONAL CALCULATION OF THE WATER POOL-SWELL IN THE	CONDENSATION CHAMBER OF A REACTOR SYSTEM (IN GERMAN)* THE	145757 P 15
OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS.	CONDENSATION IN CONTAINMENT BY EXPERIMENTS C04 AND C1 TO	148273 P 12
CONDITIONS, ON A STAINLESS STEEL TEST SECTION #STUDY ON THE	CONDENSATION OF AIR AND STEAM MIXTURES, IN TRANSIENT	149000 P 7
ELECTRONIC TRANSITIONS CONTRIBUTE TO THE THERMODYNAMICS OF	CONDENSED UO(2)? A REVIEW OF THE ARGUMENTS* #DO	148846 P 34
BY PRESSURE IN WATER-COOLED REACTORS WITH REFRIGERATED	CONDENSER (IN GERMAN)* A MULTIPLE COMPARTMENT CONTAINMENT	143329 P 10
#CREEP RUPTURE AT NON-STEADY STRESS AND TEMPERATURE LOADING	CONDITIONS (IN ENGLISH)*	143089 P 19
INVESTIGATIONS ON THE LWR FUEL ROD BEHAVIOR UNDER ACCIDENT	CONDITIONS (IN GERMAN)#OF THE THEORETICAL AND EXPERIMENTAL	143745 P 19
LOCA FOR PURPOSES OF STUDYING THE ENSUING #SPECIFICATION OF	CONDITIONS OF A NUCLEAR POWER PLANT WITH A PWR FOLLOWING A	146795 P 24
ON THE CONDENSATION OF AIR AND STEAM MIXTURES, IN TRANSIENT	CONDITIONS, ON A STAINLESS STEEL TEST SECTION (IN FRENCH)*	149000 P 7
ENGLISH)* #FLAME PROPAGATION THROUGH UNCONFINED AND	CONFINED HEMISPERICAL STRATIFIED GASEOUS MIXTURES (IN	143078 P 8
#THE ESSENTIAL SAFETY ASPECTS OF A	CONFINED NUCLEAR FUEL CYCLE (IN GERMAN)*	147102 P 15
OF SYSTEMS LIMITING THE H2-CONCENTRATION IN THE BWR	CONTAINMENT (IN GERMAN)* #INVESTIGATION AND DEVELOPMENT	147473 P 23
OF THE PHENOMEN OCCURRING WITHIN A MULTI-COMPARTMENT	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)*	148273 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)*	148297 P 16
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 MULTICOMPARTMENTED	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	145636 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* #DO ELECTRONIC TRANSITIONS	CONTRIBUTE TO THE THERMODYNAMICS OF CONDENSED UO(2)? A	148846 P 34
#THE ELECTRONIC	CONTRIBUTION TO THE THERMODYNAMICS OF MOLTEN UO(2)*	14884[illegible]

#THE ELECTRONIC	CONTRIBUTION TO THE THERMODYNAMICS OF UO2.*	143773 P 22
#BLOWDOWN - EXPERIMENT RS 109 (LOOP)	CONTROL OF THE ELECTRICAL HEATING POWER (IN GERMAN)*	142894 P 15
COMPONENTS OF DECOMMISSIONED NUCLEA- #RS236 - FINAL REPORT	CONTROLLED-BLASTING DEMOLITION OF RADIOACTIVE PRIMARY LOOP	147817 P 23
STANDARD OPERATING SYSTEM MVS (WITH FORTRAN- #REPORT ON THE	CONVERSION OF THE CASE-CODE TRAC-PI (VERSION 16.3) TO IBM	143966 P 18
COOLING (IN GERMAN)*#FUEL ELEMENT BEHAVIOR DURING A LOSS-OF-	COOLANT ACCIDENT AND INTERACTION WITH THE EMERGENCY CORE	143906 P 19
IN A LIGHT WATER REACTOR CONTAINMENT FOLLOWING A LOSS-OF-	COOLANT ACCIDENT, QUICK LOOK REPORT 1 (IN GERMAN)*	144582 P 11
IN A LIGHT WATER REACTOR CONTAINMENT FOLLOWING A LOSS-OF-	COOLANT ACCIDENT, QUICK LOOK REPORT 2 (IN GERMAN)*	144286 P 12
TRANSIENT IN A MULTICOMPARTMENTED CONTAINMENT FROM THE	COOLANT BLOWDOWN OF A WATER-COOLED REACTOR - INTERIM	145875 P 16
HEATING METHOD (ZRO2- H2O SYSTEM) (IN JAPANESE)* #FUEL	COOLANT INTERACTION EXPERIMENT BY DIRECT ELECTRICAL	148842 P 29
#LOCAL FLOW BLOCKAGE EXPERIMENTS IN 37-PIN SODIUM	COOLED BUNDLES WITH GRID SPACERS (IN ENGLISH)*	143907 P 16
PRESSURE TRANSIENTS IN FULL-PRESSURE CONTAINMENTS OF WATER-	COOLED NUCLEAR POWER PLANTS (IN GERMAN)* DETERMINATION OF	145636 P 16
CONTAINMENT FROM THE COOLANT BLOWDOWN OF A WATER-	COOLED REACTOR - INTERIM RESEARCH REPORT C 13 (IN GERMAN)*	145875 P 16
MECHANICS RESEARCH ON MATERIALS USED IN FAST SODIUM-	COOLED REACTORS (IN GERMAN)* #MEASUREMENTS IN FRACTURE	14[illegible]25 P 26
IN A MULTIPLE COMPARTMENT CONTAINMENT BY PRESSURE IN WATER-	COOLED REACTORS WITH REFRIGERATED CONDENSER (IN GERMAN)*	143329 P 16
AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-	COOLED REACTORS, CONDENSATION IN CONTAINMENT BY	148273 P 12
AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-	COOLED REACTORS, QUICK LOOK REPORT EXPERIMENT D15 (IN	145846 P 16
AFTER RUPTURE OF THE PRIMARY COOLING CIRCUITS IN WATER-	COOLED REACTORS, SUPPLEMENTAL RESEARCH DOCUMENTATION D15 (	145847 P 11
OF-COOLANT ACCIDENT AND INTERACTION WITH THE EMERGENCY CORE	COOLING (IN GERMAN)* #FUEL ELEMENT BEHAVIOR DURING A LOSS-	143906 P 19
#EMERGENCY COOLING DEPRESSURIZATION RESEARCH, BLOCKED	COOLING CHANELS WITH BWR GEOMETRY (IN GERMAN)*	148296 P 23
MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY	COOLING CIRCUIT IN WATER-COOLED REACTORS, CONDENSATION IN	148273 P 12
MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY	COOLING CIRCUIT IN WATER-COOLED REACTORS, QUICK LOOK	145846 P 16
MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY	COOLING CIRCUITS IN WATER-COOLED REACTORS, SUPPLEMENTAL	145847 P 11
WITH BWR GEOMETRY (IN GERMAN)* #EMERGENCY	COOLING DEPRESSURIZATION RESEARCH, BLOCKED COOLING CHANELS	148296 P 23
THE HEAT TRANSFER COEFFICIENT IN THE CONTAINMENT DURING A	COOLING SYSTEM BLOWDOWN (IN GERMAN)* DETERMINATION OF	148297 P 16
THERMAL TRANSIENT IN LIGHT-#COMPARATIVE INVESTIGATIONS OF A	COOLING SYSTEM BLOWDOWN ACCIDENT AND THE SUBSEQUENT	145874 P 12
MODELISATION DURING THE REFLOODING PHASE OF A PWR'S	CORE (IN FRENCH)* #FLOW AND HEAT TRANSFER THERMOHYDRAULIC	143926 P 3
LOSS-OF-COOLANT ACCIDENT AND INTERACTION WITH THE EMERGENCY	CORE COOLING (IN GERMAN)* #FUEL ELEMENT BEHAVIOR DURING A	143906 P 19
FOR ANALYZING THE THERMO-HYDRAULIC BEHAVIOR IN A PWR	CORE DURING A LOCA (IN ENGLISH)* II: A COMPUTER PROGRAM	144525 P 27
SUPPORT #INVESTIGATION OF THE VARIOUS PHASES OF THE	CORE MELT ACCIDENT AFTER THE AFTER FAILURE OF THE CORE	145756 P 25
STANDARD PLANT, #INVESTIGATION OF THE FIRST PHASE OF THE	CORE MELT ACCIDENT WITH MELSIM-1 IN A PWR AND A BWR	145458 P 25
EVENTS IN THE CONCRETE - FAILURE PHASE OF THE HYPOTHETICAL	CORE MELTDOWN ACCIDENT: CALCULATION TO IDENTIFY THE	143379 P 15
OF THE CORE MELT ACCIDENT AFTER THE AFTER FAILURE OF THE	CORE SUPPORT STRUCTURE DUE TO THE FORMATION OF MELT OR DUE	145756 P 25
STATE OF IRRADIATED OXIDE FUEL: INFLUENCE OF THE INTERNAL	CORROSION ON THE MECHANICAL PROPERTIES OF ZRY-4 TUBING (	148673 P 26
#ACTIVATED	CORROSION PRODUCTS IN LWR LOOPS (IN GERMAN)*	143083 P 24
OF THE TWO-DIMENSIONAL FLOW OF A COMPRESSIBLE MEDIUM IN	COUPLED RECTANGULAR AREAS (IN GERMAN)* FOR THE CALCULATION	146750 P 24
WITH MELSIM-1 IN A PWR AND A BWR STANDARD PLANT, AND THE	COUPLING OF MELSIM-1 AND BILAN2-1, PART I (IN GERMAN)*	145458 P 25
HYPOTHETICAL CORE MELTDOWN ACCIDENT: CALCULATION TO #THE	COURSE OF EVENTS IN THE CONCRETE - FAILURE PHASE OF THE	143379 P 15
EVENTS: EARTHQUAKES, FIRES, EXPLOSIONS AND AIRCRAFT	CRASHES (IN ENGLISH)* POWER PLANTS (NPPS) AGAINST EXTERNAL	143128 P 5
CONDITIONS (IN ENGLISH)*	# CREEP RUPTURE AT NON-STEADY STRESS AND TEMPERATURE LOADING	143089 P 19
PHASE SYSTEMS - THE #A SIMPLE HOMOGENEOUS EQUILIBRIUM	CRITICAL DISCHARGE MODEL APPLIED TO MULTI-COMPONENT, TWO-	146483 P 23
FLOW (IN FRENCH)*	# CRITICAL FLOW AND FLOW BLOCKAGE PHENOMENON FOR A TWO PHASE	143911 P 1
#EXPERIMENTAL STUDY OF	CRITICAL TWO-PHASE FLOW (IN ENGLISH)*	143900 P 7
MULTI-COMPONENT, TWO-PHASE SYSTEMS - THE COMPUTER PROGRAMS	CRITS AND CRITTER* CRITICAL DISCHARGE MODEL APPLIED TO	146483 P 23
TWO-PHASE SYSTEMS - THE COMPUTER PROGRAMS CRITS AND	CRITTER* DISCHARGE MODEL APPLIED TO MULTI-COMPONENT,	146483 P 23
BEHAVIOR OF A PWR FUEL ROD DURING A LOCA (IN FRENCH)*	# CUPIDON: A CODE DESCRIBING THE THERMAL AND MECHANICAL	145871 P 1
#ERROR ANALYSIS OF THE AMPLITUDE	CURVE, FINAL REPORT (IN GERMAN)*	148376 P 13
#THE ESSENTIAL SAFETY ASPECTS OF A CONFINED NUCLEAR FUEL	CYCLE (IN GERMAN)*	147102 P 15
ANALYSES OF A PROGRESSIVE DISTORSION PROBLEM (TENSION AND	CYCLE TORSION) (IN FRENCH)* #SPURT	146671 P 6
RELEASE FROM A DEFECTED FUEL ROD - EFFECT OF THERMAL	CYCLING (IN ENGLISH)* #ANALYSIS OF THE FISSION PRODUCT	144595 P 5
COOLED REACTORS, CONDENSATION IN CONTAINMENT BY EXPERIMENTS	C04 AND C1 TO C16 (IN GERMAN)* COOLING CIRCUIT IN WATER-	148273 P 12
CONDENSATION IN CONTAINMENT BY EXPERIMENTS C04 AND	C1 TO C16 (IN GERMAN)* CIRCUIT IN WATER-COOLED REACTORS,	148273 P 12
CONDENSATION IN CONTAINMENT BY EXPERIMENTS C04 AND C1 TO	C16 (IN GERMAN)* COOLING CIRCUIT IN WATER-COOLED REACTORS,	148273 P 12

#THE CONTAINMENT TEST FACILITY (EXPERIMENTS C AND	D) (IN GERMAN)*	143806 P 12
#COMPREHENSIVE SUMMARY OF THE THEORETICAL STUDIES ON THE	D-SERIES OF THE RESEARCH PROGRAM RS 50 (MODEL CONTAINMENT)	148274 P 11
II, WATER INGRESS, AIR INGRESS, REACTIVITY #COMPREHENSIVE	DAMAGE ANALYSIS FOR HIGH TEMPERATURE GAS REACTORS, PHASE	148673 P 18
#INSTRUMENTATION SYSTEM FOR THE	DAS-MULTIPLE TUBE RESEARCH PROGRAM (IN GERMAN)*	146796 P 13
TUBULAR TEST SECTION ON OMEGA LOOP (IN FRENCH)*	# DATA REDUCTION OF THE FIRST TEST SERIES OF BLOWDOWN ON A	143755 P 3
CIRCULATION ON LOCA PHENOMENA IN LARGEST COLD #ROSA-II TEST	DATA REPORT 12 EFFECT OF ECCS INJECTION AND PUMP	143891 P 27
AND ERROR RECONSTRUCTION WITH HELP OF TIME-OF-FLIGHT	DATA, FINAL REPORT (IN GERMAN)* #DISTURBANCE	148368 P 13
DEMOLITION OF RADIOACTIVE PRIMARY LOOP COMPONENTS OF	DECOMMISSIONED NUCLEAR POWER PLANTS (IN GERMAN)* BLASTING	147817 P 23
A PWR FOLLOWING A LOCA FOR PURPOSES OF STUDYING THE ENSUING	DECONTAMINATION AND TRANSPORT (IN GERMAN)*POWER PLANT WITH	146795 P 24
#LASER HOLOGRAPHIC AND SPECKLE PHOTOGRAPHY METHODS FOR	DEFECT DETECTION AND STRAIN EVALUATION IN PRESSURE VESSELS*	120343 P 31
#ANALYSIS OF THE FISSION PRODUCT RELEASE FROM A	DEFECTED FUEL ROD - EFFECT OF THERMAL CYCLING (IN ENGLISH)*	144595 P 5
#CALCULATING PLASTIC	DEFORMATION OF STRUCTURES (IN FRENCH)*	143875 P 3
DECOMMISSIONED #RS236 - FINAL REPORT CONTROLLED-BLASTING	DEMOLITION OF RADIOACTIVE PRIMARY LOOP COMPONENTS OF	147817 P 23
SUBROUTINE FOR GENERATING STEAM TABLES HAVING PRESSURE AND	DENSITY AS THE INDEPENDENT VARIABLES (IN ENGLISH)*	144527 P 26
RESONANCE (IN GERMAN)* #DEVELOPMENT OF A MASS-	DENSITY METHOD FOR TRANSIENT TWO PHASE STATE USING ATOMIC	144197 P 25
BWR GEOMETRY (IN GERMAN)* #EMERGENCY COOLING	DEPRESSURIZATION RESEARCH, BLOCKED COOLING CHANELS WITH	148296 P 23
#CHARACTERISTICS AND RESOLUTIONS OF THE SYSTEM OF EQUATIONS	DERIVED FROM PARTIAL HYPERBOLICS (IN FRENCH)*	147101 P 7
#MODEL OF THE FAILURE RATES OF THE VALVES OF ST. LAURENT	DES EAUX POWER PLANT ACCORDING TO INFLUENTIAL PARAMETERS*	143339 P 6
PIPE SYSTEMS* #BUBBLE BUBBLE II: A COMPUTER PROGRAM TO	DESCRIBE THERMAL NON-EQUILIBRIUM FLOW OF WATER IN SIMPLE	140447 P 22
FUEL ROD DURING A LOCA (IN FRENCH)* #CUPIDON: A CODE	DESCRIBING THE THERMAL AND MECHANICAL BEHAVIOR OF A PWR	145871 P 1
FRENCH)* #STUDY OF THE STABILITY OF VARIOUS SYSTEMS AND	DESCRIPTION OF EQUATIONS FOR HYDRODYNAMIC ANALYSIS (IN	143756 P 2
GERMAN)* #PRELIMINARY EMPIRICAL	DESCRIPTION OF THE FUEL ROD BEHAVIOUR DURING LOCA (IN	145150 P 22
HOLOGRAPHIC AND SPECKLE PHOTOGRAPHY METHODS FOR DEFECT	DETECTION AND STRAIN EVALUATION IN PRESSURE VESSELS*#LASER	120343 P 31
CONTAINMENTS OF WATER- #CO2 FLOW - A COMPUTER CODE FOR THE	DETERMINATION OF PRESSURE TRANSIENTS IN FULL-PRESSURE	145636 P 16
CONTAINMENT DURING A COOLING SYSTEM BLOWDOWN (#EXPERIMENTAL	DETERMINATION OF THE HEAT TRANSFER COEFFICIENT IN THE	148297 P 16
SHOCK WAVES IN THE ENVIRONMENT (IN GERMAN)* #INITIATION OF	DETONATION OF HYDROGEN-AIR MIXTURES AND PROPAGATION OF	144280 P 12
#EXPERIMENTAL STUDY OF THE OVERPRESSURE GENERATED BY THE	DETONATION OF SPHERICAL AIR-HYDROGEN GASEOUS MIXTURES (IN	144415 P 6
#INVESTIGATION PROGRAM FOR THE TESTING OF A FRACTURE SAFETY	DEVICE PROTECTION SYSTEM FOR REACTOR COMPONENTS (IN GERMAN)	145155 P 22
COMPONENT PARTS AT IMPACT OF FRAGMENTS AND PROJECTILES OF	DIFFERENT MASS AND VELOCITY (IN GERMAN)* MATERIALS AND	143803 P 13
GERMAN)* #COMPARISON BETWEEN A ONE- AND TWO-	DIMENSIONAL CALCULATION OF A WATER-VAPOR NOZZLE FLOW (IN	143211 P 19
CONDENSATION CHAMBER OF A #COMPUTER MODEL FOR THE TWO-	DIMENSIONAL CALCULATION OF THE WATER POOL-SWELL IN THE	145757 P 10

IN THE SAFETY EVALUATION OF NUCLEAR INSTALLATIONS (IN	ENGLISH)* #USE OF PROBABILISTIC METHODS	147021 P 5
IN 37-PIN SODIUM COOLED BUNDLES WITH GRID SPACERS (IN	ENGLISH)* #LOCAL FLOW BLOCKAGE EXPERIMENTS	143907 P 36
AND CONFINED HEMISPERICAL STRATIFIED GASEOUS MIXTURES (IN	ENGLISH)* #FLAME PROPAGATION THROUGH UNCONFINED	143078 P 6
FROM A DEFECTED FUEL ROD - EFFECT OF THERMAL CYCLING (IN	ENGLISH)* #ANALYSIS OF THE FISSION PRODUCT RELEASE	144595 P 5
DETONATION OF SPHERICAL AIR-HYDROGEN GASEOUS MIXTURES (IN	ENGLISH)* STUDY OF THE OVERPRESSURE GENERATED BY THE	144415 P 6
EARTHQUAKES, FIRES, EXPLOSIONS AND AIRCRAFT CRASHES (IN	ENGLISH)* POWER PLANTS (NPPS) AGAINST EXTERNAL EVENTS:	143138 P 5
THERMO-HYDRAULIC BEHAVIOR IN A PWR CORE DURING A LOCA (IN	ENGLISH)* #ASCOT-1: A COMPUTER PROGRAM FOR ANALYZING THE	144529 P 27
AND TECHNOLOGY OF THE FEDERAL REPUBLIC OF GERMANY, 1978 (IN	ENGLISH)* PROJECTS SPONSORED BY THE MINISTRY FOR RESEARCH	147467 P 18
THE PROTECTION SYSTEM OF THE FESSENHEIM I PWR REACTOR (IN	ENGLISH)* EVENT PROBLEM BY THE STUDY OF THE RELIABILITY OF	143779 P 7
CORROSION ON THE MECHANICAL PROPERTIES OF ZRY-4 TUBING (IN	ENGLISH)* IRRADIATED OXIDE FUEL: INFLUENCE OF THE INTERNAL	148673 P 26
OPERATING SYSTEM MVS (WITH FORTRAN-H-EXTENDED COMPILER) (IN	ENGLISH)* CASE-CODE TRAC-P1 (VERSION 16.3) TO IBM STANDARD	143966 P 14
EPRI AND JSTA, JULY 1-SEPTEMBER 30, 1978 (IN GERMAN &	ENGLISH)* REACTOR SAFETY RESEARCH PROGRAMS OF BMFT, USNRC,	144854 P 16
REPORT PERIOD OCTOBER 1 - DECEMBER 31, 1978 (IN GERMAN &	ENGLISH)*RESEARCH PROGRAMS OF BMFT, USNRC, EPRI, AND JSTA,	146813 P 17
HEAT GENERATION PROFILE IN PELLET ON FUEL FAILURE BEHAVIOR (	ENRICHMENT PARAMETER TEST IN NSRR) (IN ENGLISH & JAPANESE)*	143974 P 28
WITH A PWR FOLLOWING A LOCA FOR PURPOSES OF STUDYING THE	ENSUING DECONTAMINATION AND TRANSPORT (IN GERMAN)* PLANT	146795 P 24
HYDROGEN-AIR MIXTURES AND PROPAGATION OF SHOCK WAVES IN THE	ENVIRONMENT (IN GERMAN)* #INITIATION OF DETONATION OF	144280 P 12
FROM THE REACTOR SAFETY RESEARCH PROGRAMS OF BMFT, USNRC,	EPRI AND JSTA, JULY 1-SEPTEMBER 30, 1978 (IN GERMAN &	144854 P 16
FROM THE REACTOR SAFETY RESEARCH PROGRAMS OF BMFT, USNRC,	EPRI, AND JSTA, REPORT PERIOD OCTOBER 1 - DECEMBER 31,	146813 P 17
A CALCULATIONAL PROGRAM FOR THE SOLUTION OF THE NEUTRONIC	EQUATION OF A MULTIDIMENSIONAL HTGR MODEL (IN GERMAN)* OF	146672 P 17
#CHARACTERISTICS AND RESOLUTIONS OF THE SYSTEM OF	EQUATIONS DERIVED FROM PARTIAL HYPERBOLICS (IN FRENCH)*	147101 P 7
OF THE STABILITY OF VARIOUS SYSTEMS AND DESCRIPTION OF	EQUATIONS FOR HYDRODYNAMIC ANALYSIS (IN FRENCH)* #STUDY	143752 P 2
COMPONENT, #TWO-PHASE SYSTEMS - THE #A SIMPLE HOMOGENEOUS	EQUILIBRIUM CRITICAL DISCHARGE MODEL APPLIED TO MULTI-	146483 P 31
BUBBLE III: A COMPUTER PROGRAM TO DESCRIBE THERMAL NON-	EQUILIBRIUM FLOW OF WATER IN SIMPLE PIPE SYSTEMS* #BUBBLE	145447 P 32
GERMAN)*	# ERROR ANALYSIS OF THE AMPLITUDE CURVE, FINAL REPORT (IN	148370 P 13
FINAL REPORT (IN GERMAN)* #DISTURBANCE AND	ERROR RECONSTRUCTION WITH HELP OF TIME-OF-FLIGHT DATA:	148368 P 13
IN GERMAN)* #THE	ESSENTIAL SAFETY ASPECTS OF A CONFINED NUCLEAR FUEL CYCLE (	147102 P 15
#LNG/WATER VAPOUR EXPLOSIONS -	ESTIMATES OF PRESSURES AND YIELDS*	148843 P 23
FOR THE ATTAINMENT OF INTERPRETATIONAL MODELS FOR THE	ESTIMATION OF FLOW USING ULTRASONIC TESTS ON NUCLEAR	148486 P 13
SPECKLE PHOTOGRAPHY METHODS FOR DEFECT DETECTION AND STRAIN	EVALUATION IN PRESSURE VESSELS* #LASER HOLOGRAPHIC AND	120343 P 31
#FRACTURE PROBABILITY	EVALUATION OF A LWR PRESSURE VESSEL (IN ENGLISH)*	143870 P 5
#USE OF PROBABILISTIC METHODS IN THE SAFETY	EVALUATION OF NUCLEAR INSTALLATIONS (IN ENGLISH)*	147021 P 5
#SELECTION OF EVENTS FOR A PROBABILISTIC	EVALUATION OF PWR SAFETY (IN ENGLISH)*	147175 P 9
CALCULATIONAL PROGRAM (IN GERMAN)*	# EVALUATION OF THE 25-ROD BUNDLE TEST (RS-37C) WITH THE	143902 P 14
USING SONIC EMISSION ANALYSIS: EXPANDED INSTRUMENTATION AND	EVALUATION PROGRAM (IN GERMAN)* ON LEAKAGE MONITORING	145872 P 10
PROTECTION SYSTEM OF THE #A FIRST APPROACH OF THE RARE	EVENT PROBLEM BY THE STUDY OF THE RELIABILITY OF THE	143779 P 7
ENGLISH)* #SELECTION OF	EVENTS FOR A PROBABILISTIC EVALUATION OF PWR SAFETY (IN	147175 P 9
CORE MELTDOWN ACCIDENT: CALCULATION TO #THE COURSE OF	EVENTS IN THE CONCRETE - FAILURE PHASE OF THE HYPOTHETICAL	143379 P 15
#PROTECTION OF NUCLEAR POWER PLANTS (NPPS) AGAINST EXTERNAL	EVENTS: EARTHQUAKES, FIRES, EXPLOSIONS AND AIRCRAFT	143138 P 5
	# EXAMINATION OF 3 HSST PLATES RUPTURED IN AIR (IN GERMAN)*	146799 P 13
REACTORS, PHASE II, WATER INGRESS, AIR INGRESS, REACTIVITY	EXCURSIONS (IN GERMAN)* ANALYSIS FOR HIGH TEMPERATURE GAS	146670 P 18
ON LEAKAGE MONITORING USING SONIC EMISSION ANALYSIS:	EXPANDED INSTRUMENTATION AND EVALUATION PROGRAM (IN GERMAN)	145872 P 10
	# EXPERIENCE SUPER-CANON (IN FRENCH)*	1487.3 P 5
#RESEARCH PROGRAM ON REACTOR SAFETY, 3D	EXPERIMENT (IN GERMAN)*	14862? P 41
PRESSURE TRANSIENT OF A NUCLEAR POWER PLANT IN THE LARGE	EXPERIMENT AREA OF THE GKSS (IN GERMAN)*TUBE ARRAYS IN THE	1464?? P 14
#AN ANALYSIS OF LOFT L1-2	EXPERIMENT BY ALARM-P1 COMPUTER CODE (IN ENGLISH)*	14? 26 P 27
H2O SYSTEM) (IN JAPANESE)* #FUEL COOLANT INTERACTION	EXPERIMENT BY DIRECT ELECTRICAL HEATING METHOD (ZRO2-	?6842 P 29
IN GERMAN AND ENGLISH)* #POST-	EXPERIMENT CALCULATION OF THE NON-NUCLEAR LOFT TEST L1-5 (	148?74 P 16
COOLING CIRCUIT IN WATER-COOLED REACTORS, QUICK LOOK REPORT	EXPERIMENT D15 (IN GERMAN)* AFTER RUPTURE OF THE PRIMARY	14??46 P 16
#ACOUSTIC NOISES WITH LOSS-OF-FLOW SODIUM BOILING	EXPERIMENT IN A 19-PIN BUNDLE (IN ENGLISH)*	143776 P 36
CODE "DRUFAN" (IN GERMAN)* #RESULTS OF THE LOFT	EXPERIMENT L1-4: PEST RUN CALCULATIONS USING THE COMPUTER	148668 P 16
REFLOOD PHASE OF PWR LOCA (IN JAPANESE)*	# EXPERIMENT OF THE DOWNCOMER EFFECTIVE WATER HEAT DURING (	143893 P 28
ANALYSIS: EXPANDED INSTRUMENTATION AND #PRELIMINARY	EXPERIMENT ON LEAKAGE MONITORING USING SONIC EMISSION	145872 P 10
POWER (IN GERMAN)* #BLOWDOWN -	EXPERIMENT RS 109 (LOBI) CONTROL OF THE ELECTRICAL HEATING	143894 P 15
#PREDICTION OF ROSA-III	EXPERIMENT RUN 702 (IN JAPANESE)*	147064 P 28
HYDRAULIC BEHAVIOR IN THE INITIAL BLOWDOWN PHASE, PARTS A,	# EXPERIMENTAL AND THEORETICAL RESEARCH ON THE THERMAL	143901 P 24
SECONDARY SIDE OF A STEAM GENERATOR (IN FRENCH)* #ANA	EXPERIMENTAL AND THEORETICAL STUDY OF THE BLOWDOWN OF THE	148734 P 5
COEFFICIENT IN THE CONTAINMENT DURING A COOLING SYSTEM	# EXPERIMENTAL DETERMINATION OF THE HEAT TRANSFER	148297 P 10
A LIGHT WATER REACTOR CONTAINMENT FOLLOWING A LOSS-OF-	# EXPERIMENTAL INVESTIGATION OF THE HYDROGEN DISTRIBUTION IN	144582 P 11
A LIGHT WATER REACTOR CONTAINMENT FOLLOWING A LOSS-OF-	# EXPERIMENTAL INVESTIGATION OF THE HYDROGEN DISTRIBUTION IN	144286 P 12
PRIMARY SYSTEM OF PRESSURIZED WATER REACTORS (IN GERMAN)*	# EXPERIMENTAL INVESTIGATIONS OF THE RADIOACTIVITY IN THE	144556 P 18
UNDER ACCIDENT #STATUS AND RESULTS OF THE THEORETICAL AND	EXPERIMENTAL INVESTIGATIONS ON THE LWR FUEL ROD BEHAVIOR	143745 P 19
IN THE PRESSURE TRANSIENT OF A NUCLEAR POWER PLANT IN THE	# EXPERIMENTAL RESEARCH ON SINGLE AND MULTIPLE TUBE ARRAYS	146488 P 14
	# EXPERIMENTAL STUDY OF CRITICAL TWO-PHASE FLOW (IN ENGLISH)*	143900 P 7
DETONATION OF SPHERICAL AIR-HYDROGEN GASEOUS MIXTURES (IN	# EXPERIMENTAL STUDY OF THE OVERPRESSURE GENERATED BY THE	144415 P 6
ON REINFORCED CONCRETE, CALCULATION AND COMPARISON WITH	EXPERIMENTAL TESTS (IN GERMAN)*IMPACT OF STEEL PROJECTILES	145638 P 21
UNDER INTERNAL PRESSURE (IN FRENCH)*	# EXPERIMENTAL TESTS ON BUCKLING OF ELLIPSOIDAL VESSEL HEADS	146870 P 8
ROOM TEMPERATURE (IN FRENCH)*	# EXPERIMENTAL TESTS ON RATCHET OF 304 AUSTENITIC STEEL, AT	146872 P 4
#SUPER CANON	EXPERIMENTS (IN ENGLISH)*	143914 P 2
#REFLUX-GRS ANALYSIS OF THE REFLOOD	EXPERIMENTS (RS 62) (IN ENGLISH)*	143903 P 14
#THE CONTAINMENT TEST FACILITY (	EXPERIMENTS C AND D) (IN GERMAN)*	143806 P 12
IN WATER-COOLED REACTORS, CONDENSATION IN CONTAINMENT BY	EXPERIMENTS C04 AND C1 TO C16 (IN GERMAN)* COOLING CIRCUIT	148273 P 12
OF LOCA EXPERIMENTS WITH RELAP4J CODE (ANALYSIS OF ROSA-II	EXPERIMENTS FOR COLD LEG BREAK RUNS 413 AND 312) (IN	144532 P 27
SPACERS (IN ENGLISH)* #LOCAL FLOW BLOCKAGE	EXPERIMENTS IN 37-PIN SODIUM COOLED BUNDLES WITH GRID	143907 P 36
PART 2 (IN GERMAN)* #ANALYSIS OF THE D SERIES	EXPERIMENTS OF RESEARCH PROJECT RS 50 (MODEL CONTAINMENT)	146798 P 11
EXPERIMENTS FOR COLD LEG BREAK RUNS 413 #ANALYSIS OF LOCA	EXPERIMENTS WITH RELAP4J CODE (ANALYSIS OF ROSA-II	144532 P 27
#LNG/WATER VAPOUR	EXPLOSIONS - ESTIMATES OF PRESSURES AND YIELDS*	148843 P 23
PLANTS (NPPS) AGAINST EXTERNAL EVENTS: EARTHQUAKES, FIRES,	EXPLOSIONS AND AIRCRAFT CRASHES (IN ENGLISH)*NUCLEAR POWER	143138 P 5
16.3) TO IBM STANDARD OPERATING SYSTEM MVS (WITH FORTRAN-H-	EXTENDED COMPILER) (IN ENGLISH)*CASE-CODE TRAC-P1 (VERSION	143966 P 14
AIRCRAFT #PROTECTION OF NUCLEAR POWER PLANTS (NPPS) AGAINST	EXTERNAL EVENTS: EARTHQUAKES, FIRES, EXPLOSIONS AND	143138 P 5

#THE CONTAINMENT TEST	FACILITY (EXPERIMENTS C AND D) (IN GERMAN)*	143806 P 12
FOR THE CALCULATION OF THE RATE OF VOIDING DURING A RAPID	FAILURE (BLOWDOWN) COMPARISON OF SEVERAL MODELS (IN FRENCH)	143751 P 2
#EFFECT OF HEAT GENERATION PROFILE IN PELLET ON FUEL	FAILURE BEHAVIOR (ENRICHMENT PARAMETER TEST IN NSRR) (IN	143974 P 28
VARIOUS PHASES OF THE CORE MELT ACCIDENT AFTER THE AFTER	FAILURE OF THE CORE SUPPORT STRUCTURE DUE TO THE FORMATION	145756 P 25

CALCULATION TO #THE COURSE OF EVENTS IN THE CONCRETE -	FAILURE PHASE OF THE HYPOTHETICAL CORE MELTDOWN ACCIDENT:	143379 P 15
THESE OCCURRENCES IN FAULT TREE #UNCERTAINTY OF THE	FAILURE RATE OF COMPONENTS AND THE APPARENT INFLUENCE OF	147779 P 24
PLANT ACCORDING TO INFLUENTIAL PARAMETERS# #MODEL OF THE	FAILURE RATES OF THE VALVES OF ST. LAURENT DES EAUX POWER	143339 P 6
DUE TO THE FORMATION OF MELT OR DUE TO PRESSURE VESSEL	FAILURE, PART II (IN GERMAN)#OF THE CORE SUPPORT STRUCTURE	145756 P 25
#PROGRESS REPORT ON	FAST BREEDER REACTOR DEVELOPMENT IN JAPAN, APRIL-JUNE 1978#	143343 P 30
IN FRACTURE MECHANICS RESEARCH ON MATERIALS USED IN	FAST SODIUM-COOLED REACTORS (IN GERMAN)# MEASUREMENTS	143805 P 26
AND THE APPARENT INFLUENCE OF THESE OCCURRENCES IN	FAULT TREE ANALYSIS (IN GERMAN)#FAILURE RATE OF COMPONENTS	147779 P 24
#	# FAULT TREE ANALYSIS WITH MULTISTATE COMPONENTS (IN ENGLISH)	147860 P 26
#PATREC, A COMPUTER CODE FOR	FAULT TREE CALCULATIONS (IN ENGLISH)#	143778 P 4
PRIMARY CIRCUIT BY FUEL RODS SHOWING MANUFACTURING	FAULTS (IN FRENCH)# OF A PRESSURIZED WATER REACTOR'S	143383 P 4
RESEARCH PROJECTS IN THE AREA OF REACTOR #REPORT OF THE	FEDERAL MINISTER FOR RESEARCH AND TECHNOLOGY CONCERNING	147483 P 18
BY THE MINISTRY FOR RESEARCH AND TECHNOLOGY OF THE	FEDERAL REPUBLIC OF GERMANY, 1978 (IN ENGLISH)# SPONSORED	147467 P 18
STUDY OF THE RELIABILITY OF THE PROTECTION SYSTEM OF THE	FESSENHEIM I PWR REACTOR (IN ENGLISH)#EVENT PROBLEM BY THE	143775 P 7
#IN-SITU TESTING OF HIGH EFFICIENCY	FILTERS AT AEE WINFRITH#	145677 P 21
#MAGENTIC FLUX METHOD,	FINAL REPORT (IN GERMAN)#	146793 P 13
#ERROR ANALYSIS OF THE AMPLITUDE CURVE,	FINAL REPORT (IN GERMAN)#	148370 P 13
AND ERROR RECONSTRUCTION WITH HELP OF TIME-OF-FLIGHT DATA,	FINAL REPORT (IN GERMAN)# #DISTURBANCE	148368 P 13
PRIMARY LOOP COMPONENTS OF DECOMMISSIONED NUCLEAR #RS236 -	FINAL REPORT CONTROLLED-BLASTING DEMOLITION OF RADIOACTIVE	147817 P 23
CALCULATIONAL METHOD FOR TWO DIMENSIONAL VISCOUS FLOW USING	FINITE ELEMENT METHOD (IN FRENCH)# #	143753 P 2
POWER PLANTS (NPPS) AGAINST EXTERNAL EVENTS: EARTHQUAKES,	FIRES, EXPLOSIONS AND AIRCRAFT CRASHES (IN ENGLISH)#	143138 P 5
THE RELIABILITY OF THE PROTECTION SYSTEM OF THE #A	FIRST APPROACH OF THE RARE EVENT PROBLEM BY THE STUDY OF	143775 P 7
PWR AND A BWR STANDARD PLANT, AND THE #INVESTIGATION OF THE	FIRST PHASE OF THE CORE MELT ACCIDENT WITH MELSIM-1 IN A	145458 P 25
#NUCLEAR SAFETY PROJECT	FIRST SEMIANNUAL REPORT 1978 (IN GERMAN)#	147794 P 15
OMEGA LOOP (IN FRENCH)# #DATA REDUCTION OF THE	FIRST TEST SERIES OF BLOWDOWN ON A TUBULAR TEST SECTION ON	143755 P 3
OF THERMAL CYCLING (IN ENGLISH)# #AN YSIS OF THE	FISSION PRODUCT RELEASE FROM A DEFECTED FUEL ROD - EFFECT	144595 P 5
#AN ANALYSIS OF T... ITIONAL	FISSION PRODUCT RELEASE PHENOMENA (IN ENGLISH & JAPANESE)#	144523 P 27
#DEVELOPMENT OF A SUCTION SYSTEM FOR INSTALLATIONS AND	FITTINGS (IN GERMAN)#	144828 P 23
HEMISPHERICAL STRATIFIED GASEOUS MIXTURES (IN ENGLISH)#	# FLAME PROPAGATION THROUGH UNCONFINED AND CONFINED	143078 P 8
#DISTURBANCE AND ERROR RECONSTRUCTION WITH HELP OF TIME-OF-	FLIGHT DATA, FINAL REPORT (IN GERMAN)#	148368 P 13
#EXPERIMENTAL STUDY OF CRITICAL TWO-PHASE	FLOW (IN ENGLISH)#	143906 P 7
#CRITICAL FLOW AND FLOW BLOCKAGE PHENOMENON FOR A T. PHASE	FLOW (IN FRENCH)#	143911 P 1
AND TWO-DIMENSIONAL CALCULATION OF A WATER-VAPOR NOZZLE	FLOW (IN GERMAN)# #COMPARISON BETWEEN A ONE-	143211 P 19
TRANSIENTS IN FULL-PRESSURE CONTAINMENTS OF WATER- #CO.	FLOW - A COMPUTER CODE FOR THE DETERMINATION OF PRESSURE	145636 P 16
FRENCH)# #CRITICAL	FLOW AND FLOW BLOCKAGE PHENOMENON FOR A TWO PHASE FLOW (IN	143911 P 1
THE REFLOODING PHASE OF A PWR'S CORE (IN FRENCH)#	# FLOW AND HEAT TRANSFER THERMOHYDRAULIC MODELISATION DURING	143928 P 1
WITH GRID SPACERS (IN ENGLISH)# #LOCAL	FLOW BLOCKAGE EXPERIMENTS IN 37-PIN SODIUM COOLED BUNDLES	143907 P 30
#CRITICAL FLOW AND	FLOW BLOCKAGE PHENOMENON FOR A TWO PHASE FLOW (IN FRENCH)#	143911 P 1
- A COMPUTERCODE FOR THE CALCULATION OF THE TWO-DIMENSIONAL	FLOW OF A COMPRESSIBLE MEDIUM IN COUPLED RECTANGULAR AREAS	146750 P 20
(I) A COMPUTER PROGRAM TO DESCRIBE THERMAL NON-EQUILIBRIUM	FLOW OF WATER IN SIMPLE PIPE SYSTEMS# #BUBBLE BUBBLE	140447 P 32
SECTION (IN JAPANESE & ENGLISH)#	# FLOW REDUCTION TRANSIENT BURNOUT IN AN ANNULAR TEST	148844 P 29
ENGLISH)# #ACOUSTIC NOISES WITH LOSS-OF-	FLOW SODIUM BOILING EXPERIMENT IN A 19-PIN BUNDLE (IN	143776 P 30
#CALCULATIONAL METHOD FOR TWO DIMENSIONAL VISCOUS	FLOW USING FINITE ELEMENT METHOD (IN FRENCH)#	143753 P 2
ATTAINMENT OF INTERPRETATIONAL MODELS FOR THE ESTIMATION OF	FLOW USING ULTRASONIC TESTS ON NUCLEAR REACTORS (IN GERMAN)	146486 P 13
#UNDIMENSIONAL MODELS FOR DIPHASE	FLOWS (IN FRENCH)#	143930 P 1
PRESSURIZED #EFFICIENT COMPUTATION OF THREE-DIMENSIONAL	FLUID-STRUCTURE INTERACTIONS DURING BLOWDOWN OF A	146804 P 20
DIMENSIONAL FLOW OF A COMPRESSIBLE MEDIUM IN COUPLED	# FLUST-2D - A COMPUTERCODE FOR THE CALCULATION OF THE TWO-	146750 P 20
INTERACTIONS DURING BLOWDOWN OF A PRESSURIZED WATER REACTOR-	FLUX (IN GERMAN)# OF THREE-DIMENSIONAL FLUID-STRUCTURE	146804 P 20
#MAGENTIC	FLUX METHOD, FINAL REPORT (IN GERMAN)#	146793 P 13
OF CONDITIONS OF A NUCLEAR POWER PLANT WITH A PWR	FOLLOWING A LOCA FOR PURPOSES OF STUDYING THE ENSUING	146795 P 24
HYDROGEN DISTRIBUTION IN A LIGHT WATER REACTOR CONTAINMENT	FOLLOWING A LOSS-OF-COOLANT ACCIDENT, QUICK LOOK REPORT 1 (	144582 P 11
HYDROGEN DISTRIBUTION IN A LIGHT WATER REACTOR CONTAINMENT	FOLLOWING A LOSS-OF-COOLANT ACCIDENT, QUICK LOOK REPORT 2 (	144286 P 12
LUNG CLEARANCE PATTERNS# #ASSESSMENTS OF RISK	FOLLOWING THE INHALATION OF PLUTONIUM OXIDE USING OBSERVED	145281 P 35
THE AFTER FAILURE OF THE CORE SUPPORT STRUCTURE DUE TO THE	FORMATION OF MELT O. DUE TO PRESSURE VESSEL FAILURE, PART	145756 P 25
(VERSION 16.3) TO IBM STANDARD OPERATING SYSTEM MVS (WITH	FORTRAN-H-EXTENDED COMPILER) (IN ENGLISH)# CODE TRAC-P1	143966 P 18
SODIUM-COOLED REACTORS (IN #SONIC-EMISSION MEASUREMENTS IN	FRACTURE MECHANICS RESEARCH ON MATERIALS USED IN FAST	143805 P 26
IN ENGLISH)#	# FRACTURE PROBABILITY EVALUATION OF A LWR PRESSURE VESSEL (	143876 P 5
COMPONENTS (IN #INVESTIGATION PROGRAM FOR THE TESTING OF A	FRACTURE SAFETY DEVICE PROTECTION SYSTEM FOR REACTOR	145155 P 22
SPECIFIC REACTOR MATERIALS AND COMPONENT PARTS AT IMPACT OF	FRAGMENTS AND PROJECTILES OF DIFFERENT MASS AND VELOCITY (	143803 P 13
#THE PRACTICE OF QUALITY ASSURANCE BY	FRAMATOME (IN ENGLISH)#	148665 P 9
#THE SAFETY OF NUCLEAR REACTORS IN	FRANCE (IN ENGLISH)#	143103 P 6
ELEMENTS FROM LIGHT WATER REACTORS (IN #THE SAFETY OF	FRENCH INSTALLATIONS FOR THE STORAGE OF IRRADIATED FUEL	147103 P 3
SAFETY ADVANCED CODE FOR PWR (IN FRENCH)#	# FRENCH THERMO-HYDRAULIC STUDIES FOR THE DEVELOPMENT OF	143758 P 6
#UNDIMENSIONAL MODELS FOR DIPHASE FLOWS (IN	FRENCH)#	143930 P 1
#CALCULATING PLASTI. EFORMATION OF STRUCTURES (IN	FRENCH)#	143875 P 3
#EXPERIENCE SUPER-CANON (IN	FRENCH)#	148735 P 9
PROBABILISTIC STUDY OF VESSEL BURST IN LIGHT WATER NSSS (IN	FRENCH)# #A	146869 P 8
FLOW AND FLOW BLOCKAGE PHENOMENON FOR A TWO PHASE FLOW (IN	FRENCH)# #CRITICAL	143911 P 1
IN RATCHET OF 304 AUSTENITIC STEEL, AT ROOM TEMPERATURE (IN	FRENCH)# #EXPERIMENTAL TESTS	146872 P 4
THE LOOP OMEGA INCLUDING A 36 DIRECT HEATED ROD BUNDLE (IN	FRENCH)# #BLOWDOWN OF A PART OF	143929 P 4
DIMENSIONAL VISCOUS FLOW USING FINITE ELEMENT METHOD (IN	FRENCH)# #CALCULATIONAL METHOD FOR TWO	14375' P 2
OF ELLIPSOIDAL VESSEL HEADS UNDER INTERNAL PRESSURE (IN	FRENCH)# #EXPERIMENTAL TESTS ON BUCKLING	1'' 2 P 8
FOR THE DEVELOPMENT OF SAFETY ADVANCED CODE FOR PWR (IN	FRENCH)# #FRENCH THERMO-HYDRAULIC STUDIES	.3758 P 6
DISTORSION PROBLEM (TENSION AND CYCLE TORSION) (IN	FRENCH)# #SHORT ANALYSIS OF A PROGRESSIVE	146871 P 8
GAS INSIDE A BUBLE RISING THROUGH A LIQUID CODE CAEU 1 (IN	FRENCH)# #STUDY OF THE RANGE OF VELOCITY OF	143210 P 2
SYSTEM OF EQUATIONS DERIVED FROM PARTIAL HYPERBOLICS (IN	FRENCH)# #CHARACTERISTICS AND RESOLUTIONS OF THE	147101 P 7
AND MECHANICAL BEHAVIOR OF A PWR FUEL ROD DURING A LOCA (IN	FRENCH)# #CUPIDON: A CODE DESCRIBING THE THERMAL	145871 P 1
OF BLOWDOWN ON A TUBULAR TEST SECTION ON OMEGA LOOP (IN	FRENCH)# #DATA REDUCTION OF THE FIRST TEST SERIES	143755 P 3
CIRCUIT BY FUEL RODS SHOWING MANUFACTURING FAULTS (IN	FRENCH)# OF A PRESSURIZED WATER REACTOR'S PRIMARY	143383 P 4
THE BLOWDOWN OF THE SECONDARY SIDE OF A STEAM GENERATOR (IN	FRENCH)# #ANA EXPERIMENTAL AND THEORETICAL STUDY OF	148734 P 5
AND DESCRIPTION OF EQUATIONS FOR HYDRODYNAMIC ANALYSIS (IN	FRENCH)# #STUDY OF THE STABILITY OF VARIOUS SYSTEMS	143756 P 2
DURING THE REFLOODING PHASE OF A PWR'S CORE (IN	FRENCH)# AND HEAT TRANSFER THERMOHYDRAULIC MODELISATION	143928 P 1
TEST EFFICIENCY NOT 100%, AND 1 OUT OF 2 STANDBY SYSTEM (IN	FRENCH)# FOR PERIODIC TESTING-NONNEGLIGIBLE TEST DURATION,	145296 P 1
A RAPID FAILURE (BLOWDOWN) COMPARISON OF SEVERAL MODELS (IN	FRENCH)# FOR THE CALCULATION OF THE RATE OF VOIDING DURING	143751 P 2
TRANSIENT CONDITIONS, ON A STAINLESS STEEL TEST SECTION (IN	FRENCH)# ON THE CONDENSATION OF AIR AND STEAM MIXTURES, IN	149000 P 7
OF IRRADIATED FUEL ELEMENTS FROM LIGHT WATER REACTORS (IN	FRENCH)#THE SAFETY OF FRENCH INSTALLATIONS FOR THE STORAGE	14.103 P 3
HEATING METHOD (ZRE02 H2O SYSTEM) (IN JAPANESE)#	# FUEL COOLANT INTERACTION EXPERIMENT BY DIRECT ELECTRICAL	148842 P 29
#THE ESSENTIAL SAFETY ASPECTS OF A CONFINED NUCLEAR	FUEL CYCLE (IN GERMAN)#	147102 P 16

AFTER RUPTURE OF THE PRIMARY COOLING #INVESTIGATION OF THE	PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT	145847 P 11
AFTER RUPTURE OF THE PRIMARY COOLING #INVESTIGATION OF THE	PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT	148273 P 12
#CRITICAL FLOW AND FLOW BLOCKAGE	PHENOMENON FOR A TWO PHASE FLOW (IN FRENCH)*	143911 P 1
EVALUATION IN PRESSURE #LASER HOLOGRAPHIC AND SPECKLE	PHOTOGRAPHY METHODS FOR DEFECT DETECTION AND STRAIN	120343 P 31
NOISES WITH LOSS-OF-FLOW SODIUM BOILING EXPERIMENT IN A 19-	PIN BUNDLE (IN ENGLISH)* #ACOUSTIC	143776 P 30
#LOCAL FLOW BLOCKAGE EXPERIMENTS IN 37-	PIN SODIUM COOLED BUNDLES WITH GRID SPACERS (IN ENGLISH)*	143907 P 36
TO DESCRIBE THERMAL NON-EQUILIBRIUM FLOW OF WATER IN SIMPLE	PIPE SYSTEMS* #BUBBLE BUBBLE II: A COMPUTER PROGRAM	140447 P 32
FAILURE RATES OF THE VALVES OF ST. LAURENT DES EAUX POWER	PLANT ACCORDING TO INFLUENTIAL PARAMETERS* #MODEL OF THE	143339 P 6
TUBE ARRAYS IN THE PRESSURE TRANSIENT OF A NUCLEAR POWER	PLANT IN THE LARGE EXPERIMENT AREA OF THE GKSS (IN GERMAN)*	146488 P 14
THE ENSUING #SPECIFICATION OF CONDITIONS OF A NUCLEAR POWER	PLANT WITH A PWR FOLLOWING A LOCA FOR PURPOSES OF STUDYING	146795 P 24
MELT ACCIDENT WITH MELSIM-I IN A PWR AND A BWR STANDARD	PLANT, AND THE COUPLING OFMELSIN-I AND BILANZ-1, PART I I	145458 P 25
#SAFETY CONTAINMENT OF NUCLEAR POWER	PLANTS (IN GERMAN)*	147176 P 17
PRIMARY LOOP COMPONENTS OF DECOMMISSIONED NUCLEAR POWER	PLANTS (IN GERMAN)* BLASTING DEMOLITION OF RADIOACTIVE	147817 P 23
AND INSTRUCTION OF OPERATING PERSONNEL AT NUCLEAR POWER	PLANTS (IN GERMAN)* OF MEDIA FOR THE INTENSIVE TRAINING	143768 P 21
IN FULL-PRESSURE CONTAINMENTS OF WATER-COOLED NUCLEAR POWER	PLANTS (IN GERMAN)* DETERMINATION OF PRESSURE TRANSIENTS	145636 P 16
EXPLOSIONS AND AIRCRAFT #PROTECTION OF NUCLEAR POWER	PLANTS (NPPS) AGAINST EXTERNAL EVENTS: EARTHQUAKES, FIRES,	143138 P 5
#CALCULATING	PLASTIC DEFORMATION OF STRUCTURES (IN FRENCH)*	143875 P 3
#EXAMINATION OF 3 NSST	PLATES RUPTURED IN AIR (IN GERMAN)*	146799 P 13
#ATTITUDES ON QUESTIONS PERTAINING TO NUCLEAR ENERGY:	PLUTONIUM (IN GERMAN)*	148363 P 18
#ASSESSMENTS OF RISK FOLLOWING THE INHALATION OF	PLUTONIUM OXIDE USING OBSERVED LUNG CLEARANCE PATTERNS*	145281 P 25
#STUDY OF	POLLUTANT DISPERSION IN WATER AND AIR (IN ENGLISH)*	146801 P 8
MODEL FOR THE TWO-DIMENSIONAL CALCULATION OF THE WATER	POOL-SWELL IN THE CONDENSATION CHAMBER OF A REACTOR SYSTEM	145757 P 15
GERMAN)* #RESULTS OF THE LOFT EXPERIMENT L1-4:	POST RUN CALCULATIONS USING THE COMPUTER CODE "URLFAN" (IN	148668 P 16
R/GRS ANALYSIS OF THE NONNUCLEAR LOFT-TESTS L1-4 (PRE AND	POST TEST CALCULATIONS) (IN GERMAN)* #RELAP-	143905 P 15
L1-5 (IN GERMAN AND ENGLISH)*	POST-EXPERIMENT CALCULATION OF THE NON-NUCLEAR LOFT TEST	148674 P 16
EXPERIMENT RS 109 (LOBI) CONTROL OF THE ELECTRICAL HEATING	POWER (IN GERMAN)* #BLOWDOWN -	143894 P 15
OF THE FAILURE RATES OF THE VALVES OF ST. LAURENT DES EAUX	POWER PLANT ACCORDING TO INFLUENTIAL PARAMETERS* #MODEL	143339 P 6
MULTIPLE TUBE ARRAYS IN THE PRESSURE TRANSIENT OF A NUCLEAR	POWER PLANT IN THE LARGE EXPERIMENT AREA OF THE GKSS (IN	146488 P 14
STUDYING THE #SPECIFICATION OF CONDITIONS OF A NUCLEAR	POWER PLANT WITH A PWR FOLLOWING A LOCA FOR PURPOSES OF	146795 P 24
#SAFETY CONTAINMENT OF NUCLEAR	POWER PLANTS (IN GERMAN)*	147176 P 17
IN FULL-PRESSURE CONTAINMENTS OF WATER-COOLED NUCLEAR	POWER PLANTS (IN GERMAN)* OF PRESSURE TRANSIENTS	145636 P 16
PRIMARY LOOP COMPONENTS OF DECOMMISSIONED NUCLEAR	POWER PLANTS (IN GERMAN)* DEMOLITION OF RADIOACTIVE	147817 P 23
TRAINING AND INSTRUCTION OF OPERATING PERSONNEL AT NUCLEAR	POWER PLANTS (IN GERMAN)* OF MEDIA FOR THE INTENSIVE	143768 P 21
FIRES, EXPLOSIONS AND AIRCRAFT #PROTECTION OF NUCLEAR	POWER PLANTS (NPPS) AGAINST EXTERNAL EVENTS: EARTHQUAKES,	143138 P 5
#KFA/KWU	POWER RAMPS FUEL ROD IRRADIATION TESTS 1976/77 (GERMAN)*	148675 P 23
#THE	PRACTICE OF QUALITY ASSURANCE BY FRAMATOME (IN ENGLISH)*	148069 P 9
#RELAP-R/GRS ANALYSIS OF THE NONNUCLEAR LOFT-TESTS L1-4 (	PRE AND POST TEST CALCULATIONS) (IN GERMAN)*	143905 P 15
	PREDICTION OF ROSA-III EXPERIMENT RUN 702 (IN JAPANESE)*	147064 P 28
BEHAVIOUR DURING LOCA (IN GERMAN)*	PRELIMINARY EMPIRICAL DESCRIPTION OF THE FUEL ROD	145156 P 22
EMISSION ANALYSIS: EXPANDED INSTRUMENTATION AND EVALUATION	PRELIMINARY EXPERIMENT ON LEAKAGE MONITORING USING SONIC	145872 P 10
ON BUCKLING OF ELLIPSOIDAL VESSEL HEADS UNDER INTERNAL	PRESSURE (IN FRENCH)* #EXPERIMENTAL TESTS	146870 P 8
A COMPUTER SUBROUTINE FOR GENERATING STEAM TABLES HAVING	PRESSURE AND DENSITY AS THE INDEPENDENT VARIABLES (IN	144527 P 28
CODE FOR THE DETERMINATION OF PRESSURE TRANSIENTS IN FULL-	PRESSURE CONTAINMENTS OF WATER-COOLED NUCLEAR POWER PLANTS	145636 P 16
OF THE PROCESSES IN A MULTIPLE COMPARTMENT CONTAINMENT BY	PRESSURE IN WATER-COOLED REACTORS WITH REFRIGERATED	143329 P 16
FROM THE COOLANT BLOWDOWN OF A WATER- #INVESTIGATION OF THE	PRESSURE TRANSIENT IN A MULTICOMPARTMENTED CONTAINMENT	145875 P 10
RESEARCH ON SINGLE AND MULTIPLE TUBE ARRAYS IN THE	PRESSURE TRANSIENT OF A NUCLEAR POWER PLANT IN THE LARGE	146488 P 14
COOLED #CO₂ FLOW - A COMPUTER CODE FOR THE DETERMINATION OF	PRESSURE TRANSIENTS IN FULL-PRESSURE CONTAINMENTS OF WATER-	145636 P 16
#FRACTURE PROBABILITY EVALUATION OF A LWR	PRESSURE VESSEL (IN ENGLISH)*	143870 P 5
SUPPORT STRUCTURE DUE TO THE FORMATION OF MELT OR DUE TO	PRESSURE VESSEL FAILURE, PART II (IN GERMAN)* OF THE CORE	145756 P 25
METHODS FOR DEFECT DETECTION AND STRAIN EVALUATION IN	PRESSURE VESSELS#LASER HOLOGRAPHIC AND SPECKLE PHOTOGRAPHY	120343 P 31
#LNG/WATER VAPOUR EXPLOSIONS - ESTIMATES OF	PRESSURES AND YIELDS*	148843 P 33
OSCARM-PI: A COMPUTER PROGRAM FOR	PRESSURIZED WATER REACTOR BLOWDOWN ANALYSIS (IN ENGLISH)*	147502 P 29
FLUID-STRUCTURE INTERACTIONS DURING BLOWDOWN OF A	PRESSURIZED WATER REACTOR-FLUX (IN GERMAN)* DIMENSIONAL	146804 P 20
SHOWING MANUFACTURING FAULTS (IN FRENCH)#CONTAMINATION OF A	PRESSURIZED WATER REACTOR'S PRIMARY CIRCUIT BY FUEL RODS	143383 P 4
OF THE RADIOACTIVITY IN THE PRIMARY SYSTEM OF	PRESSURIZED WATER REACTORS (IN GERMAN)* INVESTIGATIONS	144596 P 18
IN FRENCH)* #CONTAMINATION OF A PRESSURIZED WATER REACTOR'S	PRIMARY CIRCUIT BY FUEL RODS SHOWING MANUFACTURING FAULTS (	143383 P 4
WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE	PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS.	148273 P 12
WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE	PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS. QUICK	145846 P 10
WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE	PRIMARY COOLING CIRCUITS IN WATER-COOLED REACTORS.	145847 P 11
FINAL REPORT CONTROLLED-BLASTING DEMOLITION OF RADIOACTIVE	PRIMARY LOOP COMPONENTS OF DECOMMISSIONED NUCLEAR POWER	147817 P 23
#EXPERIMENTAL INVESTIGATIONS OF THE RADIOACTIVITY IN THE	PRIMARY SYSTEM OF PRESSURIZED WATER REACTORS (IN GERMAN)*	144596 P 18
#SELECTION OF EVENTS FOR A	PROBABILISTIC EVALUATION OF PWR SAFETY (IN ENGLISH)*	147175 P 9
INSTALLATIONS (IN ENGLISH)* #USE OF	PROBABILISTIC METHODS IN THE SAFETY EVALUATION OF NUCLEAR	147021 P 5
IN FRENCH)* #A	PROBABILISTIC STUDY OF VESSEL BURST IN LIGHT WATER NSSS (	146865 P 6
ENGLISH)* #FRACTURE	PROBABILITY EVALUATION OF A LWR PRESSURE VESSEL (IN	143870 P 5
#SHORT ANALYSIS OF A PROGRESSIVE DISTORSION	PROBLEM (TENSION AND CYCLE TORSION) (IN FRENCH)*	146871 P 8
SYSTEM OF THE #A FIRST APPROACH OF THE RARE EVENT	PROBLEM BY THE STUDY OF THE RELIABILITY OF THE PROTECTION	143779 P 7
#USE OF MARKOV PROCESSES FOR RELIABILITY	PROBLEMS (IN ENGLISH)*	143777 P 3
SOLUTIONS TO NEACRP BENCHMARK BWR LATTICE CELL	PROBLEMS* #A REVIEW OF INTERNATIONAL	147795 P 32
#USE OF MARKOV	PROCESSES FOR RELIABILITY PROBLEMS (IN ENGLISH)*	143777 P 3
PRESSURE IN WATER-COOLED REACTORS #INVESTIGATION OF THE	PROCESSES IN A MULTIPLE COMPARTMENT CONTAINMENT BY	143329 P 16
THERMAL CYCLING (IN ENGLISH)* #ANALYSIS OF THE FISSION	PRODUCT RELEASE FROM A DEFECTED FUEL ROD - EFFECT OF	144595 P 5
#AN ANALYSIS OF THE ADDITIONAL FISSION	PRODUCT RELEASE PHENOMENA (IN ENGLISH & JAPANESE)*	144523 P 27
#ACTIVATED CORROSION	PRODUCTS IN LWR LOOPS (IN GERMAN)*	143083 P 22
PARAMETER TEST IN NSRR) (IN #EFFECT OF HEAT GENERATION	PROFILE IN PELLET ON FUEL FAILURE BEHAVIOR (ENRICHMENT	143974 P 28
TO MULTI-COMPONENT, TWO-PHASE SYSTEMS - THE COMPUTER	PROGRAMS CRITS AND CRITTER* DISCHARGE MODEL APPLIED	146483 P 33
30, 1978 (#LIST OF REPORTS FROM THE REACTOR SAFETY RESEARCH	PROGRAMS OF BMFT, USNRC, EPRI AND JSTA, JULY 1-SEPTEMBER	144854 P 16
OCTOBER 1 #LIST OF REPORTS FROM THE REACTOR SAFETY RESEARCH	PROGRAMS OF BMFT, USNRC, EPRI, AND JSTA, REPORT PERIOD	146813 P 17
(IN FRENCH)* #SHORT ANALYSIS OF A	PROGRESSIVE DISTORSION PROBLEM (TENSION AND CYCLE TORSION)	146871 P 8
#NUCLEAR SAFETY	PROJECT FIRST SEMIANNUAL REPORT 1978 (IN GERMAN)*	147794 P 19
#ANALYSIS OF THE D SERIES E RIMENTS OF RESEARCH	PROJECT RS 50 (MODEL CONTAINMENT) PART 2 (IN GERMAN)*	146758 P 11
MATERIALS AND COMPONENT PARTS AT IMPACT OF FRAGMENTS AND	PROJECTILES OF DIFFERENT MASS AND VELOCITY (IN GERMAN)*	143803 P 13
COMPARISON WITH EXPERIMENTAL TESTS (IN #IMPACT OF STEEL	PROJECTILES ON REINFORCED CONCRETE, CALCULATION AND	145638 P 21
MINISTER FOR RESEARCH AND TECHNOLOGY CONCERNING RESEARCH	PROJECTS IN THE AREA OF REACTOR SAFETY REPORTING PERIOD	147483 P 16
TECHNOLOGY OF THE #ANNUAL REPORT ON REACTOR SAFETY RESEARCH	PROJECTS SPONSORED BY THE MINISTRY FOR RESEARCH AND	147467 P 18
#INITIATION OF DETONATION OF HYDROGEN-AIR MIXTURES AND	PROPAGATION OF SHOCK WAVES IN THE ENVIRONMENT (IN GERMAN)*	144280 P 12
WALLED STRUCTURES (IN GERMAN)*	PROPAGATION OF SIMULATED SONIC EMISSION-IMPULSES IN THICK	144196 P 10

RESEARCH PROGRAM SPONSORED BY BMFT IN THE AREA OF REACTOR	SAFETY, JULY 1-SEPTEMBER 30, 1978 (IN GERMAN)* ON THE	145165 P 16
#RESEARCH PROGRAM ON REACTOR	SAFETY, 3D EXPERIMENT (IN GERMAN)*	146671 P 21
IN THE MEASUREMENT TECHNIQUE, OF SONIC-EMISSION ANALYSIS (	SEA) (IN GERMAN)* #IMPROVEMENT	144158 P 11
#RELIABILITY ASSESSMENT OF THE	SECONDARY CONTAINMENT OF A PWR (IN GERMAN)*	146820 P 24
EXPERIMENTAL AND THEORETICAL STUDY OF THE BLOWDOWN OF THE	SECONDARY SIDE OF A STEAM GENERATOR (IN FRENCH)* #ANA	148734 P 5
IN TRANSIENT CONDITIONS, ON A STAINLESS STEEL TEST	SECTION (IN FRENCH)* OF AIR AND STEAM MIXTURES,	149000 P 7
#FLOW REDUCTION TRANSIENT BURNOUT IN AN ANNULAR TEST	SECTION (IN JAPANESE & ENGLISH)*	148844 P 25
OF THE FIRST TEST SERIES OF BLOWDOWN ON A TUBULAR TEST	SECTION ON OMEGA LOOP (IN FRENCH)* #DATA REDUCTION	143755 P 3
SAFETY (IN ENGLISH)* #	SELECTION OF EVENTS FOR A PROBABILISTIC EVALUATION OF PWR	147175 P 5
#NUCLEAR SAFETY PROJECT FIRST	SEMIANNUAL REPORT 1978 (IN GERMAN)*	147754 P 19
#PHASE	SEPARATION (IN GERMAN)*	143904 P 24
CONTAINMENT) PART 2 (IN GERMAN)* #ANALYSIS OF THE D	SERIES EXPERIMENTS OF RESEARCH PROJECT RS 50 (MODEL	146798 P 11
(IN FRENCH)* #DATA REDUCTION OF THE FIRST TEST	SERIES OF BLOWDOWN ON A TUBULAR TEST SECTION ON OMEGA LOOP	143755 P 3
#COMPREHENSIVE SUMMARY OF THE THEORETICAL STUDIES ON THE D-	SERIES OF THE RESEARCH PROGRAM RS 50 (MODEL C[illegible]NMENT)	148272 P 11
OF VOIDING DURING A RAPID FAILURE (BLOWDOWN) COMPARISON OF	SEVERAL MODELS (IN FRENCH)#FOR THE CALCULATIO[illegible] OF THE RATE	143751 P 2
ENGLISH)* #A PROPOSED WIND	SHIFT MODEL FOR THE GERMAN REACTOR SAFETY STUDY (IN	148006 P 21
OF DETONATION OF HYDROGEN-AIR MIXTURES AND PROPAGATION OF	SHOCK WAVES IN THE ENVIRONMENT (IN GERMAN)* #INITIATION	144280 P 12
TENSION AND CYCLE TORSION) (IN FRENCH)* #	SHORT ANALYSIS OF A PROGRESSIVE DISTORSION PROBLEM (	146871 P 8
A PRESSURIZED WATER REACTOR'S PRIMARY CIRCUIT BY FUEL RODS	SHOWING MANUFACTURING FAULTS (IN FRENCH)#CONTAMINATION OF	143383 P 4
AND THEORETICAL STUDY OF THE BLOWDOWN OF THE SECONDARY	SIDE OF A STEAM GENERATOR (IN FRENCH)* #ANA EXPERIMENTAL	148734 P 5
APPLIED TO MULTI-COMPONENT, TWO-PHASE SYSTEMS - THE #A	SIMPLE HOMOGENEOUS EQUILIBRIUM CRITICAL DISCHARGE MODEL	146483 P 33
TO DESCRIBE THERMAL NON-EQUILIBRIUM FLOW OF WATER IN	SIMPLE PIPE SYSTEMS* #BUBBLE BUBBLE II: A COMPUTER PROGRAM	146447 P 32
STRUCTURES (IN GERMAN)* #PROPAGATION OF	SIMULATED SONIC EMISSION-IMPULSES IN THICK WALLED	144156 P 10
INFLUENCE OF THE INTERNAL CORROSION ON THE MECHANICAL #	SIMULATION OF THE CHEMICAL STATE OF IRRADIATED OXIDE FUEL:	148673 P 26
OF A NUCLEAR POWER PLANT IN THE #EXPERIMENTAL RESEARCH ON	SINGLE AND MULTIPLE TUBE ARRAYS IN THE PRESSURE TRANSIENT	146488 P 14
#IN-	SITU TESTING OF HIGH EFFICIENCY FILTERS AT AEE WINFRITH*	145877 P 31
#AN APPRAISAL OF SUBCOOLED BOILING AND	SLIP RATIO FROM MEASUREMENTS MADE IN LINGEN BWR*	148847 P 31
#ACOUSTIC NOISES WITH LOSS-OF-FLOW	SODIUM BOILING EXPERIMENT IN A 19-PIN BUNDLE (IN ENGLISH)*	143776 P 36
#LOCAL FLOW BLOCKAGE EXPERIMENTS IN 37-PIN	SODIUM COOLED BUNDLES WITH GRID SPACERS (IN ENGLISH)*	143907 P 36
IN FRACTURE MECHANICS RESEARCH ON MATERIALS USED IN FAST	SODIUM-COOLED REACTORS (IN GERMAN)* EMISSION MEASUREMENTS	143805 P 26
HIGH MODEL (#DEVELOPMENT OF A CALCULATIONAL PROGRAM FOR THE	SOLUTION OF THE NEUTRONIC EQUATION OF A MULTIDIMENSIONAL	148672 P 17
#A REVIEW OF INTERNATIONAL	SOLUTIONS TO NEACRP BENCHMARK BWR LATTICE CELL PROBLEMS*	147755 P 32
#PRELIMINARY EXPERIMENT ON LEAKAGE MONITORING USING	SONIC EMISSION ANALYSIS: EXPANDED INSTRUMENTATION AND	145872 P 10
GERMAN)* #PROPAGATION OF SIMULATED	SONIC EMISSION-IMPULSES IN THICK WALLED STRUCTURES (IN	144156 P 10
#IMPROVEMENT IN THE MEASUREMENT TECHNIQUES OF	SONIC-EMISSION ANALYSIS (SEA) (IN GERMAN)*	144158 P 11
ON MATERIALS USED IN FAST SODIUM-COOLED REACTORS (IN #	SONIC-EMISSION MEASUREMENTS IN FRACTURE MECHANICS RESEARCH	143805 P 26
EXPERIMENTS IN 37-PIN SODIUM COOLED BUNDLES WITH GRID	SPACERS (IN ENGLISH)* #LOCAL FLOW BLOCKAGE	143907 P 36
HAVING PRESSURE AND DENSITY AS THE INDEPENDENT VARIABLES (#	SPADE: A COMPUTER SUBROUTINE FOR GENERATING STEAM TABLES	144527 P 28
OF FRAGMENTS AND PROJECTILES OF DIFFERENT MASS #BEHAVIOR OF	SPECIFIC REACTOR MATERIALS AND COMPONENT PARTS AT IMPACT	143803 P 13
A PWR FOLLOWING A LOCA FOR PURPOSES OF STUDYING THE #	SPECIFICATION OF CONDITIONS OF A NUCLEAR POWER PLANT WITH	146795 P 24
STRAIN EVALUATION IN PRESSURE VESSELS#LASER HOLOGRAPHIC AND	SPECKLE PHOTOGRAPHY METHODS FOR DEFECT DETECTION AND	120343 P 31
STUDY OF THE OVERPRESSURE GENERATED BY THE DETONATION OF	SPHERICAL AIR-HYDROGEN GASEOUS MIXTURES (IN ENGLISH)*	144415 P 6
SEPTEMBER 30, 1978 (IN #REPORT ON THE RESEARCH PROGRAM	SPONSORED BY BMFT IN THE AREA OF REACTOR SAFETY, JULY 1-	145165 P 16
THE #ANNUAL REPORT ON REACTOR SAFETY RESEARCH PROJECTS	SPONSORED BY THE MINISTRY FOR RESEARCH AND TECHNOLOGY OF	147467 P 18
PARAMETERS* #MODEL OF THE FAILURE RATES OF THE VALVES OF	ST. LAURENT DES EAUX POWER PLANT ACCORDING TO INFLUENTIAL	143339 P 6
FOR HYDRODYNAMIC ANALYSIS (IN FRENCH)* #STUDY OF THE	STABILITY OF VARIOUS SYSTEMS AND DESCRIPTION OF EQUATIONS	143758 P 2
OF AIR AND STEAM MIXTURES, IN TRANSIENT CONDITIONS, ON A	STAINLESS STEEL TEST SECTION (IN FRENCH)* THE CONDENSATION	149000 P 7
CONVERSION OF THE CASE-CODE TRAC-P1 (VERSION 16.3) TO IBM	STANDARD OPERATING SYSTEM MVS (WITH FORTRAN-H-EXTENDED	143968 P 14
OF THE CORE MELT ACCIDENT WITH MELSIM-1 IN A PWR AND A BWR	STANDARD PLANT, AND THE COUPLING OFMELSIM-1 AND BILANZ-1,	145458 P 25
TEST DURATION, TEST EFFICIENCY NOT 100%, AND 1 OUT OF 2	STANDBY SYSTEM (IN FRENCH)* PERIODIC TESTING-NONNEGLIGIBLE	145296 P 1
CORROSION ON THE MECHANICAL #SIMULATION OF THE CHEMICAL	STATE OF IRRADIATED OXIDE FUEL: INFLUENCE OF THE INTERNAL	148673 P 26
OF A MASS-DENSITY METHOD FOR TRANSIENT TWO PHASE	STATE USING ATOMIC RESONANCE (IN GERMAN)* #DEVELOPMENT	144157 P 25
INVESTIGATIONS ON THE LWR FUEL ROD BEHAVIOR UNDER ACCIDENT #	STATUS AND RESULTS OF THE THEORETICAL AND EXPERIMENTAL	143745 P 19
TRANSIENT THERMAL-HYDRAULIC ANALYSIS* #THE	STATUS OF RELAP-UK MK III AT JULY 1976- A PROGRAM FOR	145640 P 31
ENGLISH)* #CREEP RUPTURE AT NON-	STEADY STRESS AND TEMPERATURE LOADING CONDITIONS (IN	143089 P 19
#ANALYSES OF BOILING-WATER REACTOR	STEAM CHUGGING (IN ENGLISH)*	146805 P 17
STUDY OF THE BLOWDOWN OF THE SECONDARY SIDE OF A	STEAM GENERATOR (IN FRENCH)* EXPERIMENTAL AND THEORETICAL	148734 P 5
STEEL TEST SECTION (#STUDY ON THE CONDENSATION OF AIR AND	STEAM MIXTURES, IN TRANSIENT CONDITIONS, ON A STAINLESS	149000 P 7
INDEPENDENT #SPADE: A COMPUTER SUBROUTINE FOR GENERATING	STEAM TABLES HAVING PRESSURE AND DENSITY AS THE	144527 P 28
COMPARISON WITH EXPERIMENTAL TESTS (IN GERMAN)* #IMPACT OF	STEEL PROJECTILES ON REINFORCED CONCRETE, CALCULATION AND	145638 P 21
AND STEAM MIXTURES, IN TRANSIENT CONDITIONS, ON A STAINLESS	STEEL TEST SECTION (IN FRENCH)* ON THE CONDENSATION OF AIR	149000 P 7
#EXPERIMENTAL TESTS ON RATCHET OF 304 AUSTENITIC	STEEL, AT ROOM TEMPERATURE (IN FRENCH)*	146872 P 4
REACTORS (IN #THE SAFETY OF FRENCH INSTALLATIONS FOR THE	STORAGE OF IRRADIATED FUEL ELEMENTS FROM LIGHT WATER	147103 P 3
AND SPECKLE PHOTOGRAPHY METHODS FOR DEFECT DETECTION AND	STRAIN EVALUATION IN PRESSURE VESSELS* #LASER HOLOGRAPHIC	120343 P 31
PROPAGATION THROUGH UNCONFINED AND CONFINED HEMISPERICAL	STRATIFIED GASEOUS MIXTURES (IN ENGLISH)* #FLAME	143078 P 8
#CREEP RUPTURE AT NON-STEADY	STRESS AND TEMPERATURE LOADING CONDITIONS (IN ENGLISH)*	143089 P 19
MELT ACCIDENT AFTER THE AFTER FAILURE OF THE CORE SUPPORT	STRUCTURE DUE TO THE FORMATION OF MELT OR DUE TO PRESSURE	145758 P 25
WATER #EFFICIENT COMPUTATION OF THREE-DIMENSIONAL FLUID-	STRUCTURE INTERACTIONS DURING BLOWDOWN OF A PRESSURIZED	146804 P 20
#CALCULATING PLASTIC DEFORMATION OF	STRUCTURES (IN FRENCH)*	143875 P 3
OF SIMULATED SONIC EMISSION-IMPULSES IN THICK WALLED	STRUCTURES (IN GERMAN)* #PROPAGATION	144156 P 10
PWR (IN FRENCH)* #FRENCH THERMO-HYDRAULIC	STUDIES FOR THE DEVELOPMENT OF SAFETY ADVANCED CODE FOR	143758 P 6
MODEL #COMPREHENSIVE SUMMARY OF THE THEORETICAL	STUDIES ON THE D-SERIES OF THE RESEARCH PROGRAM RS 50 (	148272 P 11
#TIRION 4 -A COMPUTER PROGRAM FOR USE IN NUCLEAR SAFETY	STUDIES*	147602 P 33
#A PROPOSED WIND SHIFT MODEL FOR THE GERMAN REACTOR SAFETY	STUDY (IN ENGLISH)*	148006 P 21
#EXPERIMENTAL	STUDY OF CRITICAL TWO-PHASE FLOW (IN ENGLISH)*	143900 P 7
* #	STUDY OF POLLUTANT DISPERSION IN WATER AND AIR (IN ENGLISH)	146801 P 8
GENERATOR (IN FRENCH)* #ANA EXPERIMENTAL AND THEORETICAL	STUDY OF THE BLOWDOWN OF THE SECONDARY SIDE OF A STEAM	148734 P 5
SPHERICAL AIR-HYDROGEN GASEOUS MIXTURES (IN #EXPERIMENTAL	STUDY OF THE OVERPRESSURE GENERATED BY THE DETONATION OF	144415 P 6
RISING THROUGH A LIQUID CODE CABU I (IN FRENCH)* #	STUDY OF THE RANGE OF VELOCITY OF GAS INSIDE A BUBLE	143210 P 2
#A FIRST APPROACH OF THE RARE EVENT PROBLEM BY THE	STUDY OF THE RELIABILITY OF THE PROTECTION SYSTEM OF THE	143779 P 7
OF EQUATIONS FOR HYDRODYNAMIC ANALYSIS (IN FRENCH)* #	STUDY OF THE STABILITY OF VARIOUS SYSTEMS AND DESCRIPTION	143758 P 2
#A PROBABILISTIC	STUDY OF VESSEL BURST IN LIGHT WATER NSSS (IN FRENCH)*	146869 P 8
TRANSIENT CONDITIONS, ON A STAINLESS STEEL TEST SECTION (#	STUDY ON THE CONDENSATION OF AIR AND STEAM MIXTURES, IN	149000 P 7
POWER PLANT WITH A PWR FOLLOWING A LOCA FOR PURPOSES OF	STUDYING THE ENSUING DECONTAMINATION AND TRANSPORT (IN	146795 P 24
LINGEN BWR* #AN APPRAISAL OF	SUBCOOLED BOILING AND SLIP RATIO FROM MEASUREMENTS MADE IN	148847 P 31
DENSITY AS THE INDEPENDENT VARIABLES (IN #SPADE: A COMPUTER	SUBROUTINE FOR GENERATING STEAM TABLES HAVING PRESSURE AND	144527 P 28
OF A COOLING SYSTEM BLOWDOWN ACCIDENT AND THE	SUBSEQUENT THERMAL TRANSIENT IN LIGHT-WATER AND HIGH	145874 P 12

#DEVELOPMENT OF A	SUCTION SYSTEM FOR INSTALLATIONS AND FITTINGS (IN GERMAN)*	144828 P 23
RESEARCH PROGRAM RS 50 (MODEL CONTAINMENT) #COMPREHENSIVE	SUMMARY OF THE THEORETICAL STUDIES ON THE D-SERIES OF THE	148272 P 11
#	SUPER CANON EXPERIMENTS (IN ENGLISH)*	143914 P 2
#EXPERIENCE	SUPER-CANON (IN FRENCH)*	148735 P 5
OF THE PRIMARY COOLING CIRCUITS IN WATER-COOLED REACTORS.	SUPPLEMENTAL RESEARCH DOCUMENTATION DIS (IN GERMAN)*	145847 P 11
THE CORE MELT ACCIDENT AFTER THE AFTER FAILURE OF THE CORE	SUPPORT STRUCTURE DUE TO THE FORMATION OF MELT OR DUE TO	145756 P 25
MODEL FOR THE TWO-DIMENSIONAL CALCULATION OF THE WATER POOL-	SWELL IN THE CONDENSATION CHAMBER OF A REACTOR SYSTEM (IN	145757 P 15
MEDIA FOR THE INTENSIVE TRAINING AND #DEVELOPMENT AND	SYNTHESIS OF AN EDUCATIONAL SYSTEM USING COMBINATION OF	143768 P 21
DURATION, TEST EFFICIENCY NOT 100%, AND 1 OUT OF 2 STANDBY	SYSTEM (IN FRENCH)*FOR PERIODIC TESTING-NONNEGLIGIBLE TEST	145296 P 1
WATER POOL-SWELL IN THE CONDENSATION CHAMBER OF A REACTOR	SYSTEM (IN GERMAN)* THE TWO-DIMENSIONAL CALCULATION OF THE	145757 P 15
TRANSFER COEFFICIENT IN THE CONTAINMENT DURING A COOLING	SYSTEM BLOWDOWN (IN GERMAN)* DETERMINATION OF THE HEAT	148297 P 10
TRANSIENT IN LIGHT-#COMPARATIVE INVESTIGATIONS OF A COOLING	SYSTEM BLOWDOWN ACCIDENT AND THE SUBSEQUENT THERMAL	145874 P 12
#DEVELOPMENT OF A SUCTION	SYSTEM FOR INSTALLATIONS AND FITTINGS (IN GERMAN)*	144828 P 23
FOR THE TESTING OF A FRACTURE SAFETY DEVICE PROTECTION	SYSTEM FOR REACTOR COMPONENTS (IN GERMAN)* PROGRAM	145155 P 22
GERMAN)* #INSTRUMENTATION	SYSTEM FOR THE GAS-MULTIPLE TUBE RESEARCH PROGRAM (IN	146796 P 13
CASL-CODE TRAC-P1 (VERSION 16.3) TO IBM STANDARD OPERATING	SYSTEM MVS (WITH FORTRAN-H-EXTENDED COMPILER) (IN ENGLISH)*	143966 P 14
FRENCH)* #CHARACTERISTICS AND RESOLUTIONS OF THE	SYSTEM OF EQUATIONS DERIVED FROM PARTIAL HYPERBOLICS (IN	147101 P 7
INVESTIGATIONS OF THE RADIOACTIVITY IN THE PRIMARY	SYSTEM OF PRESSURIZED WATER REACTORS (IN GERMAN)*	144596 P 18
PROBLEM BY THE STUDY OF THE RELIABILITY OF THE PROTECTION	SYSTEM OF THE BIESSENHEIM I PWR REACTOR (IN ENGLISH)* EVENT	143779 P 7
TRAINING AND #DEVELOPMENT AND SYNTHESIS OF AN EDUCATIONAL	SYSTEM USING COMBINATION OF MEDIA FOR THE INTENSIVE	143768 P 21
EXPERIMENT BY DIRECT ELECTRICAL HEATING METHOD (ZR(O)2- H2O	SYSTEM) (IN JAPANESE)* #FUEL COOLANT INTERACTION	148842 P 29
DISCHARGE MODEL APPLIED TO MULTI-COMPONENT, TWO-PHASE	SYSTEMS - THE COMPUTER PROGRAMS CRITS AND CRITTER*CRITICAL	146483 P 33
ANALYSIS (IN FRENCH)* #STUDY OF THE STABILITY OF VARIOUS	SYSTEMS AND DESCRIPTION OF EQUATIONS FOR HYDRODYNAMIC	143756 P 2
CONTAINMENT (IN GERMAN)* #INVESTIGATION AND DEVELOPMENT OF	SYSTEMS LIMITING THE H2-CONCENTRATION IN THE BWR	147473 P 23
TESTING-NONNEGLIGIBLE TEST DURATION, TEST EFFICIENCY NOT #	SYSTEMS WHICH ARE UNCONNECTED AND WAITING FOR PERIODIC	145296 P 1
THERMAL NON-EQUILIBRIUM FLOW OF WATER IN SIMPLE PIPE	SYSTEMS* #HUBBLE BUBBLE II: A COMPUTER PROGRAM TO DESCRIBE	140447 P 32

#SPADE: A COMPUTER SUBROUTINE FOR GENERATING STEAM	TABLES HAVING PRESSURE AND DENSITY AS THE INDEPENDENT	144527 P 28
#IMPROVEMENT IN THE MEASUREMENT	TECHNIQUES OF SONIC-EMISSION ANALYSIS (SEA) (IN GERMAN)*	144198 P 11
REACTOR #REPORT OF THE FEDERAL MINISTER FOR RESEARCH AND	TECHNOLOGY CONCERNING RESEARCH PROJECTS IN THE AREA OF	147483 P 18
PROJECTS SPONSORED BY THE MINISTRY FOR RESEARCH AND	TECHNOLOGY OF THE FEDERAL REPUBLIC OF GERMANY, 1978 (IN	147467 P 16
TESTS ON RATCHET OF 304 AUSTENITIC STEEL, AT ROOM	TEMPERATURE (IN FRENCH)* #EXPERIMENTAL	146872 P 4
INGRESS, REACTIVITY #COMPREHENSIVE DAMAGE ANALYSIS FOR HIGH	TEMPERATURE GAS REACTORS, PHASE II. WATER INGRESS, AIR	148670 P 10
#CREEP RUPTURE AT NON-STEADY STRESS AND	TEMPERATURE LOADING CONDITIONS (IN ENGLISH)*	143089 P 19
THE SUBSEQUENT THERMAL TRANSIENT IN LIGHT-WATER AND HIGH	TEMPERATURE REACTORS (IN GERMAN)* BLOWDOWN ACCIDENT AND	145874 P 12
#SHORT ANALYSIS OF A PROGRESSIVE DISTORSION PROBLEM	TENSION AND CYCLE TORSION) (IN FRENCH)*	146871 P 8
#EVALUATION OF THE 25-ROD BUNDLE	TEST (RS-37C) WITH THE CALCULATIONAL PROGRAM (IN GERMAN)*	143902 P 14
ANALYSIS OF THE NONNUCLEAR LOFT-TESTS L1-4 (PRE AND POST	TEST CALCULATIONS) (IN GERMAN)* #RELAP-R/GRS	143905 P 15
CIRCULATION ON LOCA PHENOMENA IN LARGEST COLD LEG #ROSA-II	TEST DATA REPORT 12 EFFECTS OF ECCS INJECTION AND PUMP	143891 P 27
UNCONNECTED AND WAITING FOR PERIODIC TESTING-NONNEGLIGIBLE	TEST DURATION, TEST EFFICIENCY NOT 100%, AND 1 OUT OF 2	145296 P 1
WAITING FOR PERIODIC TESTING-NONNEGLIGIBLE TEST DURATION,	TEST EFFICIENCY NOT 100%, AND 1 OUT OF 2 STANDBY SYSTEM (	145296 P 1
#THE CONTAINMENT	TEST FACILITY (EXPERIMENTS C AND D) (IN GERMAN)*	143806 P 12
IN PELLET ON FUEL FAILURE BEHAVIOR (ENRICHMENT PARAMETER	TEST IN NSRR) (IN ENGLISH & JAPANESE)* GENERATION PROFILE	143974 P 28
#POST-EXPERIMENT CALCULATION OF THE NON-NUCLEAR LOFT	TEST L1-5 (IN GERMAN AND ENGLISH)*	148674 P 16
MIXTURES, IN TRANSIENT CONDITIONS, ON A STAINLESS STEEL	TEST SECTION (IN FRENCH)*THE CONDENSATION OF AIR AND STEAM	149000 P 7
#FLOW REDUCTION TRANSIENT BURNOUT IN AN ANNULAR	TEST SECTION (IN JAPANESE & ENGLISH)*	148844 P 29
REDUCTION OF THE FIRST TEST SERIES OF BLOWDOWN ON A TUBULAR	TEST SECTION ON OMEGA LOOP (IN FRENCH)* #DATA	143755 P 3
LOOP (IN FRENCH)* #DATA REDUCTION OF THE FIRST	TEST SERIES OF BLOWDOWN ON A TUBULAR TEST SECTION ON OMEGA	143755 P 3
REACTOR COMPONENTS (IN #INVESTIGATION PROGRAM FOR THE	TESTING OF A FRACTURE SAFETY DEVICE PROTECTION SYSTEM FOR	145155 P 22
#IN-SITU	TESTING OF HIGH EFFICIENCY FILTERS AT ACE #INFRITH*	145877 P 31
#SYSTEMS WHICH ARE UNCONNECTED AND WAITING FOR PERIODIC	TESTING-NONNEGLIGIBLE TEST DURATION, TEST EFFICIENCY NOT	145296 P 1
CONCRETE, CALCULATION AND COMPARISON WITH EXPERIMENTAL	TESTS (IN GERMAN)* OF STEEL PROJECTILES ON REINFORCED	145638 P 21
#RELAP-R/GRS ANALYSIS OF THE NONNUCLEAR LOFT-	TESTS L1-4 (PRE AND POST TEST CALCULATIONS) (IN GERMAN)*	143905 P 15
INTERNAL PRESSURE (IN FRENCH)* #EXPERIMENTAL	TESTS ON BUCKLING OF ELLIPSOIDAL VESSEL HEADS UNDER	146870 P 8
MODELS FOR THE ESTIMATION OF FLOW USING ULTRASONIC	TESTS ON NUCLEAR REACTORS (IN GERMAN)* OF INTERPRETATIONAL	146486 P 13
TEMPERATURE (IN FRENCH)* #EXPERIMENTAL	TESTS ON RATCHET OF 304 AUSTENITIC STEEL, AT ROOM	146872 P 4
#KFA/KWU POWER RAMPS FUEL ROD IRRADIATION	TESTS 1976/77 (GERMAN)*	148675 P 23
FUEL ROD BEHAVIOR UNDER ACCIDENT #STATUS AND RESULTS OF THE	THEORETICAL AND EXPERIMENTAL INVESTIGATIONS ON THE LWR	143745 P 19
THE INITIAL BLOWDOWN PHASE, PARTS A, B, & #EXPERIMENTAL AND	THEORETICAL RESEARCH ON THE THERMAL HYDRAULIC BEHAVIOR IN	143901 P 24
PROGRAM RS 50 (MODEL #COMPREHENSIVE SUMMARY OF THE	THEORETICAL STUDIES ON THE D-SERIES OF THE RESEARCH	148272 P 11
A STEAM GENERATOR (IN FRENCH)* #ANA EXPERIMENTAL AND	THEORETICAL STUDY OF THE BLOWDOWN OF THE SECONDARY SIDE OF	148734 P 5
LOCA (IN FRENCH)* #CUPIDON: A CODE DESCRIBING THE	THERMAL AND MECHANICAL BEHAVIOR OF A PWR FUEL ROD DURING A	145871 P 1
PRODUCT RELEASE FROM A DEFECTED FUEL ROD - EFFECT OF	THERMAL CYCLING (IN ENGLISH)* #ANALYSIS OF THE FISSION	144595 P 5
PARTS A, B, & #EXPERIMENTAL AND THEORETICAL RESEARCH ON THE	THERMAL HYDRAULIC BEHAVIOR IN THE INITIAL BLOWDOWN PHASE,	143901 P 24
#	THERMAL INTERACTIONS BETWEEN CERROBEND AND WATER*	148848 P 22
#	THERMAL INTERACTIONS BETWEEN CERROTRU AND WATER*	1[illegible]910 P 32
SYSTEMS* #HUBBLE BUBBLE II: A COMPUTER PROGRAM TO DESCRIBE	THERMAL NON-EQUILIBRIUM FLOW OF WATER IN SIMPLE PIPE	140447 P 32
OF A COOLING SYSTEM BLOWDOWN ACCIDENT AND THE SUBSEQUENT	THERMAL TRANSIENT IN LIGHT-WATER AND HIGH TEMPERATURE	145874 P 12
OF RELAP-UK MK III AT JULY 1976- A PROGRAM FOR TRANSIENT	THERMAL-HYDRAULIC ANALYSIS* #THE STATUS	145640 P 31
ENGLISH)* #ASCOT-1: A COMPUTER PROGRAM FOR ANALYZING THE	THERMO-HYDRAULIC BEHAVIOR IN A PWR CORE DURING A LOCA (IN	144529 P 27
ADVANCED CODE FOR PWR (IN FRENCH)* #FRENCH	THERMO-HYDRAULIC STUDIES FOR THE DEVELOPMENT OF SAFETY	143758 P 6
ARGUMENTS* #DO ELECTRONIC TRANSITIONS CONTRIBUTE TO THE	THERMODYNAMICS OF CONDENSED UO(2)? A REVIEW OF THE	148846 P 34
#THE ELECTRONIC CONTRIBUTION TO THE	THERMODYNAMICS OF MOLTEN UO(2)*	148845 P 33
#THE ELECTRONIC CONTRIBUTION TO THE	THERMODYNAMICS OF UO2.*	143773 P 33
OF A PWR'S CORE (IN FRENCH)* #FLOW AND HEAT TRANSFER	THERMOHYDRAULIC MODELISATION DURING THE REFLOODING PHASE	143928 P 1
FAILURE RATE OF COMPONENTS AND THE APPARENT INFLUENCE OF	THESE OCCURRENCES IN FAULT TREE ANALYSIS (IN GERMAN)* THE	147779 P 24
#PROPAGATION OF SIMULATED SONIC EMISSION-IMPULSES IN	THICK WALLED STRUCTURES (IN GERMAN)*	144198 P 10
BLOWDOWN OF A PRESSURIZED WATER #EFFICIENT COMPUTATION OF	THREE-DIMENSIONAL FLUID-STRUCTURE INTERACTIONS DURING	146804 P 20
STUDY OF THE RANGE OF VELOCITY OF GAS INSIDE A BUBLE RISING	THROUGH A LIQUID CODE CABU I (IN FRENCH)* #	143210 P 2
GASEOUS MIXTURES (IN ENGLISH)* #FLAME PROPAGATION	THROUGH UNCONFINED AND CONFINED HEMISPERICAL STRATIFIED	143078 P 8
#DISTURBANCE AND ERROR RECONSTRUCTION WITH HELP OF	TIME-OF-FLIGHT DATA, FINAL REPORT (IN GERMAN)*	1483[illegible] P 13
STUDIES* #	TIRICH 4 -A COMPUTER PROGRAM FOR USE IN NUCLEAR SAFETY	147602 P 33
OF A PROGRESSIVE DISTORSION PROBLEM (TENSION AND CYCLE	TORSION) (IN FRENCH)* #SHORT ANALYSIS	146871 P 8
MVS (WITH #REPORT ON THE CONVERSION OF THE CASL-CODE	TRAC-P1 (VERSION 16.3) TO IBM STANDARD OPERATING SYSTEM	143966 P 14

SYSTEM USING COMBINATION OF MEDIA FOR THE INTENSIVE	TRAINING AND INSTRUCTION OF OPERATING PERSONNEL AT NUCLEAR	143768 P 21
SYSTEM BLOWDOWN (IN #EXPERIMENTAL DETERMINATION OF THE HEAT	TRANSFER COEFFICIENT IN THE CONTAINMENT DURING A COOLING	148297 P 10
REFLOODING PHASE OF A PWR'S CORE (IN FRENCH)##FLOW AND HEAT	TRANSFER THERMOHYDRAULIC MODELISATION DURING THE	143926 P 1
& ENGLISH)# #FLOW REDUCTION	TRANSIENT BURNOUT IN AN ANNULAR TEST SECTION (IN JAPANESE	148844 P 29
IN #STUDY ON THE CONDENSATION OF AIR AND STEAM MIXTURES, IN	TRANSIENT CONDITIONS, ON A STAINLESS STEEL TEST SECTION (	149900 P 7
COOLANT BLOWDOWN OF A WATER- #INVESTIGATION OF THE PRESSURE	TRANSIENT IN A MULTICOMPARTMENTED CONTAINMENT FROM THE	145873 P 10
COOLING SYSTEM BLOWDOWN ACCIDENT AND THE SUBSEQUENT THERMAL	TRANSIENT IN LIGHT-WATER AND HIGH TEMPERATURE REACTORS (IN	145874 P 12
RESEARCH ON SINGLE AND MULTIPLE TUBE ARRAYS IN THE PRESSURE	TRANSIENT OF A NUCLEAR POWER PLANT IN THE LARGE EXPERIMENT	146488 P 14
#THE STATUS OF RELAP-UK MK III AT JULY 1976- A PROGRAM FOR	TRANSIENT THERMAL-HYDRAULIC ANALYSIS#	145640 P 21
GERMAN)# #DEVELOPMENT OF A MASS-DENSITY METHOD FOR	TRANSIENT TWO PHASE STATE USING ATOMIC RESONANCE (IN	144197 P 25
FLOW - A COMPUTER CODE FOR THE DETERMINATION OF PRESSURE	TRANSIENTS IN FULL-PRESSURE CONTAINMENTS OF WATER-COOLED	145626 P 16
UO(2)? A REVIEW OF THE ARGUMENTS# #DC ELECTRONIC	TRANSITIONS CONTRIBUTE TO THE THERMODYNAMICS OF CONDENSED	148846 P 24
FOR PURPOSES OF STUDYING THE ENSUING DECONTAMINATION AND	TRANSPORT (IN GERMAN)# PLANT WITH A PWR FOLLOWING A LOCA	146755 P 29
AND THE APPARENT INFLUENCE OF THESE OCCURRENCES IN FAULT	TREE ANALYSIS (IN GERMAN)# THE FAILURE RATE OF COMPONENTS	147779 P 24
#FAULT	TREE ANALYSIS WITH MULTISTATE COMPONENTS (IN ENGLISH)#	147860 P 20
#PATREC, A COMPUTER CODE FOR FAULT	TREE CALCULATIONS (IN ENGLISH)#	143778 P 4
PLANT IN THE #EXPERIMENTAL RESEARCH ON SINGLE AND MULTIPLE	TUBE ARRAYS IN THE PRESSURE TRANSIENT OF A NUCLEAR POWER	146488 P 14
#INSTRUMENTATION SYSTEM FOR THE DAS-MULTIPLE	TUBE RESEARCH PROGRAM (IN GERMAN)#	147796 P 13
#DATA REDUCTION OF THE FIRST TEST SERIES OF BLOWDOWN ON A	TUBULAR TEST SECTION ON OMEGA LOOP (IN FRENCH)#	143755 P 3
INTERNAL CORROSION ON THE MECHANICAL PROPERTIES OF ZRY-4	TURBINE (IN ENGLISH)# OXIDE FUEL) INFLUENCE OF THE	146673 P 20
IN FRENCH)# #CALCULATIONAL METHOD FOR	TWO DIMENSIONAL VISCOUS FLOW USING FINITE ELEMENT METHOD (	143753 P 2
#CRITICAL FLOW AND FLOW BLOCKAGE PHENOMENON FOR A	TWO PHASE FLOW (IN FRENCH)#	143911 P 1
#DEVELOPMENT OF A MASS-DENSITY METHOD FOR TRANSIENT	TWO PHASE STATE USING ATOMIC RESONANCE (IN GERMAN)#	144197 P 25
IN GERMAN)# #COMPARISON BETWEEN A ONE- AND	TWO-DIMENSIONAL CALCULATION OF A WATER-VAPOR NOZZLE FLOW (	143211 P 19
CONDENSATION CHAMBER OF A REACTOR #COMPUTER MODEL FOR THE	TWO-DIMENSIONAL CALCULATION OF THE WATER FLUID-SWELL IN THE	145757 P 15
#FLUST-2D - A COMPUTERCODE FOR THE CALCULATION OF THE	TWO-DIMENSIONAL FLOW OF A COMPRESSIBLE MEDIUM IN COUPLED	146750 P 26
#EXPERIMENTAL STUDY OF CRITICAL	TWO-PHASE FLOW (IN ENGLISH)#	143900 P 7
CRITICAL DISCHARGE MODEL APPLIED TO MULTI-COMPONENT,	TWO-PHASE SYSTEMS - THE COMPUTER PROGRAMS CRITS AND CRITTER	146483 P 13

HYDRAULIC ANALYSIS# #THE STATUS OF RELAP-	UK MK III AT JULY 1976- A PROGRAM FOR TRANSIENT THERMAL-	145640 P 21
#DEVELOPMENT OF THE BUBBLE RISE MODEL IN RELAP-	UK#	145641 P 21
OF INTERPRETATIONAL MODELS FOR THE ESTIMATION OF FLOW USING	ULTRASONIC TESTS ON NUCLEAR REACTORS (IN GERMAN)#	146486 P 13
APPARENT INFLUENCE OF THESE OCCURRENCES IN FAULT TREE	# UNCERTAINTY OF THE FAILURE RATE OF COMPONENTS AND THE	147779 P 24
MIXTURES (IN ENGLISH)# #FLAME PROPAGATION THROUGH	UNCONFINED AND CONFINED HEMISPHERICAL STRATIFIED GASEOUS	143078 P 2
TEST DURATION, TEST EFFICIENCY NOT 100%, #SYSTEMS WHICH ARE	UNCONNECTED AND WAITING FOR PERIODIC TESTING-NONNEGLIGIBLE	145296 P 4
EXPERIMENTAL INVESTIGATIONS ON THE LWR FUEL ROD BEHAVIOR	UNDER ACCIDENT CONDITIONS (IN GERMAN)# THE THEORETICAL AND	143740 P 19
#EXPERIMENTAL TESTS ON BUCKLING OF ELLIPSOIDAL VESSEL HEADS	UNDER INTERNAL PRESSURE (IN FRENCH)#	146870 P 8
	# UNIDIMENSIONAL MODELS FOR DIPHASE FLOWS (IN FRENCH)#	143936 P 1
THE ELECTRONIC CONTRIBUTION TO THE THERMODYNAMICS OF MOLTEN	UO(2)# #	148845 P 23
TRANSITIONS CONTRIBUTE TO THE THERMODYNAMICS OF CONDENSED	UO(2)? A REVIEW OF THE ARGUMENTS# #DC ELECTRONIC	148846 P 24
#THE ELECTRONIC CONTRIBUTION TO THE THERMODYNAMICS OF	UO2.#	143773 P 23
#TIPSON 4 -A COMPUTER PROGRAM FOR	USE IN NUCLEAR SAFETY STUDIES#	147662 P 23
ENGLISH)#	# USE OF MARKOV PROCESSES FOR RELIABILITY PROBLEMS (IN	143777 P 3
NUCLEAR INSTALLATIONS (IN ENGLISH)#	# USE OF PROBABILISTIC METHODS IN THE SAFETY EVALUATION OF	147021 P 5
MEASUREMENTS IN FRACTURE MECHANICS RESEARCH ON MATERIALS	USED IN FAST SODIUM-COOLED REACTORS (IN GERMAN)# EMISSION	143805 P 26
OF A MASS-DENSITY METHOD FOR TRANSIENT TWO PHASE STATE	USING ATOMIC RESONANCE (IN GERMAN)# #DEVELOPMENT	144197 P 25
#DEVELOPMENT AND SYNTHESIS OF AN EDUCATIONAL SYSTEM	USING COMBINATION OF MEDIA FOR THE INTENSIVE TRAINING AND	143768 P 21
#CALCULATIONAL METHOD FOR TWO DIMENSIONAL VISCOUS FLOW	USING FINITE ELEMENT METHOD (IN FRENCH)#	143753 P 2
OF RISK FOLLOWING THE INHALATION OF PLUTONIUM OXIDE	USING OBSERVED LUNG CLEARANCE PATTERNS# #ASSESSMENTS	145281 P 25
AND #PRELIMINARY EXPERIMENT ON LEAKAGE MONITORING	USING SONIC EMISSION ANALYSIS: EXPANDED INSTRUMENTATION	145872 P 10
#RESULTS OF THE LOFT EXPERIMENT L1-4: POST RUN CALCULATIONS	USING THE COMPUTER CODE "DRUFAN" (IN GERMAN)#	148668 P 16
OF INTERPRETATIONAL MODELS FOR THE ESTIMATION OF FLOW	USING ULTRASONIC TESTS ON NUCLEAR REACTORS (IN GERMAN)#	146486 P 13
REPORTS FROM THE REACTOR SAFETY RESEARCH PROGRAMS OF BMFT,	USNRC, EPRI AND JSTA, JULY 1-SEPTEMBER 30, 1978 (IN GERMAN	144854 P 16
REPORTS FROM THE REACTOR SAFETY RESEARCH PROGRAMS OF BMFT,	USNRC, EPRI, AND JSTA, REPORT PERIOD OCTOBER 1 - DECEMBER	146813 P 17

INFLUENTIAL PARAMETERS# #MODEL OF THE FAILURE RATES OF THE	VALVES OF ST. LAURENT LES EAUX POWER PLANT ACCORDING TO	143339 P 6
BETWEEN A ONE- AND TWO-DIMENSIONAL CALCULATION OF A WATER-	VAPOR NOZZLE FLOW (IN GERMAN)# #COMPARISON	143211 P 19
#LNG/WATER	VAPOUR EXPLOSIONS - ESTIMATES OF PRESSURES AND YIELDS#	148843 P 23
STEAM TABLES HAVING PRESSURE AND DENSITY AS THE INDEPENDENT	VARIABLES (IN ENGLISH & JAPANESE)# FOR GENERATING	144527 P 26
MELTDOWN ACCIDENT: CALCULATION TO IDENTIFY THE INFLUENCE OF	VARIOUS PARAMETERS (IN GERMAN)# OF THE HYPOTHETICAL CORE	143379 P 15
FAILURE OF THE CORE SUPPORT STRUCTURE #INVESTIGATION OF THE	VARIOUS PHASES OF THE CORE MELT ACCIDENT AFTER THE AFTER	145736 P 25
HYDRODYNAMIC ANALYSIS (IN FRENCH)#STUDY OF THE STABILITY OF	VARIOUS SYSTEMS AND DESCRIPTION OF EQUATIONS FOR	143758 P 2
IMPACT OF FRAGMENTS AND PROJECTILES OF DIFFERENT MASS AND	VELOCITY (IN GERMAN)# MATERIALS AND COMPONENT PARTS AT	143603 P 13
CODE CABU 1 (IN FRENCH)# #STUDY OF THE RANGE OF	VELOCITY OF GAS INSIDE A BUBLE RISING THROUGH A LIQUID	143210 P 2
FORTRAN #REPORT ON THE CONVERSION OF THE CASE-CODE TRAC-P1 (	VERSION 16.3) TO IBM STANDARD OPERATING SYSTEM MVS (WITH	143966 P 14
#FRACTURE PROBABILITY EVALUATION OF A LWR PRESSURE	VESSEL (IN ENGLISH)#	143876 P 5
#A PROBABILISTIC STUDY OF	VESSEL BURST IN LIGHT WATER NSSS (IN FRENCH)#	146865 P 8
STRUCTURE DUE TO THE FORMATION OF MELT OR DUE TO PRESSURE	VESSEL FAILURE, PART II (IN GERMAN)# OF THE CORE SUPPORT	145736 P 25
#EXPERIMENTAL TESTS ON BUCKLING OF ELLIPSOIDAL	VESSEL HEADS UNDER INTERNAL PRESSURE (IN FRENCH)#	146870 P 8
FOR DEFECT DETECTION AND STRAIN EVALUATION IN PRESSURE	VESSELS##LASER HOLOGRAPHIC AND SPECKLE PHOTOGRAPHY METHODS	120343 P 31
#CALCULATIONAL METHOD FOR TWO DIMENSIONAL	VISCOUS FLOW USING FINITE ELEMENT METHOD (IN FRENCH)#	143753 P 2
SEVERAL MODELS (#MODEL FOR THE CALCULATION OF THE RATE OF	VOIDING DURING A RAPID FAILURE (BLOWDOWN) COMPARISON OF	143751 P 2

TEST EFFICIENCY NOT #SYSTEMS WHICH ARE UNCONNECTED AND	WAITING FOR PERIODIC TESTING-NONNEGLIGIBLE TEST DURATION,	145296 P 4
#PROPAGATION OF SIMULATED SONIC EMISSION-IMPULSES IN THICK	WALLED STRUCTURES (IN GERMAN)#	144196 P 16
#STUDY OF POLLUTANT DISPERSION IN	WATER AND AIR (IN ENGLISH)#	146801 P 8
ACCIDENT AND THE SUBSEQUENT THERMAL TRANSIENT IN LIGHT-	WATER AND HIGH TEMPERATURE REACTORS (IN GERMAN)# BLOWDOWN	145874 P 12
#EXPERIMENT OF THE DOWNCOMER EFFECTIVE	WATER HEAT DURING A REFLOOD PHASE OF PWR (1) A (IN JAPANESE)	143893 P 21

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

INTERNAL DISTRIBUTION

1. Seymour Baron
2. J. R. Buchanan
3. C. A. Burchsted
4. W. R. Casto
5. R. O. Chester
6. W. B. Cottrell
7. A. L. Crawford
8. C. E. Davis
9. R. B. Gallaher
10. C. T. Garten
11. E. W. Hagen
12. P. M. Haas
13. F. A. Heddleson
14. M. Heiskell
15. R. D. Hurt
16. W. H. Jordan
17. Milton Levenson
18. G. T. Mays
19. F. R. Mynatt
20. D. S. Queener
21. J. L. Rich
22. R. L. Scott
23. S. D. Swisher
24. H. E. Trammell
25. D. B. Trauger
26. M. L. Winton

27-28. Central Research Library

29. Y-12 Document Reference Section (DRS)

30. Laboratory Records Department Record Copy (LRD-RC)

31-47. Nuclear Safety Information Center (NSIC)

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)

☆ U. S. GOVERNMENT PRINTING OFFICE: 1979-640-245-291

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 *Nuclear Safety*, 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 *Nuclear Safety*, 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 *Nuclear Safety*, 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.