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NUREG/CR-0487
ORNL/NUREG/NSIC-154

BIBLIOGRAPHY OF MICROFICHED FOREIGN REPORTS DISTRIBUTED under the NRC REACTOR SAFETY RESEARCH FOREIGN TECHNICAL EXCHANGE PROGRAM, 1975–1977

Debbie S. Queener
Wm. B. Cottrell

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

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NSIC

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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
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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)

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BIBLIOGRAPHY OF MICROFICHED FOREIGN REPORTS DISTRIBUTED UNDER THE NRC REACTOR SAFETY RESEARCH FOREIGN TECHNICAL EXCHANGE PROGRAM, 1975—1977

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Engineering Technology Division

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FOREWORD

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

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:

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P.O. Box Y
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830

Telephone 615-483-8611, Ext. 3-7253
FTS number is 850-7253

Page(s) Missing

from

Original Document

by the Nuclear Safety Information Center reports which tabulated the documents received during those intervals.

When this exchange program was first initiated, the Department of Energy's Technical Information Center (TIC) was serving as NRC's document agent. At that time the distribution of documents was very limited, since few copies were received and were not being microfiched. In mid-1976 the document processing was transferred to the Nuclear Safety Information Center (NSIC), and the distribution was expanded by distributing the reports in microfiche form. The above changes had consequences which impacted on the documentation as well as on the availability of many of the reports:

1. Only a few of the early reports (those received prior to mid-1976) were microfiched. In general, the reports which were not microfiched are in very limited supply. (Persons interested in these reports can arrange to see a copy at the NRC offices in Silver Springs, Md.)

2. TIC established a numbering system for the reports (e.g., FRRSR-xx for French reports), but these numbers were not initially included on the microfiche headers. When NSIC took over the processing and distribution of the reports, the numbering system established by TIC was continued. Current practice is to include both the NSIC-assigned number and the number assigned by the originating organization on the microfiche.

In this document, the reports are listed in a computerized bibliography first by country, then alphabetically by organization, and finally chronologically by report date. Following the bibliography are keyword, author, and permuted-title indexes. All reports listed in this document are available on microfiche from the National Technical Information Service, U.S. Department of Commerce, 5285 Port Royal Road, Springfield, VA 22161. In addition, the hard copy received from the originating organization may be examined by special arrangements either at the NRC offices in Silver Springs, Md., or the NSIC offices in Oak Ridge, Tenn. These microfiche are also available from the microfiche contractor, NCR Corporation, Valleywood Data Processing Center, 3100 Valleywood Drive, Dayton, OH 45429; however, there is a minimum charge of $50.00 regardless of the number of microfiche ordered.

ORGANIZATION OF BIBLIOGRAPHY

The bibliography which follows contains only those light-water reactor safety research reports received by the NRC from 1975 through 1977 which were subsequently microfiched and distributed. The reports are listed first by country, then alphabetically by the originating organization, and finally chronologically by report date under each organization. The bibliography is sorted into the following categories:

1. French Reactor Safety Research Reports.
2. German (FRG) Reactor Safety Research Reports.
3. Japanese Reactor Safety Research Reports.

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576–600	16.50
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[c]Add $2.50 for each additional 100-page increment over 600 pages.

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

PARTS AND METHOD OF INDEXING ABSTRACTS

DOCUMENT TITLE

AUTHOR - AUTHORS ARE LISTED ALPHABETICALLY IN THE *AUTHOR INDEX* ALONG WITH NSIC NUMBER

ACCESSION NUMBER - SERIAL NUMBER ASSIGNED TO THE DOCUMENT BY NSIC

BIBLIOGRAPHIC DATA

ACRS ACCESSION NUMBER

PUBLICATION DATE

21408
ACRS-00023
RABIN SA + ATRAZ BG + BADER MB + BUSBOOM HJ + HAZEL, VE
EXAMINATION AND EVALUATION OF RUPTURE IN EVESR SUPERHEAT FUEL ROD WITH 0.012-INCH-THICK INCOLOY-800 CLADDING
GENERAL ELECTRIC CO., SUNNYVALE, CALIF. ADVANCED PRODUCTS OPERATION
GEAP-5416 +. 72 PAGES, FIGURES, TABLES, REFERENCES, JANUARY 1967

PREMATURE FAILURES WERE OBSERVED IN ONE 0.008-IN.-WALL ROD AND IN ONE 0.012-IN.-WALL ROD OF THE MARK-III EXPERIMENTAL FUEL. CLADDING IS INCOLOY-800. FAILURES WERE IN REGION OF MAXIMUM POWER IN PEAK-POWER ROD. MOST REASONABLE EXPLANATION IS LOW-CYCLE FATIGUE, ACCELERATED BY HIGHER-THAN-DESIGN CLADDING TEMPERATURES. NO SIGNS OF RAPID CORROSION, SUCH AS HAVE BEEN ASSOCIATED WITH TYPE-304 SS, WERE FOUND.

ABSTRACT OF ORIGINAL DOCUMENT

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

*FAILURE, FUEL ELEMENT + CLAD + FAILURE, FATIGUE + INCONEL + REACTOR, INTERNAL SUPERHEAT + THERMAL MECHANICAL EFFECT + VESR (ISR)

DOCUMENT AVAILABILITY AND PRICE

CORPORATE AUTHOR

REPORT NUMBER

KEY WORDS - LISTED ALPHABETICALLY IN THE *KEYWORD INDEX* WITH THE ACCESSION AND PAGE NUMBER

BIBLIOGRAPHY

1. FRENCH SAFETY RESEARCH REPORTS

THE FOLLOWING IS A LISTING OF MICROFICHED LIGHT-WATER REACTOR SAFETY RESEARCH REPORTS RECEIVED FROM FRANCE DURING THE PERIOD 1975-1977 UNDER THE TECHNICAL EXCHANGE AGREEMENT.

134579
GUIZOUARN L
TRANSITORY CONDENSATION OF WATER VAPOR IN PRESENCE OF AIR ON A PLANE WALL - DESCRIPTION OF THE FACILITY (IN FRENCH)
CEA CENTRE D'ETUDES NUCLEAIRES DE GRENOBLE, FRANCE
TT 508 + FRRSR-12 +. 23 PPS, 11 FIGS, SEPT. 1975

THE PURPOSES OF THE EXPERIMENT ARE PRESENTED AND A DESCRIPTION OF THE FACILITY IN WHICH THE CONDENSATION EXPERIMENTS WERE CARRIED OUT IS PROVIDED. VARIOUS GRAPHS AND PLOTS OF DATA COLLECTED IS ALSO PRESENTED.

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

*WATER VAPOR + *CONDENSATION + EXPERIMENT + SYSTEM DESCRIPTION + DATA COLLECTION + FOREIGN EXCHANGE

112600
RIEGEL B
BERTHA CODE APPLICATIONS TO EXPERIENCE EDWARDS (IN FRENCH)
SERVICE DES TRANSFERTS THERMIQUES, FRANCE
REPORT NOTE TT 510 +. 13 PPS, 6 FIGS, 8 REFS, SEPT. 1975

THIS REPORT COMPARES THE RESULTS OF BERTHA WITH THE DEPRESSURIZATION EXPERIMENT OF UKAEA 'HEALTH AND SAFETY BRANCH' MADE TO AWRE, FOULNESS, BY A. R. EDWARDS AND T. P. O'BRIEN. THIS EXPERIMENT IS ONE OF THE STANDARD EXPERIMENTS CHOSEN BY CSNI (OECD).

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

ACCIDENT ANALYSIS + *COMPUTER PROGRAM + ACCIDENT, LOSS OF PRESSURE + *COMPARISON, THEORY AND EXPERIENCE + OECD + EXPERIMENT + UKAEA

136659
GUIZOUARN L + PINET B + BARRIERE G
EXPERIMENTAL STUDY OF CRITICAL FLOWS IN A DIPLASTIC (WATER VAPOR) LEAK IN A LOW AMOUNT IN A CHANNEL WITH 7 DEGREES DIVERGENCE AT 2-7 BARS PRESSURE AT THE NECK (IN FRENCH)
CEA CENTRE D'ETUDES NUCLEAIRES DE GRENOBLE, FRANCE
TT 501 + FRRSR-14 +. 78 PPS, 25 FIGS, NO DATE

DEFINITIONS OF TWO PHASE CRITICAL FLOW ARE PROVIDED AND THE MAIN CHARACTERISTICS OF THE FACILITY ARE DESCRIBED. DATA OBTAINED AT 2, 3, 5 AND 7 BARS ARE GIVEN IN TABULAR AND GRAPHICAL FORMS. THE DATA ARE DISCUSSED AT THE END OF THE PAPER.

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

*FLOW, TWO PHASE + *FLOW, CRITICAL + WATER VAPOR + LEAK + EXPERIMENT + DATA COLLECTION + FOREIGN EXCHANGE

113034
DERUAZ R + CLEMENT P + LAMBERT M + PIC P
EMERGENCY COOLING FOR WATER REACTORS - REFLOODING (IN FRENCH)
COMMISSARIAT A L'ENERGIE ATOMIQUE, FRANCE
REPORT TT NO. 509 +. 54 PPS, 23 FIGS, 6 REFS, OCT. 1975

THE EMERGENCY COOLING BY REFLOODING OF PRESSURIZED WATER REACTORS IS STUDIED ON A BUNDLE OF 25 THERMAL SIMULATORS OF FUEL RODS. TESTS ARE DESCRIBED AND SOME RESULTS GIVEN ON VARIOUS INFLUENCING PARAMETERS (HEAT FLUX, FLOW RATES, PRESSURE...) ON MAXIMUM TEMPERATURES REACHED AND THE DURATION OF TEMPERATURE TRANSIENT. EACH METHOD USED FOR HEAT TRANSFER COEFFICIENT DETERMINATION IS PRESENTED.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

FRANCE + HYDRAULIC EXPERIMENT + REACTOR, PWR + EMERGENCY COOLING + CORE REFLOODING SYSTEM + HEAT TRANSFER COEFFICIENT + THERMAL HYDRAULIC ANALYSIS + HEATERS

112601
BLIN A
PRESENTATION AND USES OF PATREC - THE CODE (IN FRENCH)
COMMISSARIAT A L'ENERGIE ATOMIQUE, FRANCE
REPORT SETS NO. 42 +. 22 PGS, FIGS, 4 REFS, JAN. 1976

THE CODE CALCULATES THE RELIABILITY OF COMPLEX SYSTEMS UNDER FAULT TREE SHAPES. SPECIFICATIONS FOR USING THE CODES (INPUT DATA, NECESSARY 'VOLUME,' RUNNING TIME, ETC.) ARE ALSO GIVEN.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TN. 37830

COMPUTER PROGRAM + RELIABILITY ANALYSIS + FAULT TREE ANALYSIS

112629
TARTU H + SONNET A
GENERAL REPORT OF PHYSICAL TESTS ON MARVIKEN 1 (IN FRENCH)
COMMISSARIAT A L'ENERGIE ATOMIQUE, FRANCE
REPORT SETS NO. 39 +. 100 PPS. FIGS. REFS. DEC. 1975

PRESENTS THE ANALYSES OF MEASUREMENTS OBTAINED FROM 16 TESTS ON THE MARVIKEN (SWEDISH) REACTOR CONTAINMENT STRUCTURES. DISCUSSES THE STRUCTURE RESPONSE TO PRESSURE AND TEMPERATURE VARIATION DUE TO A BWR TYPE PRIMARY PIPE RUPTURE.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

REACTOR, BWR + CONTAINMENT + *CONTAINMENT, PRESSURE SUPPRESSION + TESTING + PRESSURE TRANSIENT + THERMAL TRANSIENT + PERFORMANCE + MARVIKEN (BHWR)

117924
MENESSIER D + FORGE A + PEDEL A
DANAIDES PROGRAM DE CACUL DE TRANSITOIRE DE DECOMPRESSION (IN FRENCH)
CEA DEPARTMENT DE SURETE NUCLEAIRE, FRANCE
SETS NO. 38 +. 80 PPS. FIGS. REFS. OCT. 1975

DANAIDES DESCRIBES PRIMARY CIRCUIT THERMODYNAMIC BEHAVIOR DURING DEPRESSURIZATION IN A PWR OR IN AN EXPERIMENTAL LOOP. THE PHYSICAL ASSUMPTIONS AND MATHEMATICAL METHODS CHOSEN TO SOLVE THE EQUATIONS ARE ESPECIALLY APPLICABLE TO FAST DEPRESSURIZATION.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

*COMPUTER PROGRAM + ACCIDENT, LOSS OF COOLANT + *ACCIDENT, LOSS OF PRESSURE + REACTOR, PWR

135195
HOFFMANN A + JEANPIERRE F
CODE 'TRICO' - ANALYSIS OF THREE-DIMENSIONAL STRUCTURES COMPOSED OF HULLS AND BEAMS (IN FRENCH)
CEA SERVICE DES ETUDES MECANIQUES ET THERMIQUES, FRANCE
EMT/75-24 + FRRSR-22 +. 88 PPS. FIGS. APRIL 1975

THE TRICO PART OF CEA-SEMT SYSTEM PERFORMS ELASTIC OR PLASTIC CALCULATION OF STRUCTURES MADE OF THIN HULLS AND BEAMS USING THE FINITE ELEMENT METHOD. THE VARIOUS PROCEDURES AND FEATURES OF THE CODE WITH RESPECT TO INPUT OF DATA, FORMAT, AND TYPE OF OUTPUT ARE OUTLINED.

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

*COMPUTER PROGRAM + SYSTEM DESCRIPTION + PROCEDURES AND MANUALS + *STRUCTURAL INTEGRITY + ELASTICITY + PLASTICITY + ANALYTICAL TECHNIQUE + FOREIGN EXCHANGE

118382
BLIN A + CARNION A + MOLLARD C
EVALUATION OF THE EFFECT OF DEPENDENCES ON THE RELIABILITY OF A SAFETY SYSTEM WITH THE AID OF THE PAT-REC-DE CODE (IN FRENCH)
CEA DEPARTMENT DE SURETE NUCLEAIRE, FRANCE
DSN REPORT 70 +. 34 PPS. TABS. FIGS. APRIL 21, 1975

GIVES THE RESULTS OF AN ANALYSIS OF THE AVAILABILITY OF THE INTERMEDIATE COOLING SYSTEM OF A BOILING WATER REACTOR: THIS SYSTEM ENSURES AMONG OTHER FUNCTIONS THE LONG TERM COOLING OF PUMPS AND EXCHANGERS OF THE EMERGENCY CORE COOLING SYSTEM. THE ANALYSIS WAS LIMITED TO THE HYDRAULIC AND MECHANIC PARTS OF THE SYSTEM AND TOOK INTO ACCOUNT THE COMMON MODE EFFECT RESULTING FROM FUNCTIONAL DEPENDENCES. THE RESULTS CONFIRM THE GOOD PROTECTION OF THE SYSTEM AGAINST THE COMMON MODE EFFECT AND SHOW THE AVAILABILITY OF THE SYSTEM TO BE ABOUT 0.995.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

REACTOR, BWR + *RELIABILITY ANALYSIS + *COMPONENT COOLING SYSTEM + EMERGENCY COOLING SYSTEM + FAILURE, COMMON MODE + JACOBS

112602
ROUSSEAU JC + BOUDSOCQ G + MARECHAL A
BERTHA CODE OF APPLICATIONS TO CANON EXPERIMENTS (IN FRENCH)
COMMISSARIAT A L'ENERGIE ATOMIQUE, FRANCE
REPORT TT-491 +. 33 PPS. FIGS. MARCH 1975

WITH CERTAIN HYPOTHESIS BERTHA CORRECTLY REPRESENTS PRESSURE AND TEMPERATURE EVALUATIONS IN THE CANON EXPERIMENTS. SINCE NO VOID FRACTION MEASUREMENTS ARE MADE, THIS ESSENTIAL PARAMETER IS NOT ADJUSTED. HOWEVER, VOID FRACTION MEASUREMENTS BY NEUTROGRAPHY ARE FORESEEN WITH CANON IN THE NEAR FUTURE AND A BETTER UNDERSTANDING OF DEPRESSURIZATION PHENOMENA WILL BE POSSIBLE.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 52, OAK RIDGE, TENN. 37830

112602 *CONTINUED*
*COMPUTER PROGRAM + ACCIDENT, LOSS OF PRESSURE + SPECTROMETRY, NEUTRON + EXPERIMENT + VOID FRACTION + PRESSURE TRANSIENT + THERMAL TRANSIENT

125000
DOURY A + GERALD R + PICOL M
ABACI FOR THE DIRECT EVALUATION OF ATMOSPHERIC TRANSPORT OF GASEOUS EFFLUENTS (IN FRENCH)
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
FRRSR-34 + DSN-84 +. 53 PPS, FIGS, REFS, JAN. 1976

A QUICK QUANTITATIVE PREDICTION OF ATMOSPHERIC POLLUTIONS EMITTED BY REAL OR VIRTUAL POINT SOURCES OF VARIABLE HEIGHT UNDER VARIOUS METEOROLOGICAL CONDITIONS CONSIDERED AS STATIONARY CAN BE MADE, USING THE ABACI PRESENTED IN THIS REPORT.

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

JACOBS + FRANCE + GAS + EFFLUENT + TRANSPORT + METEOROLOGY + *NUMERICAL METHOD + SOURCE, POINT + *ATMOSPHERIC DIFFUSION

116677
SIGNORET JP
APPLICATION OF RELIABILITY ANALYSIS METHODS TO THE COMPARISON OF TWO SAFETY CIRCUITS (IN FRENCH)
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
DSN 85 +. 33 PPS, 17 FIGS, JUNE 1975

TWO CIRCUITS OF DIFFERENT CONCEPTION DESIGNED TO ENSURE THE LOW PRESSURE SAFETY INJECTION FOR PWRS ARE ANALYSED USING RELIABILITY METHODS. FAILURE TREES ARE ESTABLISHED AND FAILURE PROBABILITIES ARE EVALUATED. THE TESTS AND MAINTENANCE EFFECTS ON THE RESULTS ARE SHOWN, AS WELL AS CRITICAL PATHS. THE NUMBER OF RESULTS THUS OBTAINED MAKE IT POSSIBLE TO MAKE A CHOICE ACCORDING TO THE RELIABILITY LEVEL REQUIRED FOR THIS TYPE OF CIRCUIT.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

*RELIABILITY ANALYSIS + COMPARISON + RELIABILITY, SYSTEM + *SAFETY INJECTION + REACTOR, PWR + FAULT TREE ANALYSIS + PROBABILITY + FAILURE, COMPONENT + JACOBS

122546
SIGNORET JP
AVAILABILITY OF STANDBY SYSTEMS WHICH ARE PERIODICALLY TESTED (IN FRENCH)
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
FRRSR-36 + DSN 113 +. 50 PPS, 7 REFS, SEPT. 1976

PRESENTS CALCULATIONS CONCERNING THE INSTANTANEOUS AND STEADY STATE AVAILABILITY OF PERIODICALLY TESTED STANDBY SYSTEMS. ANALYTICAL FORMULAS ARE DEVELOPED BASED ON RELIABILITY CHARACTERISTICS FOR SINGLE COMPONENTS AND MORE COMPLEX SYSTEMS. PARAMETERS INCLUDED IN THE MODEL ARE CONSTANT FAILURE RATE, REPAIR RATE, PROBABILITY NOT TO START ON DEMAND, AND TEST INTERVAL. APPROXIMATIONS ARE GIVEN TO ALLOW HAND CALCULATIONS, AND DESCRIPTIONS OF COMPUTER PROGRAMS FOR THE MORE COMPLEX SYSTEMS ARE PRESENTED. THE INTENT OF THE STUDY IS TO ASSESS NUCLEAR SAFEGUARD SYSTEMS AVAILABILITY, BUT THE METHODOLOGY MAY BE APPLIED TO ANY INDUSTRIAL STANDBY SYSTEM.

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

RELIABILITY, SYSTEM + RELIABILITY, COMPONENT + *RELIABILITY ANALYSIS + *AVAILABILITY + ANALYTICAL MODEL + *EMERGENCY SYSTEM + TEST INTERVAL + MAINTENANCE AND REPAIR + JACOBS

122545
SIGNORET JP
OPTIMIZATION OF THE TEST INTERVAL ON THE AVAILABILITY OF STANDBY SYSTEMS - APPLICATIONS TO EMERGENCY DIESEL GENERATORS (IN FRENCH)
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
FRRSR-37 + DSN 114 +. 17 PPS, 6 FIGS, 1 REF, SEPT. 1976

AN IMPROVED MATHEMATICAL MODEL IS PRESENTED FOR EVALUATING THE SCHEDULED INSTANTANEOUS AVAILABILITY, THE STEADY STATE AVAILABILITY AND OPTIMUM TEST FREQUENCY FOR DIESEL GENERATORS. RELIABILITY PARAMETERS USED IN THE MODEL INCLUDE STANDBY FAILURE RATE, REPAIR RATE, PROBABILITY NOT TO START ON DEMAND, AND TEST INTERVAL. THE EFFECTS THAT INFLUENCE ONE DIESEL GENERATOR WHILE TESTING THE SECOND ONE ARE ALSO EVALUATED. THE INTENT OF THE ANALYSIS IS TO OPTIMIZE THE TEST FREQUENCY IN TERMS OF THE RELIABILITY PARAMETERS OF THE DIESEL GENERATOR.

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

*GENERATOR, DIESEL + RELIABILITY ANALYSIS + *AVAILABILITY + *TEST INTERVAL + OPTIMIZATION + EMERGENCY POWER, ELECTRIC + ANALYTICAL MODEL + JACOBS

122575
DUFRESNE J
PROBABILISTIC STUDY ON THE RUPTURE OF LIGHT WATER REACTOR PRESSURE VESSEL (IN FRENCH)
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
FRRSR-38 + DSN 116F +. 120 PPS. 17 FIGS. 6 REFS. OCT. 1976

EVALUATION IS BASED ON USE OF THE FRACTURE MECHANICS THEORY IN A PROBABILISTIC FORM; EVALUATIONS FROM STATISTICAL STUDIES ON CONVENTIONAL VESSELS HAVE BEEN ANALYSED. IT IS CONCLUDED THAT THE NUMBER OF VESSELS REPRESENTATIVE OF NSSS IS PRESENTLY INSUFFICIENT FOR RUPTURE PROBABILITY VALUES TO BE DEDUCED. THE REFERENCE PRESSURE VESSEL IS A PWR 900 MWE, 3 - LOOP PRESSURE VESSEL HAVING THE CHARACTERISTICS ESTABLISHED BY WESTINGHOUSE. THE GENERAL METHOD OF CALCULATION IS BEING DEVELOPED. IT IS BASED ON THE USE OF HISTOGRAMS AS INPUT DATA. COVAL 1 CODE IS PRESENTLY IN USE.

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

FAILURE + PRESSURE VESSELS + REACTOR, LWR + STRESS ANALYSIS + ACCIDENT, PROBABILITY OF + REACTOR, PWR + JACOBS

125263
MONNIER P + TOUFFAIT AM + SAGLIO R
COMPARISON OF VARIOUS FLAW SIZING METHODS IN THE ULTRASONIC TESTING OF HEAVY SECTION WELDED COMPONENTS (IN FRENCH)
CEA DIVISION DE METALLURGIE, FRANCE
FRRSR-43 + N.T. 201 +. 34 PPS. 16 FIGS. 3 REFS. FEB. 1976

PRESENTS VARIOUS SIZING METHODS USED ON A HEAVY SECTION WELDED JOINT, HAVING PURPOSELY INDUCED FLAWS. FOLLOWING TESTING METHODS WERE USED: FLAT PROBES (CONTACT AND IMMERSION TESTING), FOCUSSED PROBES (IMMERSION TESTING), AND ACOUSTICAL HOLOGRAPHY. COMPARISON BETWEEN ULTRASONICALLY MEASURED SIZES AND REAL SIZES OF THE FLAWS WAS MADE BY CUTTING UP THE WELDMENTS. ONLY FOCUSSED PROBES ALLOW PROPER FLAW SIZE DETERMINATION.

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

JACOBS + FLAW + ULTRASONICS + FRANCE + HSST + WELDS

124999
CHENEBAULT P + KURKA G + VIVIER M + LASNE G
FISSION PRODUCT RELEASE FROM A DEFECTED LWR FUEL ROD AT DIFFERENT POWER LEVELS
CEA CENTRE D'ETUDES NUCLEAIRES DE GRENOBLE, FRANCE
FRRSR-45 + DMG 123/76 +. 37 PPS. 19 FIGS. 4 REFS. DEC. 1976

THE RESULTS OF MEASUREMENTS LEAD TO THE FOLLOWING CONCLUSIONS: THE TRANSIT TIME OF THE VOLATILE FISSION PRODUCTS BETWEEN THE FUEL AND THE COOLANT IS SHORTER AT 200 W/CM THAN AT HIGHER POWER LEVELS; THE FISSION PRODUCTS RELEASE IS AFFECTED BY EACH POWER, FLOW OR PRESSURE VARIATIONS WHICH MAY OCCUR IN THE PRIMARY COOLANT CIRCUIT; IODINE HAS BEEN OBSERVED TO GIVE RISE TO 'BURST' RELEASES AT STEADY POWER (330 W/CM); THE NUMBERS OF ATOMS OF THE VOLATILE ISOTOPES (RARE GASES AND IODINES) EMITTED IN THE WATER HAVE BEEN DETERMINED AGAINST POWER LEVEL AND THE DEPENDENCE ON THE DECAY CONSTANT HAS BEEN ESTABLISHED.

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

SPECTROMETRY, GAMMA + REACTOR, LWR + XENON + IODINE + JACOBS + KRYPTON

122808
POCHARD R
ECCS EXPERIMENTS - 'EPIS I' RESULTS (IN FRENCH)
CEA CENTRE D'ETUDES NUCLEAIRES DE SACLAY, FRANCE
FRRSR-50 + SEEN/INF 76-014 +. 61 PPS. FIGS. REFS. SEPT. 22. 1976

A 1/11 SCALE MODEL WAS USED TO SIMULATE STEAM-WATER MIXING IN THE COLD LEG OF A PWR DURING A LOCA. AN AIR WATER MIXTURE SIMULATED THE ACTUAL STEAM-WATER MIXTURE. RESULTS ARE PRESENTED, ALONG WITH COMPARISON WITH A CALCULATIONAL MODEL.

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

REACTOR, PWR + OUT OF PILE EXPERIMENT + FLOW THEORY AND EXPERIMENTS + FLOW, TWO PHASE + FRANCE + ACCIDENT, LOSS OF COOLANT + JACOBS

137619
POCHARD R
PROGRAM OF STUDY ON THE MECHANICS AND THERMODYNAMICS OF INTERACTIONS BETWEEN THE PRIMARY COOLANT FLOW AND EMERGENCY COOLING WATER IN A PWR REACTOR (IN FRENCH)
CEA CENTRE D'ETUDES NUCLEAIRES DE SACLAY, FRANCE
S.E.E.N-RT.76.014 + FRRSR-51 +. APPROX. 150 PPS. FIGS. MARCH 1976

PRESENTS THE DATA IN FORMS OF GRAPHS AND TABULAR FORM COLLECTED DURING TESTING CONDUCTED IN A 1/11

137619 *CONTINUED*
SCALE MODEL TO SIMULATE STEAM-WATER MIXING IN THE COLD LEG OF A PWR. THE TESTS AND RESULTS ARE PRESENTED IN FRRSR-50.

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

*REACTOR, PWR + OUT OF PILE EXPERIMENT + *FLOW THEORY AND EXPERIMENTS + ACCIDENT, LOSS OF COOLANT + *DATA COLLECTION + DATA PROCESSING + STEAM + WATER + FLOW, MIXING + FOREIGN EXCHANGE

122566
BENEZECH G + ROUSSEAU L
STUDY OF THE TRANSPORT OF RADIOACTIVITY AND CONTAMINATION IN A PWR (IN FRENCH)
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
FRRSR-53 +. 31 PPS, 6 FIGS, 5 REFS, APRIL 1976

DISCUSSES THE CONTAINMENT INTEGRITY OF FUEL RODS AND THE PRIMARY COOLING SYSTEM AS BARRIERS FOR THE TRANSPORT OF RADIOACTIVITY IN A PWR. PRESENTS MEASUREMENTS DURING NORMAL OPERATION OF THE PRIMARY COOLING SYSTEM, AIR IN THE REACTOR BUILDING, AUXILIARY AND SECONDARY SYSTEMS, AND SPENT FUEL POOL. THE METHOD FOR CALCULATING THE TRANSPORT OF RADIOACTIVITY FROM FAILED FUEL RODS IS DESCRIBED.

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

REACTOR, PWR + *CONTAINMENT INTEGRITY + MAIN COOLING SYSTEM + *FISSION PRODUCT TRANSPORT + SPENT FUEL POOL + FAILURE, FUEL ELEMENT + FUEL ROD + JACOBS

124998
BENEZECH G + EYMERY D + LEFOL C + VENTRE R
MEASUREMENT OF THE TRANSPORT OF CONTAMINATION AT THE NUCLEAR CENTER OF ARDENNES FROM MAY 16 TO AUG. 10, 1974 (IN FRENCH)
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
FRRSR-54 + DSN/SESTR R-04 +. 34 PPS, TABS, MAY 5, 1975

SERIAL OF MEASUREMENTS WERE CARRIED OUT TO THE PWR OF CHOOZ WITH A VIEW TO MEASURE THE RADIOACTIVE PRODUCTS RELEASED FROM THE PRIMARY CIRCUIT TO THE AIR OF REACTOR SURROUNDING WALL, AUXILIARIES AND SECONDARIES CIRCUITS. THE MAIN PRODUCTS MEASURED WERE 3H, 133XE, 131I. FROM THESE RESULTS IT HAS BEEN FOUND AN ESTIMATION OF RELEASE RATE FROM THE PRIMARY CIRCUIT TO THE OTHERS. FOR IODINE, IT HAS BEEN FOUND THE RATE OF MOLECULAR FORM IN THE AIR AND THE HALF-LIFE OF TRAPPING ON THE WALLS OF THE REACTOR BUILDING.

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

DEPOSITION + FISSION PRODUCT RELEASE + RATE + REACTOR, PWR + TRITIUM + XENON + IODINE + FRANCE + R AND D PROGRAM + JACOBS

124291
PORCHERON J + MICHEL R
SIMULATION OF A FUEL HANDLING ACCIDENT IN A PWR - EXPERIMENT PIREE - HANDLING
CEA CENTRE D'ETUDES NUCLEAIRES DE CADARACHE, FRANCE
FRRSR-55 + VII-SFRP/34-01 +. 12 PPS, 3 FIGS, 5 REFS, (NO DATE)

THE FUEL UNLOADING OF A PWR REACTOR REQUIRES THE OPENING OF PRIMARY CIRCUIT. THE FUEL IS TRANSPORTED UNDER THE WATER OF THE STORAGE POOL. THE WATER BECOMES THE ONLY BARRIER AGAINST THE FISSION PRODUCTS IN THE CASE OF AN ACCIDENT. THE EXPERIMENT WAS MADE TO DETERMINE THE EFFICIENCY OF THIS BARRIER. GASEOUS, MOLECULAR IODINE 131 WAS INJECTED 3 METERS UNDER THE WATER. THE SYSTEM PROVIDES CAPABILITY TO INJECT THE CONTAMINATION IN A VERY SHORT TIME. MEASUREMENTS OF WATER AND AIR CONTAMINATION SHOW THAT THE PART OF IODINE PASSING THROUGH THE WATER IS WEAK (1/415). THE COMPLETE BREAKING OF THE FUEL ASSEMBLY WAS SIMULATED

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

REACTOR, PWR + *SIMULATION + *FUEL HANDLING + *ACCIDENT ANALYSIS + OUT OF PILE EXPERIMENT + IODINE + FISSION PRODUCT TRANSPORT + FRANCE + JACOBS

124992
MANET G + PORCHERON J + ROUSSEAU L
MEASUREMENT OF THE FISSION PRODUCTS IN THE PHEBUS LOOP (IN FRENCH)
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
FRRSR-56 + DSN/SES 13/75 +. 16 PPS, 2 FIGS, 5 REFS, MARCH 1975

THE PROGRAMME OF FISSION PRODUCT MEASUREMENT IN PHEBUS LOOP IS DESCRIBED. THE EXPERIMENT IS MADE TO VERIFY THE HYPOTHESIS AS FAR AS THE RADIOLOGICAL SAFETY IS CONCERNED. THIS MEASURE IS MADE BY SAMPLE OF LIQUID AND GASES AND BY ON-LINE MEASUREMENT. THE CHEMICAL FORM OF IODINE IS SEARCHED BY SELECTIVE FILTER. THE HYDROGEN PRODUCTION IS ALSO MEASURED IF A WATER METAL INTERACTION OCCURS.

124992 *CONTINUED*
AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

IN PILE EXPERIMENT + MASS TRANSFER + METAL WATER REACTION + FRANCE + R AND D PROGRAM + REACTOR, LMFBR + HYDROGEN + IODINE + JACOBS

124275
KALLI H + LANORE JM
PROGRAM FOR THE CALCULATION OF THE UNCERTAINTY OF THE FAILURE PROBABILITY IN COMPLEX SYSTEMS BY THE MONTE CARLO METHOD (IN FRENCH)
CEA CENTRE D'ETUDES NUCLEAIRES DE SACLAY, FRANCE
FRRSR-57 + SERMA/S-263 +. 37 PPS, FIGS, 13 REFS, (NO DATE)

THE PURPOSE OF THE MONTE CARLO CODE PARTREC-MC IS TO EVALUATE THE UNCERTAINTY RANGE (I.E., THE CONFIDENCE LIMITS) OF A SYSTEM FAILURE PROBABILITY AS A FUNCTION OF THE UNCERTAINTIES IN ITS COMPONENT FAILURE DATA. FAILURE PARAMETERS ARE CONSIDERED AS RANDOM VARIABLES WITH KNOWN DISTRIBUTIONS. THESE DISTRIBUTIONS ARE USED TO SAMPLE PARAMETER VALUES. FOR EACH SET OF PARAMETER VALUES THE DETERMINIST CODE PATREC CALCULATES THE CORRESPONDING SYSTEM FAILURE PROBABILITY. THIS PAPER BRIEFLY DESCRIBES THE PATREC CODE, AND PRESENTS THE PATREC-MC VERSION. A SHORT EXAMPLE IS GIVEN AND DISCUSSED.

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

MONTE CARLO + COMPUTER PROGRAM + PROBABILITY + FAILURE + DISTRIBUTION + JACOBS

124276
KALLI H + LANORE JM
ANALYSIS OF THE UNCERTAINTY OF THE FAILURE PROBABILITY USING THE PATREC-MC CODE (IN FRENCH)
FRRSR-59 + SERMA/LEP-76-735 +. 7 PPS, 1 TAB, 2 FIGS, 3 REFS, 1976

PRESENTS THE RESULTS OF A RELIABILITY STUDY PERFORMED WITH THE MONTE CARLO PATREC-MC CODE. THIS CODE EVALUATES THE UNCERTAINTY RANGE OF A SYSTEM FAILURE PROBABILITY AS A FUNCTION OF THE UNCERTAINTIES IN ITS COMPONENT FAILURE DATA. THE FAULT TREE IS DESCRIBED AND THE RESULTS ARE DISCUSSED.

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

MONTE CARLO + FAILURE + PROBABILITY + COMPUTER PROGRAM + FAILURE, COMPONENT + FAULT TREE ANALYSIS + RELIABILITY, COMPONENT + RELIABILITY, SYSTEM + RELIABILITY ANALYSIS + JACOBS

132470
ANTALOVSKY S
RELIABILITY OF THE SYSTEM FOR THE CONTROL OF THE VOLUME AND CHEMISTRY OF REACTOR WATER (IN FRENCH)
CEA DEPARTMENT DE SURETE NUCLEAIRE, FRANCE
FRRSR-62 + DSN 126 +. 100 PPS, FIGS, REFS, AUG. 1976

THE CHEMICAL AND VOLUME CONTROL SYSTEM IS VERY IMPORTANT FOR THE SAFETY OF OPERATION OF A LIGHT WATER REACTOR. IT HAS VARIOUS FUNCTIONS TO PERFORM AND TWO OF THE MOST IMPORTANT ARE STUDIED IN THIS REPORT: THE INJECTION OF WATER IN ORDER TO PROVIDE HIGH PRESSURE SEAL WATER FOR THE REACTOR COOLANT PUMP SEALS AND THE ADJUSTMENT OF BORIC ACID CONCENTRATION OF THE REACTOR COOLANT FOR CHEMICAL SHIM CONTROL. AFTER HAVING SUMMARIZED THE OPERATION OF THE SYSTEM, SOME SIMPLIFICATIONS ARE MADE TO HELP IN THE RELIABILITY STUDY. FOR EACH OF THE STUDIED FUNCTIONS, A FAULT TREE HAS BEEN DRAWN TO ASSESS QUALITATIVELY AND QUANTITATIVELY THE SYSTEMS RELIABILITY.

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

CONTROL + CONTAINMENT, UNDERWATER + CHEMICAL SHIM + HYDRAULIC ANALYSIS + SENSITIVITY ANALYSIS + CONTAINMENT, WATER FILLED + REACTOR, LWR + WATER + FOREIGN EXCHANGE

126530
CARNINO A
APPLICATION OF STUDIES USING THE FAULT TREE METHOD (IN FRENCH)
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
FRRSR-65 + DSN 136 +. 32 PPS, 8 FIGS, 12 REFS, JUNE 23, 1976

THIS IS A BRIEF PRESENTATION ON THE ADVANTAGES AND DIFFICULTIES CONNECTED WITH THE FAULT-TREE METHOD. A PROBLEM RECENTLY DISCUSSED AMONG SPECIALISTS IN THE NUCLEAR DOMAIN IS THE PROBLEM OF RARE EVENTS. ACCIDENTS ARE RARE, THE SYSTEMS WELL DESIGNED, GIVING RISE TO ONLY FEW OR NO FAILURES DURING A PERIOD OF OBSERVATION SPANNING A HUMAN LIFETIME. HOWEVER, DATA ON THE RELIABILITY OF MATERIALS ARE ESTABLISHED ON SMALL SAMPLES, OR ON MIXED SAMPLES.

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

*FAULT TREE ANALYSIS + *ANALYTICAL TECHNIQUE + EXAMINATION + ACCIDENT ANALYSIS

126868
SIGNORET JP
APPLICATION OF FAULT-TREE ANALYSIS METHODS TO THE EVALUATION OF THE RELIABILITY OF A SAFETY SYSTEM IN A PRESSURIZED WATER POWER STATION (IN FRENCH)
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
FRRSR-66 + DSN 137 +. 21 PPS. 9 FIGS. JUNE 23, 1976

A PARTICULARLY LOGICAL AND FRUITFUL METHOD TO EVALUATE THE RELIABILITY OF SYSTEMS INTENDED TO ASSURE THE SAFETY OF THE REACTOR IS THAT WHICH MAKES USE OF THE FORMATION OF EVENT TREES AND FAULT TREES. THE PRESENT PAPER APPLIES IT TO THE (COOLANT) LOOP. WITH THE AID OF DIAGRAMS AND INFORMATION ON ITS METHOD OF FUNCTIONING, AN EVENT TREE, AND THEN A FAULT TREE ARE PLOTTED. THE LATTER IS THEN QUANTIFIED WITH THE HELP OF NUMERICAL DATA ON THE FAILURE RATES OF THE MATERIALS. THE CRITICAL PATHS ARE EXTRACTED FROM THE FAULT TREE ONCE THE NUMERICAL EVALUATION HAS BEEN DONE. THE WEAK POINTS AS REGARDS RELIABILITY ARE THUS CLEARLY MADE EVIDENT.

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

*FAULT TREE ANALYSIS + RELIABILITY, SYSTEM + REACTOR, PWR + *MAIN COOLING SYSTEM + FOREIGN EXCHANGE

125616
BLIN A
THE COMPLEX SYSTEM RELIABILITY COMPUTER PROGRAM "PATREC-DE" BASED ON THE RECOGNITION OF SHAPES (IN FRENCH)
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
FRRSR-67 + DSN 138 +. 23 PPS. 8 FIGS. 12 REFS. JUNE 1976

A NEW APPROACH IS PRESENTED WHICH MAKES IT POSSIBLE TO CALCULATE THE RELIABILITY OF A SYSTEM WHICH IS DESCRIBED IN THE FORM OF A FAULT TREE. THE COMPUTER PROGRAM PATREC-DE MAKES USE OF THE ELEMENTARY FORMS STORED IN THE MEMORY IN THE FORM OF TREES WITH THEIR MATHEMATICAL EQUIVALENT.*** THE PROBLEM OF DEPENDENCIES (SAME LEAF OF THE TREE REPEATED SEVERAL TIMES) IS SOLVED BY THE APPLICATION OF BAYES' THEOREM. THE EVALUATION OF THE EFFECT OF THE SCATTER OF INPUT PARAMETERS IS TREATED BY A MONTE CARLO METHOD. ALSO A DIFFERENT PROBABILITY LAW MAY BE APPLIED TO EACH LEAF OF THE FAULT TREE.

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

*RELIABILITY, SYSTEM + COMPUTER PROGRAM + *FAULT TREE ANALYSIS + MONTE CARLO + PROBABILITY

129672
MANESSE D + GUIRLET J + ZAMMITE R
DETERMINATION OF TYPICAL ACCIDENTS IN PRESSURIZED-WATER POWER REACTORS AND THEIR CONSEQUENCES, FOR THE PURPOSE OF PREPARING APPROPRIATE EMERGENCY PLANS (IN FRENCH)
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
FRRSR-68 + DSN 139 +. 13 PPS. 6 TABS. 2 FIGS. 3 REFS. MARCH 1977

ASSESSES AND CLASSIFIES TYPICAL ACCIDENTS THAT COULD BE EXPECTED TO OCCUR IN PWRS AND DISCUSSES THE RESULTING CONSEQUENCES RELATIVE TO RADIOACTIVITY RELEASES. THE RADIOACTIVITY RELEASES ARE THEN CHARACTERIZED AND USED IN PERFORMING DOSE CALCULATIONS. RISK ANALYSES ARE THEN PERFORMED TO DEVELOP AND PREPARE APPROPRIATE EMERGENCY PLANS.

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

ACCIDENT ANALYSIS + *ACCIDENT, CONSEQUENCES + *REACTOR, PWR + *RADIOACTIVITY RELEASE + HAZARDS ANALYSIS + DOSE CALCULATION, EXTERNAL + DOSE CALCULATION, INTERNAL + *EMERGENCY PROCEDURE + RADIATION SAFETY AND CONTROL + FRANCE + FOREIGN EXCHANGE

129707
CARNINO A + RAGGENBASS A
HUMAN COMPONENTS IN THE INITIATING PHENOMENA OF NUCLEAR ACCIDENTS (IN FRENCH)
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
FRRSR-69 + DSN 140 +. 10 PPS. 7 REFS. MARCH 1977

DISCUSSES THE HUMAN FACTORS AND PARAMETERS INVOLVED IN THE INITIATING OF POTENTIAL NUCLEAR REACTOR ACCIDENTS. DISCUSSES THE PROPER EDUCATION OF OPERATIONS PERSONNEL AND PLANT STAFFING, SELECTION CRITERIA FOR SPECIFIC JOBS, AND HUMAN RELIABILITY ANALYSIS. FAULT TREE ANALYSIS IS USED TO ANALYZE THE HUMAN FACTOR IN ACCIDENT ANALYSES.

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

ACCIDENT ANALYSIS + *POWER PLANT, NUCLEAR + *OPERATOR ACTION + STAFFING + *RELIABILITY ANALYSIS + FAULT TREE ANALYSIS + FRANCE + FOREIGN EXCHANGE

131550
ROCHE R
SOME COMMENTS ABOUT THE USE OF J(SUB 1) INTEGRAL CRITERION (IN FRENCH & ENGLISH)
CEA CENTRE D'ETUDES NUCLEAIRES DE SACLAY, FRANCE

131650 *CONTINUED*
CEA-N-1933 + FRRSR-70 +. 22 PPS. 15 REFS. JAN. 1977

IF J(SUB 1) IS PATH INDEPENDENT, IT CAN BE CONSIDERED AS A CRACK TIP SINGULARITY CHARACTERISATION AND THEN, AS A GOOD CRITERION FOR CRACK INITIATION. BUT IT IS NOT PROVED THAT J(SUB 1) IS PATH INDEPENDENT IN GENERAL CASE. THE CURRENT PRACTICE IS TO CONSIDER A MATERIAL WITH A MECHANICAL POTENTIAL ENERGY. IT SEEMS THAT J(SUB 1) IS PATH INDEPENDENT FOR RADIAL LOADING BUT THE QUESTION IS STILL OPEN FOR MORE COMPLEX WAYS OF LOADING.

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

FRANCE + CRACK + STRESS ANALYSIS + MATHEMATICAL TREATMENT + FRACTURE TOUGHNESS + ELASTICITY + FOREIGN EXCHANGE

127050
BLIN A
DESCRIPTION AND USE OF PATREC-RCM CODE (IN FRENCH)
CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
FRRSR-71 + SETSSR 150 +. 33 PPS. 3 REFS. FEB. 1977

THIS CODE, BASED ON A PATTERN RECOGNITION METHOD, PERMITS THE CALCULATION OF THE PROBABILITY OF FAILURE OF COMPLEX SYSTEMS DESCRIBED BY A FAULT TREE. THE ELEMENTARY EVENTS (LEAVES) CAN FOLLOW SEVERAL GIVEN LAWS OF PROBABILITY (1) TO SEARCH THE MINIMAL CUT SETS (OPTIONAL), (2) TO FIND THE WEAK POINTS OF THE SYSTEM AND TO EVALUATE THEIR WEIGHT IN THE TOTAL PROBABILITY, AND (3) TO GIVE AN ESTIMATION OF THE SENSITIVITY INDEXES WITH THE VARIATION OF THE RELIABILITY PARAMETERS OF THE COMPONENTS.

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

PROBABILITY + FAILURE + RELIABILITY, COMPONENT + FAULT TREE ANALYSIS + COMPUTER PROGRAM + FOREIGN EXCHANGE

2. GERMAN (FRG) SAFETY RESEARCH REPORTS

THE FOLLOWING IS A LISTING OF MICROFICHED LIGHT-WATER REACTOR SAFETY RESEARCH REPORTS RECEIVED FROM THE FEDERAL REPUBLIC OF GERMANY DURING THE PERIOD 1975-1977 UNDER THE TECHNICAL EXCHANGE AGREEMENT.

111127
STRESS MEASUREMENTS IN A MODEL CONTAINMENT WITH REINFORCED CONCRETE (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
BF-RS 50-23-1 +. 55 PPS, 8 TABS, 40 FIGS, AUG. 1975

FOR EXPERIMENTAL INVESTIGATIONS ON THE SAFETY OF NUCLEAR POWER STATIONS, A MODEL CONTAINMENT OF REINFORCED CONCRETE WAS CONSTRUCTED. THE INTERNAL PRESSURE OF THE CONTAINMENT WAS INCREASED STEPWISE IN SEVERAL EXPERIMENTS. THE MEASUREMENTS MADE IN THESE EXPERIMENTS COVERED THE STRAIN OF THE REINFORCING STEEL IN THE PRESSURE- RESISTANT CONCRETE SHELL, THE WIDENING OF HAIRLINE CRACKS IN THE CONCRETE, AND THE INTEGRAL ELONGATION OF THE WHOLE CONTAINMENT BOTH IN PERIPHERAL AND MERIDIAN DIRECTION. THE RESULTS INDICATE, THAT THE FORMATION OF HAIRLINE CRACKS IN THE CONCRETE DUE TO TENSILE LOAD IS MODERATE.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

MOCKUP + CONTAINMENT INTEGRITY + *STRESS ANALYSIS + GERMANY + STRESS STRAIN DATA + R AND D PROGRAM + PRESSURE TRANSIENT + *CONCRETE, REINFORCED + CRACK

111128
PRESSURE AND TEMPERATURE HISTORIES IN BLOWDOWN EXPERIMENTS IN A MODEL CONTAINMENT (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
BF-RS 50-62-2 +. 27 PPS, 1 TAB, 12 FIGS, OCT. 1975

PRESSURE AND TEMPERATURE HISTORIES OCCURING UNDER PWR CONDITIONS DURING THE SIMULATION OF A ONE-END RUPTURE OF THE MAIN COOLANT PIPE IN A MODEL CONTAINMENT ARE SHOWN. THE EXPERIMENTAL FACILITY, WHICH IS A QUARTER-SCALE MODEL OF A CONTAINMENT OF A 1200-MWE PWR NUCLEAR POWER PLANT, IS BRIEFLY DESCRIBED.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

CONTAINMENT + *ACCIDENT, LOSS OF COOLANT + MOCKUP + REACTOR, PWR + TEST, PLANT RESPONSE + GERMANY + SIMULATION + R AND D PROGRAM + *PRESSURE TRANSIENT + *THERMAL TRANSIENT

111498
EXPERIMENTAL INVESTIGATIONS OF PRESSURE AND TEMPERATURE LOADS ON A CONTAINMENT AFTER A LOSS-OF-COOLANT ACCIDENT
BATTELLE-INSTITUTE, FRANKFURT, F.R. GERMANY
GERRSR-14 +. 16 PPS, 15 FIGS, 1 REF, OCT. 1975

THE PHENOMENA OCCURING WITHIN A CONTAINMENT DURING A LOCA ARE CURRENTLY INVESTIGATED THROUGH EXPERIMENTS WITH A MODEL CONTAINMENT AT BATTELLE-INSTITUT FRANKFURT ON BEHALF OF THE BUNDESMINISTERIUM FUR FORSCHUNG UND TECHNOLOGIE, BONN. THE EXPERIMENTAL RESULTS ARE TO BE COMPARED WITH THE RESULTS OF MODEL CALCULATIONS IN ORDER TO IMPROVE THE CALCULATIONAL METHODS. UP TO NOW THE TEST FACILITY HAS BEEN USED FOR FOUR TRIAL RUNS AND EIGHT PWR-LOCA-EXPERIMENTS WITH SINGLE- AND DOUBLE-ENDED PIPE RUPTURES OF 100 MM DIAMETER IN A STEAM GENERATOR COMPARTMENT AND IN THE NOZZLE COMPARTMENT. A COMPARISON BETWEEN EXPERIMENTS AND CALCULATIONS SHOWS DISCREPANCIES INDICATING THE NEED FOR FURTHER DEVELOPMENTS OF CALCULATIONAL METHODS.

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

HYDRAULIC EXPERIMENT + ACCIDENT, LOSS OF COOLANT + REACTOR, PWR + CONTAINMENT INTEGRITY + TEMPERATURE GRADIENT + BLOWDOWN + JACOBS

125264
ERROR ANALYSIS FOR THE DATA ACQUISITION AND PROCESSING SYSTEM OF THE CONTAINMENT TEST FACILITY (IN GERMAN)
BATTELLE-INST. E.V., FRANKFURT AM MAIN, F.R. GERMANY
GERRSR-30 + BF-RS 50-21-3 +. 41 PPS, 2 TABS, 4 FIGS, APRIL 1976

THE EXPERIMENTAL FACILITY 'NUCLEAR REACTOR CONTAINMENT' USED IN THE RESEARCH PROJECT RS 50 HAS TO MEET STRINGENT REQUIREMENTS WITH RESPECT TO DATA ACQUISITION AND PROCESSING. A LARGE NUMBER OF MEASURED VALUES MUST BE ACQUIRED AND PROCESSED RELIABLY AND WITH SUFFICIENT ACCURACY IN A ONCE-THROUGH PROCEDURE (WHICH COULD BE REPEATED ONLY AT GREAT EFFORT), AT HIGH TIME RESOLUTION, AND UNDER EXTREME ENVIRONMENTAL CONDITIONS USING AN EXPERIMENTAL SET-UP EXTENDING OVER SEVERAL ROOMS. AN ERROR ANALYSIS FOR THE EXPERIMENTAL FACILITY RESULTS IN THE PROBABLE (P = 95.4%, GAUSSIAN 2 DELTA) RELATIVE TOTAL ERROR LIMITS FOR THE INDIVIDUAL MEASURED VALUES. THE RELATIVE MEASURING ERRORS RELATE TO THE MEASURING TIME AND RANGE.

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

DATA PROCESSING + CONTAINMENT + TESTING + JACOBS + DATA COLLECTION

134200
JAX P + EISENBLATTER J
INVESTIGATIONS ON A CONTINUOUSLY OPERATING SYSTEM FOR CRACK GROWTH SURVEILLANCE IN PRESSURE VESSELS - PART IV (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY

134200 *CONTINUED*
BF-R-61.958-11 + GERRSR-93 +. 31 PPS. 3 TABS. 13 FIGS. APRIL 1976

PART IV DISCUSSES FURTHER DEVELOPMENT OF ACOUSTICS EMISSION ANALYSIS WITH REGARD TO APPLICATION AT THE REACTOR VESSEL.

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

*ACOUSTICS + NOISE ANALYSIS + *PRESSURE VESSELS + CRACK + GROWTH/DEVELOPMENT + FOREIGN EXCHANGE

116930
SKOUTAJAN R
R&D REGARDING THE VISCOSITY OF MOLTEN LWR CORE MATERIALS AND INVESTIGATION OF THE COMPATIBILITY BETWEEN THESE MATERIALS AND CRUCIBLE-MATERIALS (IN GERMAN)
BATTELLE-INSTITUT E.V.. FRANKFURT AM MAIN. GERMANY
BMFT-RS 71 +. 80 PPS. 9 TABS. 20 FIGS. 26 REFS. JUNE 1976

FINDINGS WHICH ARE OF IMPORTANCE ARE: MELTING OF LARGE BATCHES OF CORE MELT IN AN ELECTRIC ARC FURNACE DOES NOT PROVIDE HOMOGENEOUS SAMPLES AS REQUIRED FOR MEASUREMENTS; TANTALUM CARBIDE CRUCIBLES ARE LOCALLY ATTACKED BY THE CORE MELT TO A HIGHER DEGREE THAN WAS EXPECTED FROM THE COMPATIBILITY EVALUATION; SUBSTANTIAL QUANTITIES OF CORE MELT EVAPORATE DURING MELTING. ON ACCOUT OF THESE FINDINGS AND THE RECENT DIFFERENTIATION IN THE POSSIBLE TYPES OF CORE MELT. THE INVESTIGATIONS WERE TERMINATED AND THE OBJECTIVES OF THE FURTHER INVESTIGATIONS AT BATTELLE WERE ADAPTED TO THE NEW FINDINGS. MEASUREMENTS ON OXIDISED CORE MELT IN THORIUM OXIDE CRUCIBLES OF OPTIMISED DIMENSIONS AND QUALITY WILL BE CARRIED OUT.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER. P.O. BOX 62. OAK RIDGE. TENN. 37830

MOLTEN FUEL + *CORE MELTDOWN + GERMANY + REACTOR. LWR + *R AND D PROGRAM + CORE CATCHER

126650
INVESTIGATIONS OF THE PHENOMENA OCCURRING WITHIN A REACTOR CONTAINMENT DURING A LOSS OF COOLANT ACCIDENT IN A WATER-COOLED REACTOR
BATTELLE-INSTITUT E.V.. FRANKFURT AM MAIN. F.R. GERMANY
GERRSR-97 + BF RS 50-32-C16 +. APPROX. 225 PPS. FIGS. REFS. NOV. 1976

THE OBJECTIVE OF THIS RESEARCH PROJECT IS TO INVESTIGATE THE PHENOMENA OCCURRING WITHIN A REACTOR CONTAINMENT DURING A LOSS OF COOLANT ACCIDENT. THE EXPERIMENTAL RESULTS ARE TO BE COMPARED WITH THE RESULTS OF MODEL CALCULATIONS. THIS REPORT CONTAINS THE CONFIGURATION OF THE EXPERIMENTAL FACILITY AND THE DATA MEASURED IN EXPERIMENT OF C16. THE MAIN GOAL OF THIS EXPERIMENT WAS THE INVESTIGATION OF THE PROPAGATION OF THE WATER/STEAM-MIXTURE FROM THE SITE OF RUPTURE WITHIN A LARGE COMPARTMENT. SECONDARY GOALS WERE THE MEASUREMENTS OF THE JET FORCE AND THE JET FORCE DISTRIBUTION BY MEANS OF A DEFELCTOR PLATE AS WELL AS THE INVESTIGATION OF THE PRESSURIZATION OF THE CONTAINMENT.

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

REACTOR. PWR + ACCIDENT. LOSS OF COOLANT + BLOWDOWN + OUT OF PILE EXPERIMENT + COMPARISON. THEORY AND EXPERIENCE + FLOW. TWO PHASE + GERMANY + FOREIGN EXCHANGE

139834
DESCRIPTION OF THE PROGRAM BLOCKS RS DV FOR DATA PROCESSING (IN GERMAN)
BATTELLE-INSTITUT E.V.. FRANKFURT AM MAIN. F.R. GERMANY
BF-RS00168-21-5 + BF-RS50-21-5 + GERRSR-94 +. 96 PPS. FIGS. REFS. JAN. 1977

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

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

COMPUTER PROGRAM + *DATA PROCESSING + SYSTEM DESCRIPTION + FOREIGN EXCHANGE

131130
INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A REACTOR CONTAINMENT DURING A LOSS OF COOLANT ACCIDENT: EXPERIMENT C14 (IN GERMAN)
BATTELLE-INSTITUT E.V.. FRANKFURT AM MAIN. F.R. GERMANY
BF RS 50-32-C14 + GERRSR-92 +. 100 PPS. TABS. FIGS. FEB. 1977

SUMMARIZES THE CONDITIONS AND RESULTS OF THE EXPERIMENT WHICH SIMULATES A DOUBLE-ENDED RUPTURE OF THE COOLANT PIPE IN A NOZZLE COMPARTMENT BY THE SIMULTANEOUS OPENING OF TWO RUPTURE DISKS AT OPPOSITE PIPE ENDS. HOWEVER. A SINGLE-END PIPE RUPTURE ON THE SIDE OF THE SMALL VESSEL WAS PRODUCED (DIAPHRAGM DIAMETER 100). THE RESULTING PROCESSES IN THE CONTAINMENT WERE ANALYZED AND ARE DESCRIBED IN THE FORM OF DIAGRAMS.

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

GERMANY + R AND D PROGRAM + ACCIDENT. LOSS OF COOLANT + SYSTEM OPERABILITY IN ACCIDENT + CONTAINMENT +

131130 *CONTINUED*
BEHAVIOR + SIMULATION + ANALYTICAL MODEL + FOREIGN EXCHANGE + COMPARISON, THEORY AND EXPERIENCE

131129
INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A REACTOR CONTAINMENT DURING A LOSS OF COOLANT ACCIDENT: EXPERIMENT C7 (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
BF RS 50-32-C7 + GERRSR-95 +. 125 PPS. TABS. FIGS. FEB. 1977

SUMMARIZES THE CONDITIONS AND RESULTS OF THE EXPERIMENT WHICH SIMULATES A DOUBLE-ENDED RUPTURE OF THE COOLANT PIPE IN A NOZZLE COMPARTMENT BY THE SIMULTANEOUS OPENING OF TWO RUPTURE DISKS AT OPPOSITE PIPE ENDS. THE RESULTING PROCESSES IN THE CONTAINMENT WERE ANALYZED AND ARE DESCRIBED IN THE FORM OF DIAGRAMS.

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

GERMANY + R AND D PROGRAM + ACCIDENT, LOSS OF COOLANT + TESTING + SIMULATION + CONTAINMENT + SYSTEM OPERABILITY IN ACCIDENT + ANALYTICAL MODEL + COMPARISON, THEORY AND EXPERIENCE + BEHAVIOR + FOREIGN EXCHANGE

127278
INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
GERRSR-106 + BF RS 50-30-D1 +. 90 PPS. FIGS. REFS. MARCH 1977

THE PRECEDING EXPERIMENTS OF SERIES C SHOWED THAT COMPUTER CODES USED FOR DESIGN OF REACTOR CONTAINMENTS PARTLY LEAD TO ESTIMATES THAT ARE MUCH TOO HIGH. THE DEVISED EXPERIMENTS OF SERIES D ARE TO PROVIDE EXPERIMENTAL DATA FOR DEVELOPMENT OF MORE ACCURATE COMPUTER CODES. THE PRESSURES GENERATED IN THE CONTAINMENT COMPARTMENTS AND ALSO THE DIFFERENTIAL PRESSURES BETWEEN THE COMPARTMENTS REMAINED DISTINCTLY BELOW THE COMPUTED VALUES. THUS, QUALITATIVELY THE SAME OVERESTIMATE RESULTS AS IN THE PRECEDING EXPERIMENTS OF SERIES C.

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

CONTAINMENT + COMPUTER PROGRAM + PRESSURE, INTERNAL + REACTOR, LWR + ACCIDENT, LOSS OF COOLANT

126651
RESEARCH RESULTS ON BEHAVIOR OF A TWO-PHASE LEVEL IN A TANK (HEIGHT 11.2 M) WITHOUT INTERNALS AFTER A STEAM LINE RUPTURE (IN GERMAN)
BATTELLE INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
GERRSR-107 + RS 0016 B +. APPROX. 75 PPS. FIGS. MARCH 1977

THE RAPID DEPRESSURIZATION AFTER A STEAM LINE BREAK TAKING PLACE IN A BWR PRESSURE VESSEL OR IN A PWR STEAM GENERATOR LEADS TO FLASHING OF THE WATER. AS THE PHASE SEPARATION PROCESSES ARE COMPARATIVELY SLOW, THE FLASHING PROCESS CAUSES THE MIXTURE OF WATER AND STEAM TO RISE TO THE DISCHARGE NOZZLE. THE PRESENT REPORT COVERS THE RESULTS OF THE RS 0016 B EXPERIMENT SWR 2R, WHICH RELATE TO THE ABOVE-DESCRIBED PROCESS. IN ADDITION, SOME RESULTS FROM PROJECTS RS 16 AND RS 50 ON THE SAME SUBJECT ARE PRESENTED IN THE APPENDIX.

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

REACTOR, LWR + ACCIDENT, STEAM LINE RUPTURE + ACCIDENT, LOSS OF PRESSURE + OUT OF PILE EXPERIMENT + GERMANY + FOREIGN EXCHANGE

127615
QUICK LOOK REPORT EXPERIMENT D6 - INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
GERRSR-112 + BF-RS 50-30-D6 +. APPROX. 120 PPS. TABS. FIGS. MAY 1977

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

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

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

129977
MODEL CALCULATIONS TO VERIFY THE SHORT-TERM BEHAVIOR OF ABSOLUTE AND DIFFERENTIAL PRESSURES MEASURED IN THE RS

129977 *CONTINUED*
50 EXPERIMENTS C13 AND C15 (IN GERMAN)
BATTELLE-INSTITUTE E.V., FRANKFURT AM MAIN, F.R. GERMANY
GERRSR-120 + BF-RS 50-24-2 +. 67 PPS. TABS. FIGS. MAY 1977

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

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

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

127852
MODEL CALCULATIONS TO VERIFY THE SHORT-TERM BEHAVIOR OF ABSOLUTE AND DIFFERENTIAL PRESSURES MEASURED IN THE RS 50 EXPERIMENTS, SERIES C
BATTELLE INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
GERRSR-121 + BF-RS 50-24-1 +. 112 PPS. FIGS. MARCH 1977

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

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

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

128450
INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER COOLED REACTORS - COMPLETE EXPERIMENT DOCUMENTATION D6 (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
GERRSR-139 + BF RS 50-32-D6 +. APPROX. 120 PPS. FIGS. MAY 1977

EXPERIMENTS OF SERIES D (SIMPLIFIED IDEALIZED EXPERIMENTS) WERE RUN TO PROVIDE DATA FOR THE DEVELOPMENT OF MORE ACCURATE COMPUTER CODES. A COMPARISON OF EXPERIMENTAL RESULTS WITH CALCULATIONS PERFORMED WITH THE CONTAINMENT CODE ZOCO 6 (GRS/MUNCHEN) SHOWED THE PRESSURE BUILD-UP IN THE COMPARTMENT ZONES AND THE PRESSURE DIFFERENCES BETWEEN COMPARTMENTS WAS--AS IN THE EARLIER EXPERIMENTS--REMARKABLY LOWER THAN PREDICTED BY CALCULATIONS PERFORMED WITH STANDARD SETS OF CALCULATIONAL PARAMETERS. EFFORTS TO CLARIFY THE DISCREPANCIES BETWEEN CALCULATIONAL AND EXPERIMENTAL RESULTS MUST THEREFORE CONTINUE.

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

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

129745
BAUKAL W + NIXDORF J + SKOUTAJAN R + WINTER H
STUDY OF THE SIGNIFICANCE AND OF THE FEASIBILITY OF MEASURING THE HEATS OF CHEMICAL REACTIONS DURING CORE MELTING, AND OF THE TOTAL INTEGRATED HEAT FROM CORE MELTING (IN GERMAN)
BATTELLE INSTITUTE E.V., FRANKFURT AM MAIN, F.R. GERMANY
GERRSR-133 + BMFT-RS-197 +. 85 PPS. TABS. FIGS. JUNE 1977

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

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

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

128374
INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER COOLED REACTORS - QUICK LOOK REPORT EXPERIMENT D7 (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
GERRSR-138 + BF-RS 50-30-D7 +. APPROX. 135 PPS. TABS. FIGS. JUNE 1977

THE MAIN RESULTS OF EXPERIMENT D6 WERE CONFIRMED. IN ADDITION THE CALCULATIONS OF D7 AND THE ANALYTICAL RESEARCH SHOWED THAT USING THE EXISTING VERSION OF ZOCO 6 IT IS NOT POSSIBLE TO ACHIEVE FULL AGREEMENT BETWEEN MEASURED AND CALCULATED PRESSURES WITH PHYSICALLY REALISTIC INPUT VALUES FOR THE HEAT TRANSFER COEFFICIENT. A MORE PROFOUND ANALYSIS OF THE INTERCOMPARTMENT FLOW IS NECESSARY FOR IMPROVING THE FLOW MODELS.

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

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

131127
INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS - COMPLETE EXPERIMENT DOCUMENTATION D7 (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
BF RS 50-32-D7 + GERRSR-140 +. APPROX. 125 PPS. FIGS. JUNE 1977

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

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

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

131560
INVESTIGATION OF THE PHENOMENA OCCURRING WITHIN A MULTI-COMPARTMENT CONTAINMENT AFTER RUPTURE OF THE PRIMARY COOLING CIRCUIT IN WATER-COOLED REACTORS. QUICK LOOK REPORT EXPERIMENT D3 (IN GERMAN)
BATTELLE-INSTITUT E.V., FRANKFURT AM MAIN, F.R. GERMANY
BF RS 50-30-D3 + GERRSR-151 +. 175 PPS. TABS. FIGS. SEPT. 1977

THE CONTAINMENT EXPERIMENTS OF SERIES D ARE TO BE CARRIED OUT UNDER SIMPLIFIED IDEALIZED CONDITIONS TO PROVIDE EXPERIMENTAL DATA FOR IMPROVING AVAILABLE CONTAINMENT CODES. IMPORTANT PARAMETERS OF EXPERIMENT D3 ARE DESCRIBED. PRECALCULATIONS AND POST-EXPERIMENT CALCULATIONS HAVE BEEN CARRIED OUT FOR EXPERIMENT D3. THE RESULTS OF A COMPARISON WITH EXPERIMENTAL DATA ARE GIVEN.

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

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

132894
FRIZ G + RIEBOLD W + SCHULZE W
FINAL REPORT OF RESEARCH PROJECT RS-77: RESEARCH ON THE THERMODYNAMICS OF TWO PHASE FLOW (IN GERMAN)
COMMISSION OF THE EUROPEAN COMMUNITY
RS-77 + GERRSR-131 +. 130 PPS. FIGS. 1976

BRIEF DESCRIPTION OF THE TEST APPARATUS WITH THE REPORT CONSISTING PRIMARY OF DATA IN THE FORM OF GRAPHS.

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

GERMANY + R AND D PROGRAM + THERMODYNAMICS + FLOW, TWO PHASE + FOREIGN EXCHANGE + FLOW THEORY AND EXPERIMENTS

132892
HOHMANN H + KOTTOWSKI HM + TOSELLI F
EXPERIMENTAL INVESTIGATIONS OF VAPOR EXPLOSIONS (IN GERMAN)
COMMISSION OF THE EUROPEAN COMMUNITIES
BMFT-RS 76 + GERRSR-117 +. 50 PPS. 27 FIGS. 4 REFS. MARCH 1977

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

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

132892 *CONTINUED*
GERMANY + R AND D PROGRAM + FUEL COOLANT INTERACTION + EXPLOSION + WATER VAPOR + FOREIGN EXCHANGE

127825
LANGHEIM H
IMPACT TESTS WITH REINFORCED CONCRETE SLABS. PART I (IN GERMAN)
ERNST-MACH INSTITUT, GERMANY
GERRSR-114 + EMI-E 9/76 + RS-102-07-3 +. 60 PPS, 6 TABS, 26 FIGS, 3 REFS, JUNE 1976

DESCRIBES THE DEVELOPMENT AND EVALUATION OF IMPACT STUDIES ON 53 REINFORCED CONCRETE SLABS. CYLINDRICAL, SOLID C 110 W 1 - STEEL PROJECTILES WITH CALIBERS 15, 20 AND 40 MM WERE FIRED WITH VELOCITIES RANGING FROM 100 UP TO 2000 M/S ON 700 X 700 MM SIZED REINFORCED CONCRETE SLABS OF 200 UP TO 400 MM THICKNESS. THE MEASURED PENETRATION DEPTHS ARE COMPARED WITH THE PENETRATION DEPTHS OF THE CLASSIC PENETRATION FORMULAE (PETRY, BRL, COE AND HN-NDRC). DEPENDING UPON PROJECTILE CALIBER AND VELOCITY RANGE EITHER THE PETRY OR THE HN-NDRC-FORMULA SUPPLY SATISFACTORY CONSISTENCY WITH OUR TEST RESULTS.

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

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

127837
LANGHEIM H
IMPACT TESTS WITH REINFORCED CONCRETE SLABS. PART II (IN GERMAN)
ERNST-MACH-INSTITUT, GERMANY
GERRSR-115 + EMI-E 14/76 + RS-102-07-4 +. 56 PPS, 4 TABS, 23 FIGS, 3 REFS, OCT. 1976

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

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

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

126717
LIST OF REPORTS ON REACTOR SAFETY RESEARCH FROM BMFT AND USNRC - REPORTING PERIOD OCTOBER-DECEMBER 1976
GESELLSCHAFT FUR REAKTORSICHERHEIT MBH, F.R. GERMANY
GERRSR-96 + GRS-F-34 +. 63 PPS, FEB. 1977

THIS LIST REVIEWS REPORTS FROM THE FEDERAL REPUBLIC OF GERMANY AND FROM THE USA CONCERNING SPECIAL PROBLEMS IN THE FIELD OF REACTOR SAFETY RESEARCH WHICH HAVE BEEN COLLECTED IN THE GESELLSCHAFT FUR REAKTORSICHERHEIT. 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. THIS EDITION CONTAINS ALL REPORTS AS REGISTERED FROM OCTOBER-DECEMBER 1976.

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

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

129906
ULLRICH R + ROHDE J + MULLER M + HICKEN EF
ANALYTICAL RESULTS OF THE NON-NUCLEAR LOFT TESTS L1 AND L2. PROGRESS REPORT (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT MBH, F.R. GERMANY
GERRSR-132 + GRS-A-7 +. 141 PPS, TABS, FIGS, 16 REFS, FEB. 1977

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

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

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

127240
PROGRESS REPORT ON RESEARCH INTO THE FIELD OF LIGHT WATER REACTOR SAFETY - OCTOBER 1-DECEMBER 31, 1976
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GERRSR-103 + GRS-F-35 +. 462 PPS. FIGS. MARCH 1977

OBJECTIVE OF THIS PROGRAM IS TO INVESTIGATE IN GREATER DETAIL THE SAFETY MARGINS OF NUCLEAR ENERGY PLANTS AND THEIR SYSTEMS AND THE FURTHER DEVELOPMENT OF SAFETY TECHNOLOGY. EACH PROGRESS REPORT REPRESENTS A COMPILATION OF INDIVIDUAL REPORTS ABOUT THE DIFFERENT PROJECTS OF THE RESEARCH PROGRAM. THE INDIVIDUAL REPORTS ARE PREPARED BY THE CONTRACTORS THEMSELVES. THIS WORK IS SPONSORED BY GERMANY'S FEDERAL MINISTER FOR RESEARCH AND TECHNOLOGY (BMFT).

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

GERMANY + R AND D PROGRAM + SAFETY ANALYSIS + SAFETY PROGRAM + REACTOR, BWR + REACTOR, PWR + REACTOR, LWR + FOREIGN EXCHANGE

128813
REPORT ON RESEARCH SPONSORED BY THE FEDERAL MINISTRY OF RESEARCH AND TECHNOLOGY ON THE SUBJECT OF REACTOR SAFETY; REPORT PERIOD JANUARY 1-MARCH 31, 1977 (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT MBH, F.R. GERMANY
GERRSR-141 + GRS-F-40 +. 386 PPS. TABS. FIGS. JUNE 1977

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

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

*GERMANY + *R AND D PROGRAM + REACTOR, LWR + SAFETY ANALYSIS + SAFETY MARGIN

132895
DESCRIPTION OF THE CURRENT PRACTICE IN THE DEVELOPMENT OF SAFETY CRITERIA FOR NUCLEAR POWER PLANTS. SAFETY CRITERIA 8.2: EXPLORATION OF BASIC SAFETY REQUIREMENT (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GERRSR-143 +. 19 PPS. 17 REFS. JULY 1977

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

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

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

131561
UNUSUAL OCCURRENCES IN NUCLEAR POWER PLANTS IN THE FEDERAL REPUBLIC OF GERMANY IN THE YEARS 1965-1977 (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GERRSR-147 +. 17 PPS. JULY 1977

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

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

*INCIDENT COMPILATION + *GERMANY + REACTOR, POWER + REACTOR, BWR + REACTOR, PWR + FAILURE, EQUIPMENT + FAILURE, INSTRUMENT + FOREIGN EXCHANGE

131558
FRANZEN LF + GUTSCHMIDT WD
PRIORITY LISTING OF TECHNICAL SAFETY ASPECTS OF NUCLEAR POWER PLANTS (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
GRS-S-17 + GERRSR-149 +. 62 PPS. 5 TABS. 21 FIGS. 17 REFS. JULY 1977

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

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

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

131559
HELLER H + MAY H
NUCLEAR POWER PLANT BIBLIS. UNIT A: DESCRIPTION OF EXPERIENCES IN 1975 & 1976 (IN GERMAN)
GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH. F.R. GERMANY
GRS-S-18 + GERRSR-150 +. 17 PPS. 9 FIGS. 9 REFS. JULY 1977

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

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

REACTOR. PWR + GERMANY + POWER PLANT. NUCLEAR + OPERATING EXPERIENCE + OPERATING EXPERIENCE SUMMARY + FOREIGN EXCHANGE

126214
HOLLSTEIN T + BLAUEL JG + URICH B
ON THE ASSESSMENT OF CRACKS FOR ELASTIC-PLASTIC MATERIAL BEHAVIOR (IN GERMAN)
INST. FUR FESTKORPERMECHANIK. F.R. GERMANY
GERRSR-99 +. 115 PPS. TABS. FIGS. REFS. NOV. 1976

VALUES OF THE CRACK TIP OPENING STRETCH COS. THE STRESS INTENSITY FACTOR K AND THE ENERGY INTEGRAL J ARE DETERMINED FROM DIRECT OPTICAL AND CLIP GAGE MEASUREMENTS AS WELL AS FROM LOAD DISPLACEMENT CURVES FOR THREE MODIFICATIONS OF THE STEEL 22 NIMOCR 3 7 AT ROOM TEMPERATURE. THESE PARAMETERS AND THEIR CORRELATIONS ARE DESCRIBED AS A FUNCTION OF THE PREVAILING CONDITIONS OF GEOMETRY AND MICROSTRUCTURE WHICH INFLUENCE THE DEFORMATION BEHAVIOR IN THE PLASTIC ZONE AT THE CRACK TIP. A LINEAR RELATIONSHIP BETWEEN COS AND J IS SHOWN TO EXIST FOR SITUATIONS EVEN BEYOND ONSET OF STABLE CRACK GROWTH WHICH IS DETERMINED AS A SAFE LIMIT AGAINST FAILURE USING THE POTENTIAL METHOD AND THE INTERRUPTED LOAD PROCEDURE.

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

CRACK + STRESS + MICROSTRUCTURE + STEEL + NICKEL + ALLOY

126213
HODULAK L + KORDISCH H + KUNZELMANN S + SOMMER E
FRACTURE PROPERTIES OF PART-THROUGH CRACKS (IN GERMAN)
INSTITUT FUR FESTKORPERMECHANIK. F.R. GERMANY
GERRSR-98 +. 99 PPS. FIGS. REFS. JAN. 1977

THE EXTENSION OF PART-THROUGH CRACKS IN PRECRACKED THICK-WALLED PLATES AND TUBES OF THE REACTORSTEEL 22 NI MO CR 3 7 UNDER FATIGUE LOADING HAS BEEN INVESTIGATED. GROWTH CHARACTERISTICS HAVE BEEN DETERMINED BY ANALIZING THE FRACTURE SURFACES. FOR THIS PURPOSE ARTIFICIAL MARKINGS WERE INDUCED ON THE FRACTURE SURFACES BY AN OVERLOAD TECHNIQUE. THE CRACK EXTENSION IN THE DIRECTION OF THE SEMIAXES AS WELL AS THE CRACK GROWTH ALONG THE WHOLE FRONT HAS BEEN EVALUATED AND INTERPRETED IN TERMS OF THE PARAMETERS: RELATIVE CRACK DEPTH. RATIO OF CRACK WIDTH TO CRACK DEPTH. GEOMETRY OF THE SPECIMENS. THE DEGREE OF STRESS STATE AT THE CRACK FRONT AND THE BEHAVIOR OF THE MATERIAL.

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

CRACK + FRACTURE TOUGHNESS + PLATE + TUBING + FATIGUE + STEEL + NICKEL + ALLOY + FOREIGN EXCHANGE

135826
RAUSER P
CLASSIFICATION OF MISSILES. PROJECTILES. AND FRAGMENTS DURING POSSIBLE ACCIDENT SEQUENCES (IN GERMAN)
ERNST-MACH-INSTITUT. F.R. GERMANY
RS 102-7 + GERRSR-23 +. 70 PPS. 26 TABS. 119 REFS. OCT. 1974

THIS WORK SUPPLEMENTS ALREADY KNOWN CLASSIFICATION SCHEMES FOR FAILURES AND ACCIDENTS ESPECIALLY IN VIEW OF IMPACT EVENTS FROM OUTSIDE AND INSIDE OF NUCLEAR POWER PLANTS. IN ORDER TO DEFINE THE RANGE OF PROBABLE DAMAGES BY IMPACT THIS WORK PRESENTS LISTS OF REACTOR SPECIFIC TARGETS AND CONSTRUCTION ELEMENTS. THE MOST IMPORTANT OF WHICH ARE CONCRETE. REINFORCED CONCRETE STEEL AND SPECIAL MATERIALS. SIMILARLY. POSSIBLE PROJECTILES AND FRAGMENTS ORIGINATING FROM THE INSIDE AND OUTSIDE OF NUCLEAR POWER PLANTS ARE SUMMARIZED IN TERMS OF REACTOR SPECIFIC COMPONENT PART FAILURES. ENVIRONMENTAL ACCIDENTS AND INFLUENCE BY THIRD PARTY (SABOTAGE).

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

*ACCIDENT ANALYSIS + POWER PLANT. NUCLEAR + FAILURE. COMPONENT + STRUCTURAL INTEGRITY + SABOTAGE + FOREIGN EXCHANGE + *MISSILE GENERATION AND PROTECTION

120985
KORBER H
INVESTIGATION OF SOME IMPORTANT PARAMETERS IN A HYPOTHETICAL CORE MELTDOWN ACCIDENT (IN GERMAN)
UNIVERSITAT STUTTGART. F.R. GERMANY

120985 *CONTINUED*
BMFT-RS 119 +. 130 PPS. FIGS. REFS. JAN. 1975

THE COMPUTER PROGRAMS CHEMLOC, NURLOC AND RELAP 3 HAVE BEEN USED IN ORDER TO STUDY THE INFLUENCE OF VAPOR PRODUCTION RATE, ZR-H(SUB 2)O-REACTION, POWER DISTRIBUTION AND CORE STRUCTURE AS WELL AS THE DECAY HEAT ON THE MELTDOWN PROCESS OF A TYPICAL LWR, FOLLOWING A HYPOTHETICAL LOCA. THE AMOUNT OF WATER AT THE BOTTOM OF THE REACTOR VESSEL IS ALSO A MAJOR INFLUENCE BECAUSE IT IS THE SOURCE FOR THE STEAM FLOWING THROUGH THE CORE. THE INFLUENCE OF THESE FACTORS ON THE COURSE OF A LOCA IS DISCUSSED, AND RESULTS ARE GIVEN.

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

REACTOR, LWR + ACCIDENT, LOSS OF COOLANT + ACCIDENT, CORE DISRUPTIVE + ACCIDENT, FUEL SLUMP + COMPUTER PROGRAM, DIGITAL + SUBCHANNEL ANALYSIS + BLOWDOWN + GERMANY + FUEL MELTDOWN + FUEL COOLANT INTERACTION

116818
BLAUEL JG + STAHN D + KALTHOFF JF
INVESTIGATION INTO BRITTLE FRACTURE BEHAVIOR OF THICK-WALLED HOLLOW CYLINDERS SUBJECTED TO THERMAL SHOCK (IN GERMAN)
INSTITUT FUR FESTKORPERMECHANIK, GERMANY
BMFT RS 61 +. 88 PPS. 28 FIGS. SEPT. 1975

MODEL EXPERIMENTS WERE CARRIED OUT TO INVESTIGATE BRITTLE FRACTURE BEHAVIOR OF A REACTOR PRESSURE VESSEL DURING AN EMERGENCY CORE COOLING. UNIFORMLY HEATED HOLLOW CYLINDERS MADE OF GLASS WITH WELL DEFINED PRECRACKS WERE SHOCK-COOLED BY FLOWING WATER. TEMPERATURES AND TIMES TO FRACTURE WERE MEASURED AND THE FRACTURE SURFACES CREATED BY ONE OR A SERIES OF SHOCK LOADINGS WERE INSPECTED AND EVALUATED AFTER THE EXPERIMENT. TESTS PROVIDED INFORMATION ABOUT THE TIME-DEPENDENT THREE-DIMENSIONAL CRACK EXTENSION WHICH CAN BE INTERPRETED BY THEORETICAL-NUMERICAL ANALYSES ON THE BASIS OF LINEAR-ELASTIC FRACTURE MECHANICS. THE APPLICABILITY OF THESE RESULTS FOR A REAL REACTOR SITUATION IS DISCUSSED UNDER SPECIAL CONSIDERATION OF THE MATERIAL CONDITIONS.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

*BRITTLE FRACTURE + PRESSURE VESSELS + *MODEL TESTING + *EMERGENCY COOLING + GLASS + CRACK

125266
OTTO W + KOTTHOFF K
COMPARISON OF COMPUTER PROGRAMS FOR THE RELIABILITY ANALYSIS OF NUCLEAR POWER PLANTS, FINAL REPORT (IN GERMAN)
INSTITUT FUR REAKTORSICHERHEIT DER T.U.V. E.V., F.R. GERMANY
GERRSR-31 + RS 172 +. APPROX. 200 PPS. FIGS. APRIL 1976

DESCRIBES AND COMPARES COMPUTER PROGRAMS USED IN FAULT TREE ANALYSIS, RELABILITY ANALYSIS, AVAILABILITY STUDIES, AND FAILURE RATE STUDIES. USER DIRECTIONS AND DESCRIPTIONS OF OUTPUT ARE PRESENTED.

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

JACOBS + GERMANY + COMPARISON + COMPUTER PROGRAM + RELIABILITY ANALYSIS + FAULT TREE ANALYSIS + AVAILABILITY

126153
STATUS REPORT - REACTOR PRESSURE VESSEL, VOLUME 3. PRESENTATION OF THE SIGNIFICANT RESULTS OF THE HEAVY SECTION STEEL TECHNOLOGY PROGRAM, SUPPLEMENTED BY THE PERTINENT LITERATURE: THROUGH 1973 (IN GERMAN)
INSTITUT FUR REAKTORSICHERHEIT DER T.U.V., E.V., F.R. GERMANY
GERRSR-104 + BERICHT SB 4 +. 170 PPS. FIGS. DEC. 1976

PRESENTS THE SIGNIFICANT RESULTS OF THE HEAVY SECTION STEEL TECHNOLOGY PROGRAM AS OF THE SPRING OF 1973. STUDIES SHOWED THAT WITH A WALL THICKNESS OF 12 INCHES, TOUGHNESS DECREASES IN THE DIRECTION OF THE CENTER OF THE CROSS-SECTION. FOR A WALL THICKNESS OF 6 INCHES, THE EFFECT IS BARELY OBSERVED. IRRADIATION IMPROVES TENSILE STRENGTH AND THE IRRADIATION TESTS SHOWED THAT HARDENING CAN BE REDUCED AT MOST BY 70% BY HEAT TREATMENT AT 343 DEGREES C. OTHER TOPICS INCLUDE CRITERIA DEVELOPMENT FOR FRACTURE BEHAVIOR, STRESS INTENSITY FACTORS, FRACTURE MECHANICS FORMULAS, CRACK ARREST FRACTURE TOUGHNESS, AND INFLUENCE OF PRE-STRESSING ON FRACTURE BEHAVIOR.

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

GERMANY + R AND D PROGRAM + *HSST + *FRACTURE TOUGHNESS + STEEL + PRESSURE VESSELS + METALLURGY + RADIATION EFFECT + TENSILE PROPERTY + HEAT TREATMENT

131755
RECENT WORLDWIDE DEVELOPMENTS IN RADIOLOGICAL PROTECTION FROM THE UTILIZATION OF NUCLEAR TECHNOLOGY
INSTITUT FUR REAKTORSICHERHEIT DER T.U.V., F.R. GERMANY
GERRSR-148 + IRS-T-29 +. 163 PPS. TABS. FIGS. DEC. 1976

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

131755 *CONTINUED*
AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161

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

126083
FINAL REPORT LWR DEVELOPMENT OF THE HYDRODYNAMIC BEARINGS FOR SWR ROTATING PUMPS
KSB RESEARCH & CONSTRUCTION. F.R. GERMANY
GERRSR-90 +. 75 PPS. 111 TABS. NO DATE

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

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

GERMANY + R AND D PROGRAM + REACTOR. LWR + BEARING + PUMPS + HYDRODYNAMIC ANALYSIS + COMPARISON. THEORY AND EXPERIENCE

129683
THEORETICAL AND EXPERIMENTAL EXAMINATION REGARDING THE CARRYING CAPACITY OF HYDRODYNAMIC CYLINDRICAL RADIAL BEARINGS
KLEIN. SCHANZLIN & BECKER. GERMANY
GERRSR-134 + RB 78/K 40 +. 144 PPS. TABS. FIGS. NO DATE

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

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

*HYDRODYNAMIC ANALYSIS + *BEARING + TESTING + NUMERICAL METHOD + DATA COLLECTION + GERMANY + FOREIGN EXCHANGE + COMPARISON. THEORY AND EXPERIENCE

131055
SCHWEICKERT H
REACTOR SAFETY RESEARCH PROGRAM FINAL REPORT. PART A: PRESSURE OSCILLATIONS (IN GERMAN)
KRAFTWERK UNION. ERLANGEN. F.R. GERMANY
GERRSR-101 + BMFT RS 78 C +. 146 PPS. 19 TABS. 74 FIGS. 3 REFS. 1976

PRESENTS INFORMATION ON THE EXPERIMENTS CONDUCTED TO DETERMINE THE FORCES ON THE PRESSURE SUPPRESSION (BWR) TORUS WALLS AND SUPPORTS FOR CONDITIONS AS WOULD OCCUR DURING A LOCA.

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

GERMANY + R AND D PROGRAM + REACTOR. BWR + CONTAINMENT. PRESSURE SUPPRESSION + TORUS + PRESSURE PULSE + PRESSURE TRANSIENT + STRESS STRAIN DATA + FOREIGN EXCHANGE

126215
SCHWEICKERT H
RESEARCH PROGRAM REACTOR SAFETY FINAL REPORT KEYWORD: CONDENSATION IV. PART II PROJECT PART B: CROSSBEAM LOADS (IN GERMAN)
KRAFTWERK UNION. ERLANGEN. F.R. GERMANY
GERRSR-102 + BMFT RS 78 C +. 135 PPS. TABS. FIGS. OCT. 1976

THE INFLUENCE OF WATER TEMPERATURE AND MASS-FLOW-DENSITY ON THE CROSSBEAM POWERS IS INVESTIGATED. THEN THERE ARE INVESTIGATIONS OF HOW FAR TUBES. NOT FIXED TOGETHER BY CROSSBEAMS ARE INFLUENCING EACH OTHER. FURTHERMORE THE INFLUENCE OF THE TUBE-CROSSBEAM IS SHOWN IN CONNECTION WITH A COMPARISON OF THE CROSSBEAM MEASUREMENTS OF A SINGLE TUBE IN A NUCLEAR POWER PLANT AND IN A BIG TANK. FINALLY. THE SUPERPOSED LOADS OF TUBES. CONNECTED BY CROSSBEAMS. ARE DETERMINED.

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

R AND D PROGRAM + WATER + TEMPERATURE + FLOW + TUBING

126645
VOIGT H
SAFETY TECHNOLOGY RESEARCH PROGRAM CONDENSATION AND BOILING RESEARCH FINAL REPORT. KEYWORD: CONDENSATION IV. PART I (IN GERMAN)

126645 *CONTINUED*
KRAFTWERK UNION, ERLANGEN, F.R. GERMANY
GERRSR-100 + BMFT RS 78 B +. APPROX. 260 PPS. TABS. FIGS. REFS. NOV. 1976

PHENOMENA OF ASYMMETRIC POOL SWELL AND VIBRATIONS OF THE STEAM BUBBLES AT THE END OF THE VENT PIPES IN THE BWR-PRESSURE SUPPRESSION SYSTEM DURING A LOCA ARE INVESTIGATED. SEVERAL MODELS ARE DEVELOPED AND TESTED BY COMPARISON WITH EXPERIMENTAL DATA. THE ASYMMETRY OF POOL SWELL IS SIMULATED BY USING AN AIR-STEAM-MIXTURE, THE AIR CONTENT OF WHICH INCREASES WITH THE DISTANCE FROM THE BREAK AREA. THE VIBRATION OF THE STEAM BUBBLE IS CONTROLLED BY A SELF-EXITING MECHANISM BASED ON THE COUPLING OF STEAM FLOW INTO THE BUBBLE, CONDENSATION AT THE PHASE BOUNDARY STEAM-WATER AND THE MOTION OF THE BUBBLE.

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

REACTOR, BWR + CONTAINMENT, PRESSURE SUPPRESSION + VIBRATION + BUBBLE + ACCIDENT, LOSS OF COOLANT + ACCIDENT ANALYSIS + FLOW, TWO PHASE + GERMANY + FOREIGN EXCHANGE

127845
WERNER A + WIELING N
RESEARCH ON THE SELECTIVE CORROSION BEHAVIOR OF LIGHT WATER REACTOR MATERIALS PART 1: LITERATURE STUDY OF CREVICE AND PITTING CORROSION AND CORROSION FATIGUE. (IN GERMAN)
KRAFTWERK UNION ERLANGEN, F.R. GERMANY
GERRSR-122 + BMFT RS 159 +. 108 PPS. 4 TABS. 27 FIGS. 179 REFS. FEB. 1977

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

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

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

127859
ROSLER U
RESEARCH ON THE SELECTIVE CORROSION BEHAVIOR OF LIGHT WATER REACTOR MATERIALS. PART 2 - LITERATURE STUDY OF STRESS-INDUCED CORROSION, INTERGRANULAR STRESS CORROSION AND HYDROGEN INDUCED STRESS-CORROSION CRACKING (IN GERMAN)
KRAFTWERK UNION, ERLANGEN, F.R. GERMANY
GERRSR-123 + BMFT RS 159 +. 99 PPS. 7 TABS. 39 FIGS. 156 REFS. MARCH 1977

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

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

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

127857
ROSLER U
RESEARCH ON THE SELECTIVE CORROSION BEHAVIOR OF LIGHT-WATER REACTOR MATERIALS. PART 3 - RESEARCH PROGRAM (IN GERMAN)
KRAFTWERK UNION, ERLANGEN, F.R. GERMANY
GERRSR-124 + BMFT RS 159 +. 27 PPS. 10 REFS. MARCH 1977

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

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

CORROSION + REACTOR, LWR + MATERIAL + CRACK + STRESS CORROSION + SAFETY MARGIN + FOREIGN EXCHANGE

127858
ROSLER U
RESIDUAL STRESSES IN AUSTENITIC STRIPCLAD REACTOR PRESSURE VESSEL STEEL (IN GERMAN)
KRAFTWERK UNION, ERLANGEN, F.R. GERMANY
GERRSR-125 + BMFT RS 91 +. 179 PPS. 58 FIGS. 78 REFS. MARCH 1977

127858 *CONTINUED*

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

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

PRESSURE VESSELS + STEEL, STAINLESS + STRESS + WELDING + CLADDING + FOREIGN EXCHANGE

127860
BENDER N + HOFER G + MIKSCH M
RESIDUAL STRESSES IN AUSTENITIC STRIPCLAD REACTOR PRESSURE VESSEL STEEL. PART 2 - RESIDUAL STRESSES IN THE VICINITY OF BONDED STRIP CLADDING (IN GERMAN)
KRAFTWERK UNION, ERLANGEN, F.R. GERMANY
GERRSR-126 + BMFT RS 91 +. 108 PPS, 9 TABS, 49 FIGS, 16 REFS, APRIL 1977

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

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

PRESSURE VESSELS + STEEL, STAINLESS + STRESS + WELDING + CLADDING + STRESS ANALYSIS + X-RAY + FOREIGN EXCHANGE

129671
NEEB KH
REDUCTION OF PRIMARY SYSTEM CONTAMINATION THROUGH THE INSTALLATION OF ELECTROMAGNETIC FILTERS (IN GERMAN)
KRAFTWERK UNION, ERLANGEN, F.R. GERMANY
GERRSR-127 + BMFT-RS 171 +. 91 PPS, 14 TABS, 19 FIGS, 13 REFS, APRIL 1977

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

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

*REACTOR, PWR + *FILTER, HIGH TEMPERATURE + MAIN COOLING SYSTEM + *CORROSION + TEST, FILTER + CONTAMINATION + IRON + FOREIGN EXCHANGE

128392
HERZOG J + ZELLINSKY G + CHRIST R + SCHNEIDER K
DECONTAMINATION AND TRANSPORTATION PROBLEMS RELATED TO THE DECOMMISSIONING OF NUCLEAR POWER PLANTS AND THE ELIMINATION OF FAILURE CONSEQUENCES. MAIN WORK, VOL. 1: DECOMMISSIONING, JAN. 1, 1975-JUNE 30, 1976
TRANSNUKLEAR GMBH, F.R. GERMANY
GERRSR-128 K+ NUKEM-312 +. 127 PPS, 20 TABS, 23 FIGS, 28 REFS, MAY 1977

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

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

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

128393
HERZOG J + ZELLINSKY G + CHRIST R + SCHNEIDER K
DECONTAMINATION AND TRANSPORTATION PROBLEMS RELATED TO THE DECOMMISSIONING OF NUCLEAR POWER PLANTS AND THE ELIMINATION OF FAILURE CONSEQUENCES. (APPENDIX 1) - PRINCIPLES, JAN. 1, 1975-JUNE 30, 1976
TRANSNUKLEAR GMBH, F.R. GERMANY
GERRSR-129 + NUKEM-312 +. 61 PPS, 10 TABS, 7 FIGS, MAY 1977

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

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

128393 *CONTINUED*
SHIPPING CONTAINER + DECOMMISSIONING + FOREIGN EXCHANGE + *TRANSPORTATION AND HANDLING + REGULATION, IAEA

128394
CHRIST R + SCHNEIDER K + SCHRODER R
DECONTAMINATION AND TRANSPORTATION PROBLEMS RELATED TO THE DECOMMISSIONING OF NUCLEAR POWER PLANTS AND THE ELIMINATION OF FAILURE CONSEQUENCES. APPENDIX II - SHIPMENT OF HEAVY COMPONENTS AND IRRADIATED SODIUM. JAN. 1, 1975-JUNE 30, 1976
TRANSNUKLEAR GMBH, F.R. GERMANY
GERRSR-130 + NUKEM-312 +. 62 PPS. TABS. FIGS. MAY 1977

CHAPTER 1 DEALS WITH THE SHIPMENT OF ACTIVATED AND/OR CONTAMINATED HEAVY COMPONENTS FROM LIGHT WATER REACTORS FROM DECOMMISSIONING OPERATIONS; THE SECOND ONE THE TRANSFER AND OFF-SITE REMOVAL OF IRRADIATED SODIUM FROM SODIUM-COOLED FAST REACTORS.

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

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

139833
MAYINGER F + REINEKE H-H + SCHRAMM R + STEINBERNER U
BEHAVIOR OF CORE MELT DURING A HYPOTHETICAL ACCIDENT - ANNUAL REPORT 1976 (IN GERMAN)
INSTITUT FUR VERFAHRENSTECHNIK, TECHNISCHE UNIV. HANNOVER, F.R. GERMANY
BMFT RS 166 + GERRSR-105 +. 128 PPS. FIGS. 18 REFS. 1976

THE INVESTIGATIONS PERFORMED IN 1976 WERE ESSENTIALLY CONCERNED WITH THE DEVELOPMENT OF A HEAT TRANSFER MODEL OF THE INTERACTIONS BETWEEN MOLTEN CORE AND CONCRETE. BESIDES THIS THE EXPERIMENTS NECESSARY FOR THE VERIFICATION OF THE CODE, COMPUTING THE HEAT TRANSFER IN THE REACTOR PRESSURE VESSEL, WERE FINISHED. ALSO EXPERIMENTS WERE CARRIED OUT TO INVESTIGATE THE HEAT TRANSFER OF A TURBULENT BOUNDARY LAYER AT THE WALLS.

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

*ACCIDENT, CORE DISRUPTIVE + CORE MELTDOWN + CONCRETE + *HEAT TRANSFER ANALYSIS + PRESSURE VESSELS + *ANALYTICAL MODEL

128956
VOLF K
PARAMETER IDENTIFICATION OF A BWR NUCLEAR PLANT MODEL FOR USE IN OPTIMAL CONTROL
TECHNISCHE UNIVERSITAT MUNCHEN, F.R. GERMANY
GERRSR-89 + MRR-155 +. 100 PPS. 3 TABS. 23 FIGS. 46 REFS. FEB. 1976

THE PROBLEM CONSIDERED IS THE MODELING OF A NUCLEAR POWER PLANT FOR THE DEVELOPMENT OF AN OPTIMAL CONTROL SYSTEM OF THE PLANT. CURRENT SYSTEM IDENTIFICATION CONCEPTS, COMBINING INPUT/OUTPUT INFORMATION WITH A-PRIORI STRUCTURAL INFORMATION ARE EMPLOYED. TWO OF THE KNOWN PARAMETER IDENTIFICATION METHODS, I.E., A LEAST SQUARES METHOD AND A MAXIMUM LIKELIHOOD TECHNIQUE, ARE STUDIED AS WAYS OF PARAMETER IDENTIFICATION FROM MEASUREMENT DATA. A LOW ORDER STATE VARIABLE STOCHASTIC MODEL OF A BWR NUCLEAR POWER PLANT IS PRESENTED AS AN APPLICATION OF THIS APPROACH. THE MODEL CONSISTS OF A DETERMINISTIC AND A NOISE PART.

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

REACTOR, BWR + CONTROL SYSTEM + *OPTIMIZATION + *REACTOR CONTROL + NOISE ANALYSIS + COMPUTER PROGRAM + SIMULATION + REACTOR DYNAMICS + ANALYTICAL MODEL + MATHEMATICAL TREATMENT + FOREIGN EXCHANGE

121372
WOHAK B + MEINLSCHMIDT G
AN INFORMATION SYSTEM FOR THE ESTABLISHMENT OF EMPERICALLY-TESTED RELIABILITY DATA FOR POWER PLANT COMPONENTS (IN GERMAN)
TECHNISCHE UNIVERSITAT MUNCHEN, F.R. GERMANY
GERRSR-75 + MRR-159 +. 100 PPS. FIGS. REFS. JUNE 1976

THE INCREASING IMPORTANCE OF SAFETY AND AVAILABILITY OF POWER PLANT UNITS DEMANDS MORE DETAILED AND PROVEN RELIABILITY DATA FOR PLANT COMPONENTS THAN OPERATIONAL EXPERIENCE COULD PROVIDE TILL NOW. IN THE FEDERAL REPUBLIC OF GERMANY AN EXTENSIVE COLLECTION OF FIELD DATA FROM A CONVENTIONAL POWER PLANT HAS BEEN COMPILED SINCE 1972. BY THIS REPORT THE AUTHORS PRESENT A COMPUTER-BASED INFORMATION SYSTEM FOR THE ESTABLISHMENT OF EMPIRICALLY TESTED RELIABILITY DATA FOR POWER PLANT COMPONENTS.

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

GERMANY + *DATA COLLECTION + *RELIABILITY, COMPONENT + FAILURE, COMPONENT + INFORMATION RETRIEVAL + JACOBS

121374

121374 *CONTINUED*
BUTTNER WE
SAFETY COMPARISON OF A HARD-WIRED DYNAMIC REACTOR PROTECTION SYSTEM WITH A COMPUTERIZED PROTECTION SYSTEM (PART 1) (IN GERMAN)
TECHNISCHE UNIVERSITAT MUNCHEN, F.R. GERMANY
GERRSR-76 + MRR-161 +. 80 PPS. 9 FIGS. 52 REFS. JULY 1976

THE PRESENTED RESEARCH COMPARES A CONVENTIONAL HARD-WIRED DYNAMIC REACTOR PROTECTION SYSTEM WITH A COMPUTERIZED PROTECTION SYSTEM. IN THE COMPARISON THE UNEQUIVOCALLY SAFETY-ORIENTED PROTECTION ACTIONS ARE CONSIDERED WITH RESPECT TO THE SAFETY TECHNIQUES. IN THE FIRST PART THE DIFFERENT STRUCTURES OF BOTH SYSTEMS AND THE METHOD OF VERIFICATION FOR THEIR FUNCTIONAL SAFETY WILL BE DESCRIBED. IN THE SECOND PART THE MEAN UNAVAILABILITY IN CASE OF DEMAND FOR BOTH SYSTEMS UNDER DEFINED CONDITIONS WILL BE DETERMINED.

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

COMPARISON + *REACTOR PROTECTION SYSTEM + COMPUTER, DIGITAL + COMPUTER CONTROL + *EQUIPMENT DESIGN + COMPUTER PROGRAM + JACOBS

120342
WARNEMUNDE R
EXCURSIONS IN SOLUTIONS OF FISSIONABLE MATERIALS AS A CONSEQUENCE OF THE STEP-INPUT OF REACTIVITY (IN GERMAN)
TECHNICAL UNIVERSITY, MUNICH, GERMANY
MRR-162 +. 55 PPS. 13 TABS. 21 FIGS. AUG. 1976

THE PROMPT CRITICAL POINT KINETIC EQUATIONS ARE SOLVED FOR LINEAR AND QUADRATIC TEMPERATURE FEED BACK FOR 'STEP INPUT' REACTIVITIES IN FRAME OF THE HANSEN-FUCHS-MODEL. THE INFLUENCE OF THE PRECURSOR NEUTRONS ON ENERGY IN THE POWER MAXIMUM CAN BE CALCULATED IN THE FIRST APPROXIMATION. COMPARISONS OF AIREK-CODE CALCULATIONS AND KEWB EXPERIMENTS CONFIRM THE VALIDITY OF NEGLECTING THE PRECURSOR NEUTRONS FOR INPUT REACTIVITY R > OR = TC $1.2. THE ANALYTICAL CONSIDERATIONS ARE MOSTLY CONSERVATIVE AND AGREE WITH THE KEWB EXPERIMENTS WITHIN 30%. FINALLY, THE MAGNITUDE AND INFLUENCE OF THE PARAMETERS ARE DESCRIBED, WHICH GOVERN THE 'STEP INPUT' EXCURSIONS.

AVAILABILITY - INIS SECTION, INTERNATIONAL ATOMIC ENERGY AGENCY, P.O. BOX 590, A-1011 VIENNA, AUSTRIA

*SOLUTION + *EXCURSION, LARGE + ACCIDENT, REACTIVITY + PROMPT CRITICALITY + TEMPERATURE COEFFICIENT + TEMPERATURE REACTIVITY EFFECT + COMPUTER PROGRAM + JACOBS

122796
ULLRICH W + FRISCH W
RESEARCH ON MANAGING THE CONSEQUENCES OF THE FAILURE OF THE REACTOR SHUTDOWN SYSTEM (ATWS) AND OTHER SELECTED SAFETY ACTIVITIES (IN GERMAN)
TECHNISCHE UNIVERSITAT MUNCHEN, F.R. GERMANY
GERRSR-83 + MRR-163 +. 175 PPS. FIGS. SEPT. 1976

THE FIRST THREE CHAPTERS DESCRIBE THE ATWS PROBLEMS. AN OUTLINE OF THE PROCEDURE ON HOW TO EXECUTE THE ORDERS AND A COMPLETE DOCUMENTATION OF THE INTERIM REPORTS. CHAPTERS 4 AND 5 EVALUATE THE AMERICAN LITERATURE AND THE DOCUMENTS BEING PREPARED BY GERMAN MANUFACTURERS. COMPUTER PROGRAMS USED BY THE KRAFTWERK UNION ARE DESCRIBED AND MODELS ARE COMPARED. CHAPTERS 6 AND 7 DESCRIBE THE MOST ESSENTIAL RESULTS OF ATWS FOR LWRS AS WELL AS A REACTOR PHYSICAL STUDY ON THE 'MULTIPLE STUCK ROD' PROBLEM. CHAPTER 8 GIVES THE RELIABILITY OR THE NON-AVAILABILITY OF IMPORTANT SAFETY SYSTEMS. COMMON MODE FAILURES ARE DEALT WITH IN CHAPTER 9. IN CHAPTER 10 CONCLUSIONS ARE DRAWN FROM ALL ANALYSES AND PROPOSALS ARE MADE FOR FUTURE PROCEEDINGS.

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

REACTOR PROTECTION SYSTEM + *ACCIDENT, ATWS + *ACCIDENT ANALYSIS + GERMANY + FAILURE, COMMON MODE + DESIGN STUDY

124735
DRESSLER E
THEORETICAL BACKGROUND OF THE COMPUTER CODES SAFTL AND CRESS AND CALCULATION OF THE RELIABILITY OF SYSTEMS
TECHNISCHE UNIVERSITAT MUNCHEN, F.R. GERMANY
GERRSR-85 + MRR-164 +. 51 PPS. 3 FIGS. 15 REFS. SEPT. 1976

A BRIEF PRESENTATION OF THE METHODS USED IN THE COMPUTER CODE SAFTL AND CRESS. APART FROM THE SPECIFIC PROCEDURES OF THESE CODES, THE BACKGROUND NECESSARY FOR UNDERSTANDING THEM IS DESCRIBED. THEY CAN BE USED TO DETERMINE CONFIDENCE LIMITS AND RELIABILITY MEASUREMENTS OF REACTOR SYSTEMS.

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

COMPUTER PROGRAM + RELIABILITY, SYSTEM + MONTE CARLO + JACOBS

123256
SADTLER E
IDENTIFICATION OF THE VIBRATION BEHAVIOR AND ESTIMATION OF SYSTEM PARAMETERS FOR PRESSURE VESSEL AND INTERNALS

123256 *CONTINUED*
OF PWR BIBLIS-A (IN GERMAN)
TECHNISCHE UNIVERSITAT MUNCHEN, F.R. GERMANY
GERRSR-86 + MRR-165 +. 75 PPS, FIGS, OCT .1976

PREVIOUS INVESTIGATIONS OF THE REACTOR VIBRATIONS HAVE DEMONSTRATED THAT THE PRESSURE VESSEL AND ITS INTERNALS PERFORM PENDULAR MOTICNS. IDENTIFICATION OF THE VIBRATIVE BEHAVIOR BY USE OF MODERN ESTIMATION METHODS IS DESCRIBED. BASED ON A DOUBLE-PENDULUM LUMPED PARAMETER MODEL, THE STATE VECTOR ENLARGED BY UNKNOWN SYSTEM PARAMETERS IS ESTIMATED FROM NOISY PRE-OPERATIONAL MEASUREMENTS OF THE PWR BIBLIS-A. FOR THIS TASK A MAXIMUM-A-POSTERIORI (MAP) IDENTIFICATION FILTER ALGORITHM WAS EMPLOYED.

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

VIBRATION + REACTOR, PWR + ANALYTICAL MODEL + JACOBS

127268
NIECKAU E
CONCLUSIONS OF THE RELIABILITY ANALYSIS OF REACTOR PROTECTION SYSTEMS
TECHNISCHE UNIVERSITAT MUNCHEN, F.R. GERMANY
GERRSR-87 + MRR-166 +. 33 PPS, 2 FIGS, 34 REFS, NOV. 1976

SUMMARIZES THE EXPERIENCES WHICH ARE GAINED WITH THE RELIABILITY ANALYSIS OF THE REACTOR PROTECTION SYSTEMS. CONSIDERATIONS ARE GIVEN TO THE PROCEDURE OF THESE ANALYSES AND ALSO TO THE RULES FOR POSSIBLE OMISSIONS. FURTHERMORE, PARAMETERS INFLUENCING THEIR CONSTRUCTION ARE DISCUSSED. THIS REPORT DEALS PARTICULARLY WITH THE ANALYSES OF HARD-WIRED REACTOR PROTECTION SYSTEMS, WHEREAS THE PROTECTIVE COMPUTERS ARE SIMPLY MENTIONED.

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

*RELIABILITY ANALYSIS + RELIABILITY, SYSTEM + *REACTOR PROTECTION SYSTEM + GERMANY + FOREIGN EXCHANGE

127118
WAHBA AB
CALCULATION OF THE TIME DEPENDENT LOCAL HEAT TRANSFER COEFFICIENT ON THE REWETTING OF HEATED INSULATED TUBES
TECHNISCHE UNIVERSITAT MUNCHEN, F.R. GERMANY
GERRSR-88 + MRR-167 +. 68 PPS, 35 FIGS, 6 REFS, NOV. 1976

THE COMPUTER PROGRAM INSTHTC WAS DEVELOPED TO CALCULATE TIME-DEPENDENT LOCAL HEAT TRANSFER COEFFICIENTS. THE MODEL WAS APPLIED TO THE REWETTING EXPERIMENTS PERFORMED IN CONNECTION WITH THE BMFT-RESEARCH PROJEKT RS 62. IN THE ADVANCED VERSION INSTHTC 2 THE INSULATION OF THE HEATED TUBE WAS ALSO SIMULATED. THE CALCULATED SURFACE TEMPERATURE OF THE INSULATION AGREES WELL WITH THE ONE DERIVED FROM CALORIC MEASUREMENTS. THE TIME-DEPENDENT HEAT-TRANSFER COEFFICIENT AT THE INNER SURFACE OF THE TUBE WAS CALCULATED USING THE APPROPRIATE SATURATION TEMPERATURE OF THE WATER ACCORDING TO THE MEASURED PRESSURE. THE CAUSES FOR THE OSCILLATION OF THE CALCULATED HEAT-TRANSFER COEFFICIENTS ARE DISCUSSED IN DETAIL.

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

HEAT TRANSFER COEFFICIENT + HEAT TRANSFER, BOILING + COMPUTER PROGRAM, DIGITAL + COMPARISON, THEORY AND EXPERIENCE + OUT OF PILE EXPERIMENT + GERMANY

127623
HORTNER H + NIECKAU E + SPINDLER H
RESULTS OF A RELIABILITY ANALYSIS OF THE ENGINEERED SAFETY FEATURES FOR THE DESIGN BASIS ACCIDENT (IN GERMAN)
TECHNISCHE UNIVERSITAT MUNCHEN, GARCHING, F.R. GERMANY
GERRSR-118 + MRR-168 +. 160 PPS, 8 TABS, 25 FIGS, 61 REFS, DEC. 1976

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

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

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

128494
HAWICKHORST W
SIMULATION EXPERIMENTS OF THE FUNCTIONAL TESTING OF ENGINEERED SAFETY SYSTEMS WITH PROCESS CALCULATIONS. PART 2: RESULTS OF SIMULATION EXPERIMENTS WITH A NEW CONCEPT OF COMPUTERIZED SYSTEM INSPECTION TECHNIQUE (IN GERMAN)
TECHNISCHE UNIVERSITAT MUNCHEN, F.R. GERMANY

128494 *CONTINUED*
GERRSR-142 + NRR-169 +. 173 PPS. 46 FIGS. 60 REFS. DEC. 1976

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

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

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

124988
STURM D
REACTOR PRESSURE VESSELS: A SERIES OF REPORTS (IN GERMAN)
UNIV. OF STUTTGART. F.R. GERMANY
BMFT-IB RS 101(MPA-TASKS 830/1-2. 5-9. 12-13) + GERRSR-42--GERRSR-72 +. VP. FIGS. REFS. 1973-1975

PRESENTS A SERIES OF 31 GERMAN R&D REPORTS BY VARIOUS AUTHORS DESCRIBING SUCH TOPICS AS FASTENING DEVICES FOR LARGE SAMPLES. DEFORMATION AND FRACTURE MECHANICS. WELDING OF LONGITUDINAL SEAMS. WELDING OF CIRCUMFERENTIAL SEAMS. SIMULATION WITH ELECTRON BEAM WELDING. SIMULATED SAMPLES OF SIMULATED WELDS. AND VESSELS OF INTERMEDIATE SIZES.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM. 1717 H STREET. WASHINGTON. D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)

R AND D PROGRAM + PRESSURE VESSELS + STEEL + WELDING + WELDS + FRACTURE TOUGHNESS + DEFORMATION + FASTENER + GERMANY + JACOBS + FOREIGN EXCHANGE

129670
BENZ R + FROHLICH G + UNGER H
LITERATURE STUDIES ON VAPOR EXPLOSIONS (IN GERMAN)
UNIVERSITAT STUTTGART. GERMANY
GERRSR-136 + BMFT-RS 76 +. APPROX. 210 PPS. 33 TABS. 49 FIGS. 274 REFS. FEB. 1976

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

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

BIBLIOGRAPHY + INCIDENT COMPILATION + *GAS + *VAPOR PRESSURE + *EXPLOSION + ANALYTICAL MODEL + COMPUTER PROGRAM + GERMANY + FOREIGN EXCHANGE

129993
BENZ R + FROHLICH G + GOLDAMMER H
THEORETICAL STUDIES OF VAPOR EXPLOSIONS (IN GERMAN)
UNIVERSITAT STUTTGART. F.R. GERMANY
GERRSR-137 + BMFT-RS 76 +. 100 PPS. 1 TAB. 31 FIGS. 14 REFS. APRIL 1976

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

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

FUEL COOLANT INTERACTION + CORE MELTDOWN + ANALYTICAL MODEL + GERMANY + FOREIGN EXCHANGE

129281
BENZ R + SCHWALBE W + UNGER H
AN ESTIMATE OF ENERGY RELEASE AND PRESSURE BUILDUP DURING VAPOR EXPLOSIONS IN REACTOR GEOMETRY (IN GERMAN)
UNIVERSITAT STUTTGART. GERMANY
GERRSR-135 + BMFT-RS 206 +. 130 PPS. 8 TABS. 43 FIGS. 42 REFS. DEC. 1976

THE PRESSURE BUILDUP DURING VAPOR EXPLOSIONS IN LIGHT WATER REACTORS HAS BEEN ESTIMATED BY MEANS OF A SENSITIVITY STUDY APPLYING SIMPLE ENGINEERING METHODS. THE INITIAL DATA REQUIRED FOR THIS PURPOSE AND THE MATERIAL PROPERTIES OF THE CORE MELT HAVE BEEN OBTAINED FROM THE COURSE OF A

129281 *CONTINUED*
POSTULATED, HYPOTHETICAL CORE MELTDOWN ACCIDENT. FOR THE ESTIMATION THE BEHAVIOR OF THE PRESSURE WITH RESPECT TO TIME HAS BEEN DIVIDED INTO TWO PHASES. A VERY SHORT, BUT HIGHLY TRANSIENT ONE FOLLOWED BY A LONGER, QUASISTATIC PHASE.

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

ACCIDENT ANALYSIS + EXPLOSION + ACCIDENT MODEL + VAPOR PRESSURE + REACTOR, SAFETY RESEARCH + THEORETICAL INVESTIGATION + FOREIGN EXCHANGE

131128
BENZ R + FROHLICH G + GOLDAMMER H + HOHMANN H
THEORETICAL AND EXPERIMENTAL INVESTIGATIONS OF VAPOR EXPLOSIONS (IN GERMAN)
UNIVERSITAT STUTTGART, F.R. GERMANY
EUR/C-IS/116/77D + GERRSR-116 +. 138 PPS, FIGS, REFS, JAN. 1977

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

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

GERMANY + R AND D PROGRAM + *FUEL COOLANT INTERACTION + ANALYTICAL MODEL + THERMAL HYDRAULIC ANALYSIS + *WATER VAPOR + GAS + EXPLOSION + SHOCK WAVE + METAL WATER REACTION + FOREIGN EXCHANGE

3. JAPANESE SAFETY RESEARCH REPORTS

THE FOLLOWING IS A LISTING OF MICROFICHED LIGHT-WATER REACTOR SAFETY RESEARCH REPORTS RECEIVED FROM JAPAN DURING THE PERIOD 1975-1977 UNDER THE TECHNICAL EXCHANGE AGREEMENT.

125249
STUDY ON THE EFFICIENCY OF THE BWR TOP SPRAY EMERGENCY COOLING WITH AN 8X8 SIMULATED FUEL ROD BUNDLE
HITACHI LTD., JAPAN
JPNRSR-67 + REPORT SI-STAE-59 +. 66 PPS. TABS. FIGS. 10 REFS. SEPT. 1976

THE EFFICIENCY OF THE BWR TOP SPRAY EMERGENCY COOLING WAS EXAMINED WITH AN 8X8 SIMULATED FUEL ROD BUNDLE AND JET PUMP BYPASS OVER THE RANGE OF CONDITION EXPECTED IN A LOSS-OF-COOLANT ACCIDENT. IT WAS FOUND THAT WATER HEAD ON THE UPPER TIE PLATE AND THE DYNAMIC PRESSURE OF THE SPRAY SCARCELY AFFECTED THE COUNTER CURRENT FLOW. A COUNTER-CURRENT FLOW LIMITING (CCFL) CORRELATION AGREED WELL WITH EXPERIMENTAL RESULTS. CONVECTIVE HEAT TRANSFER COEFFICIENTS DURING TOP SPRAY COOLING WERE SCARCELY AFFECTED BY SPRAY TEMPERATURE, BYPASS OF STEAM FROM THE JET PUMP, OR STEAM FLOW FROM THE LOWER PLENUM, AND WERE LARGER THAN ONES USED IN CURRENT LOCA ANALYSIS.

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

ACCIDENT. LOSS OF COOLANT + REACTOR. BWR + EMERGENCY COOLING SYSTEM + HEAT TRANSFER COEFFICIENT + OUT OF PILE EXPERIMENT + JACOBS + JAPAN + CORE SPRAY + COMPARISON. THEORY AND EXPERIENCE

137446
EXPERIMENTAL RESEARCH ON MUTUAL INTERACTIONS BETWEEN NUCLEAR REACTOR STRUCTURES AND THE GROUND FOUNDATION (IN JAPANESE)
JAPAN ARCHITECTURAL SOCIETY
JPNRSR-61 +. 140 PPS. TABS. FIGS. REFS. DEC. 1975

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

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

*INTERACTION. SOIL AND STRUCTURE + *INTERACTION. FOUNDATION AND STRUCTURE + POWER PLANT. NUCLEAR + R AND D PROGRAM + EXPERIMENT + FOREIGN EXCHANGE

113032
FURUTA T + OTOMO T + HASHIMOTO M
DEFORMATION AND INNER OXIDATION OF THE FUEL ROD IN A LOCA CONDITION (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-6339 +. 17 PPS. 12 FIGS. 6 REFS. DEC. 1975

IN ORDER TO STUDY BEHAVIORS OF THE FUEL ROD IN A LOSS-OF-COOLANT ACCIDENT, FOUR KINDS OF PRE-PRESSURIZED SHAM FUEL RODS FILLED WITH AL(SUB 2)O(SUB 3) PELLETS TO SIMULATE UO2 WERE HEATED IN STEAM UNDER HYPOTHETICAL LOCA CONDITIONS. CIRCUMFERENTIAL STRAIN OF THE SHAM FUEL ROD INCREASES WITH RISE OF THE INITIAL PRESSURE, SINCE BURST TEMPERATURE OF THE SINGLE SHAM FUEL ROD DECREASES WITH INCREASE OF THE INITIAL PRESSURE. OXIDE THICKNESS OF THE INNER TUBE WALL IS LARGER THAN THAT OF THE OUTER TUBE WALL, WHEN AMOUNT OF THE SUPPLIED STEAM TO THE INNER TUBE WALL IS LESS. THE THICKER COARSE INNER OXIDE FILM CONSISTS OF MONOCLINIC AND TETRAGONAL ZIRCONIA, WHILE THE THINNER OUTER OXIDE FILM IS OF MONOCLINIC ZIRCONIA ALONE. THE LONGITUDINAL INNER OXIDE FILM IS OBSERVED WITHIN 76 MM OF THE RUPTURE.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER. P.O. BOX 62. OAK RIDGE. TENN. 37830

ACCIDENT. LOSS OF COOLANT + OXIDATION + URANIUM OXIDE + STRUCTURAL INTEGRITY + FUEL ELEMENTS

119700
SHIBA M + ADACHI H + NAMATAME K
EXPERIMENTAL PROGRAM ROSA-II (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INSTITUTE. TOKAI
JAERI-M-6362 +. 110 PPS. 73 FIGS. DEC. 1975

THE EXPERIMENTAL PROGRAM ROSA-II IS CARRIED OUT TO STUDY THE THERMAL AND HYDRODYNAMIC BEHAVIORS IN THE PRIMARY COOLING SYSTEM OF A PRESSURIZED WATER REACTOR IN A LOSS-OF-COOLANT ACCIDENT; THE ROSA-II (RIG OF SAFETY ASSESSMENT) IS AN INTEGRAL TEST FACILITY. CONCERNING THE PROGRAM, ITS PURPOSE, THE FACILITY, AND REQUIREMENTS FOR THE EXPERIMENT ITSELF REVEALED BY A PRE-ANALYSIS, ARE DESCRIBED.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER. P.O. BOX 62. OAK RIDGE. TENN. 37830

*THERMAL HYDRAULIC ANALYSIS + EXPERIMENT + ACCIDENT. LOSS OF COOLANT + EMERGENCY COOLING + REACTOR. PWR + MAIN COOLING SYSTEM

116411
TAKEDA T
CALCULATION OF THE ADDITIONAL FPS RELEASE THROUGH A DEFECT HOLE ON THE CLADDING OF A FUEL ROD (CODE CCDAC-ARFP) (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INSTITUTE. TOKYO
JAERI-M-6399 +. 22 PPS. 12 FIGS. 1976

116411 *CONTINUED*
USING SIMPLE INPUT, THE CODE CALCULATES THE RELASED RADIONUCLIDE YIELDS IN COMPLEX DECAY CHAINS. THE RELEASE BEHAVIOR OF FISSION PRODUCTS DEPENDS ON THE BEHAVIOR OF WATER (OR STEAM) THROUGH THE HOLE IN THE FUEL-ROD CLADDING. THE MECHANISM OF THE ADDITIONAL RELEASE BEHAVIOR, E.G., THE BEHAVIOR AFTER INCREASE OF THE COOLANT PRESSURE, IS PARTIALLY CLARIFIED.

AVAILABILITY - INIS SECTION, INTERNATIONAL ATOMIC ENERGY AGENCY, P.O. BOX 590, A-1011 VIENNA, AUSTRIA

COMPUTER PROGRAM + CLADDING + *FISSION PRODUCT RELEASE + FUEL ROD + JACOBS

116625
FURUTA T + KAWASAKI S + HASHIMOTO M + OTOMO T
FACTORS INFLUENCING DEFORMATION AND WALL THICKNESS OF FUEL RODS UNDER A LOSS-OF-COOLANT ACCIDENT (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INSTITUTE, TOKAI
JAERI-M-6542 +. 30 PPS, 28 FIGS, APRIL 20, 1976

THE DEFORMATION OF FUEL RODS UNDER HYPOTHETICAL LOCA CONDITIONS WAS EXPERIMENTALLY INVESTIGATED BY USING FOUR TYPES OF PRE-PRESSURIZED SHAM FUEL RODS FILLED WITH AL2O3 PELLETS TO SIMULATE UO2. THE EFFECTS OF HEATING METHOD, HEATING RATE, ATMOSPHERE, AND PRESENCE OF PRE-OXIDIZED FILM ARE DISCUSSED.

AVAILABILITY - INIS SECTION, INTERNATIONAL ATOMIC ENERGY AGENCY, P.O. BOX 590, A-1011 VIENNA, AUSTRIA

ACCIDENT, LOSS OF COOLANT + DEFORMATION + JAPAN + HEAT TRANSFER EXPERIMENT + FUEL ELEMENTS + JACOBS

119183
HONMA K + HASHIMOTO M + FURUTA T
BEHAVIOR OF INNER SURFACE OXIDATION OF THE ZIRCALOY CLADDING TUBE IN A LOSS OF COOLANT ACCIDENT (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-6602 +. 22 PPS, 20 FIGS, 14 REFS, JUNE 3, 1976

IN SIMULATION OF INNER SURFACE OXIDATION OF THE FUEL CLADDING IN A LOCA, OXIDATION OF THE ZIRCALOY CLADDING IN STAGNANT STEAM HAS BEEN STUDIED. WEIGHT GAIN OF THE OXIDIZED SPECIMEN IN STAGNANT STEAM IS LESS THAN THAT OF THE OXIDIZED SPECIMEN IN FLOWING STEAM. RING COMPRESSION TEST SHOWS, HOWEVER, THAT DUCTILITY LOSS OF THE OXIDIZED SPECIMEN IN STAGNANT STEAM IS MORE THAN THAT OF THE OXIDIZED SPECIMEN IN FLOWING STEAM DESPITE ITS LESS OXIDATION. THE OXIDIZED SPECIMEN IN STAGNANT STEAM ABSORBS HYDROGEN BUT THE OXIDIZED SPECIMEN IN FLOWING STEAM DOESN'T. THE DUCTILITY IS DECREASED WITH THE HYDROGEN CONTENT.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

CLADDING + ACCIDENT, LOSS OF COOLANT + OXIDATION + STEAM + OUT OF PILE EXPERIMENT + HYDRIDE + JAPAN + REACTOR, LWR + ZIRCALOY + DUCTILITY + JACOBS

119149
OHNISHI N
HEATRN-FEM: A FINITIE-ELEMENT CONDUCTION CODE FOR TRANSIENT THERMAL RESPONSE OF THE FUEL ROD IN AXISYMMETRIC AND PLANE CONFIGURATIONS (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-6665 +. 73 PPS, AUG. 1976

THE MATHEMATICAL METHODS, INPUT AND OUTPUT, AND SOME EXAMPLE CALCULATIONS ARE GIVEN FOR HEATRN-FEM.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

COMPUTER PROGRAM + HEAT TRANSFER, CONDUCTION + MATHEMATICAL STUDY + FUEL ROD + THERMAL TRANSIENT + JACOBS

119150
EXPERIMENT ON PERFORMANCE OF UPPER HEAD INJECTION SYSTEM WITH ROSA-II (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-6707 +. 170 PPS, 16 TABS, 131 FIGS, SEPT. 1976

THERMO-HYDRAULIC BEHAVIOR IN THE PRIMARY COOLING SYSTEM OF A PWR WITH AN UPPER HEAD INJECTION SYSTEM IN A POSTULATED LOSS-OF-COOLANT ACCIDENT WAS STUDIED WITH ROSA-II TEST FACILITY. SIMULATED UHI AND INTERNAL STRUCTURES OF THE PRESSURE VESSEL WERE INSTALLED FOR THE EXPERIMENT. NINE MAXIMUM SIZED DOUBLE-ENDED BREAK TESTS AND ONE MEDIUM SIZED SPLIT BREAK TEST WERE PERFORMED FOR THE COLD-LEG BREAK CONDITION WITH THE FOLLOWING RESULTS: (1) FLUID MIXING IN THE UPPER HEAD IS NOT PERFECT AND (2) COLD WATER INJECTION INTO THE STEAM OR TWO-PHASE FLUID CAUSES VIOLENT DEPRESSURIZATION DUE TO CONDENSATION.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

*COMPUTER PROGRAM + ACCIDENT, LOSS OF COOLANT + REACTOR, PWR + SAFETY INJECTION + FLOW, MIXING + *THERMAL HYDRAULIC ANALYSIS + LPCI + HPCI + JACOBS

120584
SOBAJIMA M + ADACHI H + MATSUMOTO I
ROSA-II TEST DATA REPORT, 5 (RUNS 310, 311, 312, 313, 317) (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO
JAERI-M-6709 +. VP, FIGS, TABS, AUG. 19, 1976

THE OBJECTIVE OF THE ROSA-II (RIG OF SAFETY ASSESSMENT) PROGRAM IS TO INVESTIGATE BLOWDOWN PHENOMENA AND ECCS PERFORMANCE DURING A LOSS-OF-COOLANT ACCIDENT IN A PRESSURIZED WATER REACTOR. THE FACILITY CONSISTS OF A PRESSURE VESSEL WITH A SIMULATED CORE, ONE OPERATING PRIMARY LOOP AND ONE BROKEN PRIMARY LOOP BOTH WITH A STEAM GENERATOR AND A PUMP IN ADDITION TO A PRESSURIZER ATTACHED TO THE OPERATING PRIMARY LOOP. ECCS IS ALSO PROVIDED TO THE FACILITY WITH SEVERAL INJECTION LOCATIONS. IN THE PRESENT REPORT, A PART OF THE ROSA-II EXPERIMENTAL RESULTS ARE GIVEN WITH INTERPRETATIONS OF THE DATA.

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

REACTOR, PWR + ACCIDENT, LOSS OF COOLANT + BLOWDOWN + EMERGENCY COOLING SYSTEM + JAPAN + OUT OF PILE EXPERIMENT + JACOBS

125248
MURAO Y + IGUCHI T + SUDOH Y + SUGIMOTO J + NIITSUMA Y + FUKAYA Y + HIRANO K
REPORT ON SERIES 2A REFLOOD EXPERIMENT
JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO
JPNRSR-64 + JAERI-M-6787 +. 87 PPS, 6 TABS, 79 FIGS, SEPT, 1976

REPORTS ON THE SERIES 2A REFLOOD EXPERIMENT, IN WHICH WATER WAS INJECTED INTO THE CORE AT A CONSTANT FLOW RATE UNDER ATMOSPHERIC PRESSURE. STUDY OF THE DATA SHOWED THAT SEVERAL HEAT TRANSFER PHASES EXIST IN THE CORE, I.E., ADIABATIC, DROPLET-DISPERSED VAPOR FLOW, FILM BOILING, QUENCH, AND NUCLEATE BOILING PHASE. THE RELATION BETWEEN HEAT TRANSFER PHASES AND HEAT TRANSFER COEFFICIENTS IS DISCUSSED QUALITATIVELY.

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

REACTOR, PWR + ACCIDENT, LOSS OF COOLANT + OUT OF PILE EXPERIMENT + EMERGENCY COOLING SYSTEM + CORE REFLOODING SYSTEM + BLOWDOWN + HYDRAULIC EXPERIMENT + JAPAN + JACOBS + THERMAL EXPERIMENT

125001
ISHIKAWA M + TOMII K
QUARTERLY PROGRESS REPORT ON THE NSRR EXPERIMENTS (2) APRIL 1976-JUNE 1976 (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO
JPNRSR-65 + JAERI-M-6790 +. 33 PPS, FIGS, OCT. 1976

TEN TESTS WERE CARRIED OUT DURING THIS QUARTER. EXPERIMENTS IN THE 4 TEST SERIES EXAMINE: THE EFFECT OF FUEL ENRICHMENT BY THE USE OF JPDR-II TYPE TEST FUEL (2.6%) AND STANDARD TEST FUEL (10%) WITH 150 AND 200 CAL/G/UO2 ENERGY DEPOSITION; INFLUENCE OF THE RUN-OUT POWER BY CHANGING SCRAM TIME OF THE NSRR REACTOR; FUEL BEHAVIOR WITH 150 CAL/G/UO2 ENERGY DEPOSITIONS BY REPEATED IRRADIATION TESTS; THE EFFECT OF THE HOLE IN CLADDING ON FUEL FAILURE BY WATERLOGGED FUEL TESTS. THE RESULTS OF THE TESTS, TEST PROCEDURES, AND PRESENT STATUS OF ANALYTICAL WORKS ARE DESCRIBED.

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

JACOBS + *JAPAN + *REACTOR, SAFETY RESEARCH + *IN PILE EXPERIMENT + URANIUM-235 + CONCENTRATION + FUEL, NUCLEAR + REACTOR TRANSIENT + FAILURE, FUEL ELEMENT WATERLOGGING + TRIGA (RR)

125246
MURAO Y + IGUCHI T + SUDOH T + SUGIMOTO J + NIITSUMA Y + FUKAYA Y + HIRANO K
REPORT ON SERIES 2B REFLOOD EXPERIMENT (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO
JPNRSR-71 + JAERI-M-6788 +. 83 PPS, FIGS, REFS, DEC. 1976

REPORTS ON THE SERIES 2B REFLOOD EXPERIMENT, IN WHICH COOLANT WAS INJECTED INTO THE TEST SECTION FROM THE DOWNCOMER AT A CONSTANT PRESSURE OF ONE ATMOSPHERE. THE POWER DENSITY REMAINED CONSTANT WITH HEATER ROD TEMPERATURE UP TO 600 C. THE OBJECTIVES WERE TO EXAMINE QUANTITATIVELY SYSTEM EFFECTS AND TO CHECK THE PERFORMANCE OF THE REFLOOD TEST RIG.

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

ACCIDENT, LOSS OF COOLANT + BLOWDOWN + REACTOR, PWR + EMERGENCY COOLING SYSTEM + CORE REFLOODING SYSTEM + OUT OF PILE EXPERIMENT + HYDRAULIC EXPERIMENT + JACOBS + JAPAN + THERMAL EXPERIMENT

124845
FURUTA T + KAWASAKI S + HASHIMOTO M + OTOMO T
INFLUENCE OF DEFORMATION ON THE SUBSEQUENT STEAM OXIDATION OF ZIRCALOY CLADDING (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO
JPNRSR-72 + JAERI-M-6869 +. 20 PPS, 16 FIGS, 30 REFS, JAN. 1977

124845 *CONTINUED*

THE SUBSEQUENT OXIDATION CAUSED BY DEFORMATION HAS BEEN STUDIED BY FUEL-ROD BURST TESTS AND HIGH-TEMPERATURE TENSILE TESTS IN A STEAM FLOW. IN HIGH-TEMPERATURE TENSILE TESTS, TENSILE DEFORMED SPECIMENS AND NORMAL SPECIMENS WERE EXPOSED, AT THE SAME TIME, TO STEAM AT 1000 C. OXIDATION KINETICS WERE INVESTIGATED USING METALLOGRAPHIC METHODS. APPARENT THICKNESS OF THE GROWING OXIDE DEPENDS ON TEMPERATURE AND DEGREE OF THE DEFORMATION. SEVERE OXYGEN PENETRATION AS WELL AS INCREASE OF THE GROWING OXIDE, IS FOUND IN THE RUPTURE LIPS OF DEFORMED SPECIMENS.

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

ZIRCALOY + CLADDING + ACCIDENT, LOSS OF COOLANT + OXIDATION + STEAM + DEFORMATION + JACOBS

124846

SUZUKI M + KAWASAKI S + FURUTA T

ZIRCALOY-STEAM REACTION AND EMBRITTLEMENT OF THE OXIDIZED ZIRCALOY TUBE UNDER POSTULATED LOSS OF COOLANT ACCIDENT CONDITIONS (IN JAPANESE)

JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO

JPNRSR-73 + JAERI-M-6879 +. 23 PPS, 4 TABS, 13 FIGS, 10 REFS, JAN. 1977

ZIRCALOY-STEAM REACTION AND EMBRITTLEMENT OF THE ZIRCALOY TUBES HAVE BEEN STUDIED UNDER POSTULATED LOCA CONDITIONS. OVER THE REACTION TEMPERATURE RANGE OF 1184 TO 1330 C, THE WEIGHT GAIN, OXIDE LAYER THICKNESS CHANGES FOLLOW A PARABOLIC RATE LAW. DUCTILITY CHANGE OF THE OXIDIZED SPECIMENS WAS EXAMINED AT 100 C BY RING COMPRESSION TEST. THE DUCTILITY DECREASES WITH INCREASE OF THE WEIGHT GAIN. FOR THE SAME WEIGHT GAIN, THE DUCTILITY OF THE SPECIMENS OXIDIZED AT HIGH TEMPERATURE IS LOWER THAN THOSE AT LOW.

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

ZIRCALOY + EMBRITTLEMENT + TUBING + SIMULATION + ACCIDENT, LOSS OF COOLANT + JACOBS

122859

HARAYAMA Y + IZUMI F + FUJITA M

USER'S GUIDE FOR FREG-3: A COMPUTER PROGRAM TO ANALYZE PELLET-CLADDING GAP CONDUCTANCE IN ACCORDANCE WITH FUEL-ROD IRRADIATION HISTORY (IN JAPANESE)

JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO

JPNRSR-74 + JAERI-M-6742 +. 127 PPS, FIGS, REFS, 1976

FREG-3 ESTIMATES THE TEMPERATURE DISTRIBUTION IN A FUEL ROD AND THE STORED ENERGY BASED ON THE DISTRIBUTION. THE TEMPERATURE DISTRIBUTION IS CALCULATED IN ACCORDANCE WITH FUEL-ROD IRRADIATION HISTORY. MECHANICAL PROPERTIES AND MODELS IN HANDLING SPECIFIC PROBLEMS, SUCH AS DENSIFICATION AND RELOCATION, ARE OPTIONAL IN THE PROGRAM. THE OPTIONS ARE TO BE GIVEN BY KEY WORD. IF APPROPRIATE OPTIONS ARE SELECTED, THE PROGRAM IS USED NOT ONLY AS A SAFETY EVALUATION CODE, BUT ALSO AS A BEST EVALUATION CODE.

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

COMPUTER PROGRAM + FUEL, PELLET TYPE + HEAT CONDUCTANCE, FUEL TO CLAD + TEMPERATURE + DISTRIBUTION + FUEL ROD + JACOBS

122858

HARAYAMA Y + IZUMI F

ANALYSIS OF PELLET CENTER TEMPERATURES MEASURED IN HALDEN IFA-224 USING PROGRAM FREG-3 (IN JAPANESE)

JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO

JPNRSR-75 + JAERI-M-6878 +. 29 PPS, 13 FIGS, 15 REFS, 1977

TO VERIFY THE PROGRAM FREG-3, CALCULATIONS WERE COMPARED WITH MEASUREMENTS IN THE HALDEN INSTRUMENTED FUEL ASSEMBLY IFA-224. THE CODE GENERALLY GIVES HIGHER PELLET CENTER TEMPERATURES. ESTIMATES OF THE STORED ENERGY IN THE FUEL RODS IS ON THE SAFE SIDE WHEN BASED ON TEMPERATURE DISTRIBUTION CALCULATED BY FREG-3.

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

COMPUTER PROGRAM + TEMPERATURE + DISTRIBUTION + FUEL ROD + JACOBS

124847

HARAYAMA Y + YAMADA R + IZUMI F

DISPLACEMENT, STRESS AND STRAIN IN TWO-DIMENSIONAL PROBLEMS OF SOLID AND HOLLOW CYLINDERS (IN JAPANESE)

JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO

JPNRSR-76 + JAERI-M-6626 +. 30 PPS, TABS, JULY 1976

AXIALLY SYMMETRICAL TWO-DIMENSIONAL PROBLEMS IN CALCULATING STRESS AND STRAIN FOR SOLID AND HOLLOW CYLINDERS ARE IMPORTANT FOR VARIOUS COMPONENTS IN NUCLEAR TECHNOLOGY. THE PRESENT REPORT ARRANGES THE RELATIONSHIPS OF DISPLACEMENTS, STRESSES AND STRAINS UNDER PLANE STRESS, LAME'S CONDITION AND PLANE STRAIN OF TWO-DIMENSIONAL PROBLEMS, TAKING INTO ACCOUNT ELASTIC AND THERMAL STRAIN.

124847 *CONTINUED*
AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

STRESS ANALYSIS + MATHEMATICAL TREATMENT + CYLINDER + JACOBS

123232
HARAYAMA Y + IZUMI F + YAMADA R
EFFECT OF THE FUEL PELLET DENSIFICATION ON THE FUEL ROD INTERNAL PRESSURE (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO
JPNRSR-77 + JAERI-M-6631 +. 18 PPS, 8 FIGS, JUNE 25, 1976

DENSIFICATION OF LWR FUEL PELLETS IS INDUCED BY IRRADIATION THUS REDUCING THE VOLUME OF THE PELLETS. CALCULATIONS WERE MADE TO DETERMINE THE EFFECT OF THE INCREASED FREE SPACE IN THE PLENUM UPON THE FUEL ROD INTERNAL PRESSURE. NEGLECTING THE ABSORPTION OF THE HELIUM BY THE FUEL PELLETS, THE CALCULATIONS WHICH ARE PRESENTED SHOW THAT THE INCREASE OF THE VOLUME IN THE PLENUM RESULTS IN A DECREASE OF THE INTERNAL PRESSURE OF THE FUEL ROD FOR A SHORTER IRRADIATION TIME.

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

*FUEL DENSIFICATION + *PRESSURE, INTERNAL + FUEL, PELLET TYPE + REACTOR, LWR + REACTOR, BWR + REACTOR, PWR + JAPAN + JACOBS + FUEL ROD

124995
TAKEDA T
MAP OF CALCULATED RADIOACTIVITY OF FISSION PRODUCTS (VOL. I) - MAPS OF RADIOACTIVITY OF OVERALL TOTAL, ELEMENT TOTAL AND EACH NUCLIDE (NI-ZR) (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO
JPNRSR-78 + JAERI-M-6937 +. 210 PPS, TABS, FIGS, JAN. 1977

IN THIS WORK, THE RADIOACTIVITIES OF FISSION PRODUCTS WERE CALCULATED AND SUMMARIZED IN CONTOUR MAPS AND TABLES DEPENDING ON IRRADIATION AND COOLING TIMES. VALUES OF THE NEUTRON FLUX AND THE ENRICHMENT TREATED HERE ARE REPRESENTATIVE FOR COMMON LWRS. THE MAPS AND TABLES OF 560 NUCLIDES ARE DIVIDED AND COMPLIED.

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

RADIOISOTOPE + FUEL BURNUP + IRRADIATION TESTING + THEORETICAL INVESTIGATION + REACTOR, LWR + NICKEL + ZIRCONIUM + JAPAN + R AND D PROGRAM + JACOBS

124996
MAP OF CALCULATED RADIOACTIVITY OF FISSION PRODUCT. (VOL. II) - MAPS OF RADIOACTIVITY OF EACH NUCLIDE (NB-SB) (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO
JPNRSR-K79 + JAERI-M-6938 +. 209 PPS, JAN. 1977

IN THIS WORK, THE RADIOACTIVITIES OF FISSION PRODUCTS WERE CALCULATED AND SUMMARIZED IN CONTOUR MAPS AND TABLES DEPENDING ON IRRADIATION AND COOLING TIMES. VALUES OF THE NEUTRON FLUX AND THE ENRICHMENT TREATED HERE ARE REPRESENTATIVE FOR COMMON LWRS. THE MAPS AND TABLE OF 560 NUCLIDES ARE DIVIDED AND COMPLIED.

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

RADIOISOTOPE + FUEL BURNUP + IRRADIATION TESTING + THEORETICAL INVESTIGATION + REACTOR, LWR + NIOBIUM + ANTIMONY + JAPAN + R AND D PROGRAM

124997
TAKEDA T
MAP OF CALCULATED RADIOACTIVITY OF FISSION PRODUCT (VOL. III). - MAPS OF RADIOACTIVITY OF EACH NUCLIDE TE-TM (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO
JPRNRSR-80 + JAERI-M-6939 +. 208 PPS, TABS, FIGS, JAN. 1977

IN THIS WORK, THE RADIOACTIVITIES OF FISSION PRODUCTS WERE CALCULATED AND SUMMARIZED IN CONTOUR MAPS AND TABLES DEPENDING ON IRRADIATION AND COOLING TIMES. VALUES OF THE NEUTRON FLUX AND THE ENRICHMENT TREATED HERE ARE REPRESENTATIVE FOR COMMON LWRS. THE MAPS AND TABLES OF 560 NUCLIDES ARE DIVIDED AND COMPLIED.

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

RADIOISOTOPE + FUEL BURNUP + IRRADIATION TESTING + THEORETICAL INVESTIGATION + REACTOR, LWR + URANIUM + TELLURIUM + THULIUM + JAPAN + R AND D PROGRAM + JACOBS

122857
KOBAYASHI K + NAMATAME K + AKIMOTO M

122857 *CONTINUED*
DIGITAL COMPUTER SUBROUTINE STEAM FOR JSME STEAM TABLE (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO
JPNRSR-81 + JAERI-M-6967 +. 12 PPS. 4 TABS. 1977

THE STEAM DIGITAL COMPUTER SUBPROGRAM CALCULATES GIBBS FUNCTION, HELMHOLTZ FUNCTION AND THE SATURATION LINE, AND ALSO PROPERTY VALUES DERIVED THEREFROM, WHICH ARE SHOWN IN JSME STEAM TABLE. WITH PRESSURE AND TEMPERATURE AS INDEPENDENT VARIABLES OF THE SUBROUTINE, SPECIFIC VOLUME, SPECIFIC ENTHALPY, SPECIFIC ENTROPY AND SPECIFIC ISOBARIC HEAT CAPACITY ARE OBTAINED AS THE DEPENDENT VARIABLES. THE JSME STEAM TABLE AND THE COMPUTED RESULTS OF SPECIFIC VOLUME, SPECIFIC ENTHALPY AND SPECIFIC ENTROPY ARE IN COMPLETE AGREEMENT, EXCEPT NEAR THE CRITICAL POINT. THE AGREEMENT IN SPECIFIC ISOBARIC HEAT CAPACITY IS VERY GOOD BELOW PRESSURE 500KG/CM2.

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

COMPUTER PROGRAM + STEAM + PROPERTY, PHYSICAL + THERMAL PROPERTY + JACOBS

126858
TASAKA K
DCHAIN: A CODE FOR ANALYSIS OF BUILD-UP AND DECAY OF NUCLIDES (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO
JPNRSR-82 + JAERI-1250 +. 25 PPS. 2 TABS. 3 FIGS. 20 REFS. MARCH 1977

DCHAIN IS A ONE POINT DEPLETION CODE WHICH SOLVES THE COUPLED EQUATIONS OF RADIOACTIVE GROWTH AND DECAY FOR A LARGE NUMBER OF NUCLIDES BY THE BATEMAN METHOD. THE CODE CAN TREAT ANY TYPE OF TRANSMUTATION THROUGH DECAYS OR NEUTRON INDUCED REACTIONS. IT CAN TREAT CYCLIC CHAINS BY AN APPROXIMATION. POWER, NEUTRON FLUX, NEUTRON SPECTRUM, AND FISSION RATIO AND FISSION ENERGY OF EACH FISSILE NUCLIDE CAN BE VARIED FOR EACH TIME STEP. THE CODE USES VARIABLE DIMENSION ARRAYS AND THERE IS LITTLE LIMITATION IN NUMBER OF NUCLIDES OR LENGTH OF A CHAIN.

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

COMPUTER PROGRAM + ELEMENTS AND ISOTOPES + FUEL BURNUP + PRODUCTION + ASSAY + PLUTONIUM + URANIUM + FISSION PRODUCT ACTIVITY, GROSS + FOREIGN EXCHANGE

127037
TASAKA K + SODA K + SHIBA M
ANALYSIS OF LOFT L1-2 EXPERIMENT BY CODE RELAP-4J (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO
JPNRSRP-85 + JAERI-M-7037 +. 141 PPS. 11 TABS. 104 FIGS. APRIL 1977

THE L1-2 IS A SIMPLE ISOTHERMAL BLOWDOWN TEST WITH A CORE SIMULATOR AND NO ECC ACTIVATION. IT PROVIDES THE BASIS FOR FUTURE LOSS OF COOLANT EXPERIMENTS WITH A NUCLEAR CORE AND ECC ACTIVATION. RESULTS ARE: (1) THE CALCULATED SYSTEM PRESSURE TRANSIENT AGREES WELL WITH EXPERIMENT. (2) THE PRESSURIZER PRESSURE DECREASES RAPIDLY, COMPARED WITH EXPERIMENT. (3) THE CALCULATED DENSITY TRANSIENT IN THE COLD LEG AGREES WELL WITH EXPERIMENT. (4) EFFECT OF THE PUMP CHARACTERISTICS ON ANALYTICAL RESULTS IS INSIGNIFICANT IN THE ISOTHERMAL TEST.

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

ACCIDENT, LOSS OF FLOW + COMPUTER PROGRAM + PRESSURE TRANSIENT + PRESSURIZER + BLOWDOWN + COMPARISON, THEORY AND EXPERIENCE + FOREIGN EXCHANGE

131132
REPORT ON SERIES 3 REFLOOD EXPERIMENT (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-6981 + JPNRSR-88 +. 71 PPS. TABS. FIGS. REFS. FEB. 1, 1977

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

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

JAPAN + R AND D PROGRAM + TESTING + CORE REFLOODING + SAFETY INJECTION + THERMOCOUPLE + REACTOR, PWR + ENGINEERED SAFETY FEATURE + HYDRAULIC EFFECT + FOREIGN EXCHANGE

131133
IGUCHI T + MURAO Y + SUDOH T
DATA REPORT ON SERIES 3 REFLOOD EXPERIMENT (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO
JAERI-M-6983 + JPNRSR-89 +. 298 PPS. TABS. FIGS. FEB. 1, 1977

131133 *CONTINUED*
THE DATA OBTAINED IN THE SERIES 3 REFLOOD EXPERIMENT FROM DECEMBER 1975 TO JANUARY 1976 ARE PRESENTED. IN THIS SERIES OF EXPERIMENTS, THE FLOW RESISTANCE IN A PRIMARY LOOP AND THE TEMPERATURE OF THE FLOW HOUSING WERE INVESTIGATED. PERFORMANCE AND DURABILITY OF THE IMPROVED THERMOCOUPLE ATTACHMENT WERE ALSO INVESTIGATED AND ARE REPORTED.

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

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

131134
MURAO Y + IGUCHI T + SUDOH T + SUDO Y
REPORT ON SERIES 4 REFLOOD EXPERIMENT (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-6982 + JPNRSR-90 +. 44 PPS, 8 TABS, 31 FIGS, 10 REFS, FEB. 1, 1977

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

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

JAPAN + R AND D PROGRAM + SIMULATION + FUEL ROD + HEATERS + CORE REFLOODING + SAFETY INJECTION + THERMAL HYDRAULIC ANALYSIS + FOREIGN EXCHANGE

131136
SUGIMOTO J
STUDY ON THERMOCOUPLE ATTACHMENT IN REFLOOD EXPERIMENTS (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-6985 + JPNRSR-92 +. 36 PPS, FIGS, REFS, FEB. 1, 1977

DISCUSSES THE METHOD OF THERMOCOUPLE ATTACHMENT TO HEATER RODS FOR SURFACE TEMPERATURE MEASUREMENT IN REFLOOD EXPERIMENTS. THE METHOD USED THUS FAR IN JAERI'S REFLOOD EXPERIMENTS DO NOT ESTIMATE THE QUENCH TIMES EXACTLY. VARIOUS ATTACHMENT METHODS HAVE BEEN TESTED AND THOSE PROVED TO BE EFFECTIVE ARE DESCRIBED.

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

JAPAN + *THERMOCOUPLE + CORE REFLOODING + SAFETY INJECTION + *MEASUREMENT + INSTRUMENT, TEMPERATURE + HEATERS + FUEL ROD + FASTENER + FOREIGN EXCHANGE

131137
SOBAJIMA M + ADACHI H + MATSUMOTO I + MURATA H
ROSA-II TEST DATA REPORT NO. 7 - WORST BREAK CONDITION AND PUMP EFFECT - (RUNS 318, 320, 321, 322, 323) (IN JAPANESE)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7106 + JPNRSR-93 +. 181 PPS, TABS, FIGS, MAY 9, 1977

A SIMULATED EXPERIMENT OF LOCA INCLUDING ECCS OPERATION WAS PERFORMED WITH THE USE OF THE ROSA-II TEST FACILITY WHICH IS DESIGNED TO SIMULATE THE TROJAN REACTOR IN TERMS OF THERMAL HYDRAULIC RESPONSE. PART OF THE EXPERIMENT RESULTS ARE GIVEN WITH AN INTERPRETATION OF THE DATA. THE OBJECTIVE OF EACH RUN WHICH IS DESCRIBED WAS TO EXAMINE THE WORST CONDITION OF A BREAK PREDICTED BY PRELIMINARY ANALYSIS WITH RELAP-3. TWO KINDS OF BREAK CONDITIONS WERE TAKEN WITH OR WITHOUT PUMP REVOLUTION AFTER A BREAK. ECCS WAS PROVIDED IN HOT LEGS AND COLD LEGS FOR HPCI, IN HOT LEGS ONLY FOR LPCI, AND IN COLD LEGS ONLY FOR THE ACCUMULATORS.

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

JAPAN + R AND D PROGRAM + TESTING + EMERGENCY COOLING SYSTEM + *BLOWDOWN + *ACCIDENT, LOSS OF COOLANT + SIMULATION + REACTOR, PWR + FOREIGN EXCHANGE

131552
SUDOH T + MURAO Y + IGUCHI T + SUDO Y
DATA REPORT ON SERIES 4 REFLOOD EXPERIMENT (IN JAPANESE & ENGLISH)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7169 + JPNRSR-94 +. 193 PPS, 5 TABS, 4 FIGS, 10 REFS, JUNE 28, 1977

MEASURED RESULTS AND CALCULATED HEAT TRANSFER BEHAVIOR OF THE SERIES 4 REFLOOD EXPERIMENT ARE PRESENTED. EXPERIMENTS WERE CARRIED OUT IN JUNE AND JULY 1976. THE OBJECTIVE WAS TO INVESTIGATE THE REFLOOD PROCESS UNDER DIFFERENT SYSTEM PRESSURE USING HEATER RODS OF SIMULATED HEAT CAPACITY.

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

131552 *CONTINUED*
JAPAN + R AND D PROGRAM + CORE REFLOODING + SAFETY INJECTION + SIMULATION + FUEL ROD + HEATERS + HEAT TRANSFER ANALYSIS + FOREIGN EXCHANGE

131553
NAMATAME K + KOBAYASHI K + SODA K
SUBCOOLED DECOMPRESSION ANALYSIS OF LOFT L1-3A TEST DATA WITH DIGITAL COMPUTER CODE DEPCO-MULTI
JAPAN ATOMIC ENERGY RESEARCH INST., TOKYO
JAERI-M-7173 + JPNRSR-95 +. 18 PPS, 13 FIGS, 5 REFS, JULY 1977

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

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

COMPUTER PROGRAM + BLOWDOWN + ACCIDENT, LOSS OF COOLANT + SUBCOOLING + REACTOR, PWR + JAPAN + FOREIGN EXCHANGE

131554
INABE T + OHNISHI N
QUASI STEADY STATE MULTI-DIMENSIONAL SPACE-DEPENDENT KINETIC CODE, EUREKA-SPACE (IN JAPANESE & ENGLISH)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7183 + JPNRSR-96 +. 77 PPS, 3 TABS, 10 FIGS, JUNE 30, 1977

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

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

*COMPUTER PROGRAM + THERMAL HYDRAULIC ANALYSIS + ACCIDENT, REACTIVITY + REACTOR KINETICS + REACTOR, LWR + JAPAN + FOREIGN EXCHANGE

131555
TAKEDA T
USER'S GUIDE FOR CODAC-NO 4: A COMPUTER PROGRAM TO CALCULATE NUCLIDE YIELDS IN COMPLEX DECAY AND ACTIVATION CHAIN (IN JAPANESE & ENGLISH)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7230 + JPNRSR-97 +. 96 PPS, 3 TABS, 29 FIGS, 5 REFS, JULY 29, 1977

THIS REPORT IS PREPARED AS A USER'S INPUT MANUAL AND PROVIDES A GENERAL DESCRIPTION OF THE CODE AND INSTRUCTIONS FOR ITS USE. THE CODE IS DESCRIBED IN DETAIL AND BY THE USE OF VARIOUS EXAMPLES. THE CODE REPRESENTS A VERSION OF THE CODAC-NO. 1 CODE. THE CODE DEVELOPED IS CAPABLE OF CALCULATING RADIOACTIVE NUCLIDE YIELDS IN AN ANY GIVEN COMPLEX DECAY AND ACTIVATION CHAIN IN ACCORDANCE WITH A GIVEN IRRADIATION HISTORY. LISTS OF FORTRAN STATEMENTS, LISTS OF INPUT PARAMETERS AND DEFINITIONS AND EXAMPLES OF INPUT DATA AND OUTPUT ARE GIVEN IN THIS REPORT. IN ADDITION TO PREVIOUS EXAMPLES, DETAILS OF INPUT DATA PREPARED FOR CONCRETE COMPLEX CHAIN INVOLVED IN FISSION AND ACTIVATION ARE ALSO GIVEN.

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

JAPAN + GUIDE + COMPUTER PROGRAM + FISSION PRODUCT RELEASE + FISSION PRODUCT TRANSPORT + ACTIVATION + ACTIVATION PRODUCT + FOREIGN EXCHANGE

131556
BEHAVIOR OF ZIRCALOY CLADDING UNDER A POSTULATED LOSS-OF-COOLANT ACCIDENT CONDITION (IN JAPANESE & ENGLISH)
JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
JAERI-M-7247 + JPNRSR-98 +. 101 PPS, TABS, FIGS, REFS, AUG. 18, 1977

PRESENTS A REVIEW OF STUDIES ON THE BEHAVIOR OF ZIRCALOY CLADDING UNDER A POSTULATED LOSS-OF-COOLANT ACCIDENT CONDITION. EXPERIMENTAL DATA OF ZIRCALOY-STEAM REACTION RATE, EMBRITTLEMENT OF A STEAM-OXIDIZED ZIRCALOY AND CLADDING DEFORMATION ARE SUMMARIZED. SENSITIVITY STUDIES OF VARIOUS PARAMETERS ON LOCA ANALYSIS BY A COMPUTER CODE ARE PRESENTED.

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

JAPAN + CLADDING + BEHAVIOR + ZIRCALOY + ACCIDENT, LOSS OF COOLANT + EMBRITTLEMENT + FUEL SWELLING + FOREIGN EXCHANGE

117045
DESIGN OF NUCLEAR POWER PLANTS; RESEARCH ON COUNTERMEASURES AGAINST EARTHQUAKES
JAPAN ELECTRIC ASSOCIATION + SCIENCE & TECHNOLOGY AGENCY, JAPANESE GOVERNMENT
95 PPS, FIGS, REFS, FEB. 1976

SIX SITES WERE SELECTED AS POSSIBLE LOCATIONS FOR NUCLEAR POWER REACTORS, AND STUDIES OF EACH SITE WERE CONDUCTED. EARTHQUAKE GENERATING MECHANISMS BASED ON THE MOST RECENT INFORMATION, ASSUMED FOUNDATION MODELS, AND ANALYSIS OF THE VERTICAL COMPONENTS OF SEISMIC MOTIONS WERE INCORPORATED INTO THE STUDY. ONLY VERY LIMITED RESULTS WERE OBTAINED THAT CAN BE USED DIRECTLY IN EARTHQUAKE PROOF DESIGN. DATA ON PAST EARTHQUAKES AT EACH OF THE 6 SITES WERE REFERENCED TO MAKE COMPUTATIONS OF THE MAGNITUDE OF EFFECTS AS FUNCTIONS OF THE DISTANCE FROM THE EPICENTER FROM WHICH THE MAXIMUM VELOCITY AMPLITUDES, PERIODIC PROPERTIES, ENVELOPING/LITERAL/FUNCTIONS, AND ELAPSED TIMES WERE ESTIMATED. A METHOD TO SYNTHESIZE SIMULATED EARTHQUAKE WAVES WAS DEVELOPED.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TN 37830

*SITING + *EARTHQUAKE + SIMULATION + EARTHQUAKE RECORDS + *POWER PLANT, NUCLEAR

123389
EXPERIMENTAL STUDY ON THE FIRE RESISTANCE OF USED NUCLEAR FUEL SHIPPING CONTAINERS
SCIENCE & TECHNOLOGY AGENCY, TOKYO, JAPAN
JPNRSR-33 +. 93 PPS, TABS, FIGS, JAN. 16, 1976

DESCRIBES THE TESTS CONDUCTED ON 1/2.5 AND 1/2 CASK MODELS TO GATHER BASIC DATA FOR EVALUATING THE FIRE RESISTANCE OF THE CASKS INCLUDING THE EFFECTS OF A LEAD EXPANSION CHAMBER AND TO EVALUATE THE EFFECTS OF VOIDS AFTER THE LEAD HAS MELTED, FLOWED OUT, AND RESOLIDIFIED. THE TEST CONDITIONS INCLUDE HEATING TO 800 C FOR 30 MINUTES IN AN OPEN FIRE. TEST RESULTS ARE SUMMARIZED.

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

*TRANSPORTATION AND HANDLING + SHIPPING CONTAINER + REGULATION, IAEA + SAFETY ANALYSIS + JACOBS + TEST, DROP

122571
STUDY ON STRUCTURAL DESIGN CODE OF FATIGUE AND CREEP INTERACTION OF NUCLEAR MATERIALS (IN JAPANESE)
JAPAN WELDING ENGINEERING SOCIETY
JPNRSR-69 + JWES-AE-7605 + REPORT 51-STAE-113 +. 306 PPS, TABS, FIGS, OCT. 1976

TESTS OF FATIGUE AND CREEP INTERACTION WERE CARRIED OUT USING TYPE 304 AND 316 STAINLESS STEELS AND 2 1/4 CR-1MO STEEL. THERMAL AND MECHANICAL RATCHETING TESTS WERE PERFORMED IN WHICH THREE BAR MODELS AND CYLINDRICAL PIPES WERE USED AND THE DEFORMATION WAS MEASURED. EXPERIMENTAL RESULTS OF MECHANICAL RATCHETING WERE COMPARED WITH THEORETICAL VALUES. FATIGUE LIFE OF THESE MATERIALS IS INFLUENCED BY STRAIN RATE, HOLDING TIME OF STRAIN AND STRESS BETWEEN 450 AND 650 DEG C. EXPERIMENTAL RESULTS OF MECHANICAL RATCHETING OF CYLINDRICAL PIPES, SHOWS THAT FATIGUE LIFE AND DEFORMATION ARE INFLUENCED BY THICKNESS, TEMPERATURE AND MEAN STRESS.

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

STEEL, STAINLESS + STEEL, FERRITIC + FATIGUE + CREEP + DEFORMATION + TESTING + JACOBS

122572
STUDY ON LOCAL STRUCTURAL BEHAVIOR AND SAFETY ANALYSIS OF NUCLEAR PIPING SYSTEM (IN JAPANESE)
JAPAN WELDING ENGINEERING SOCIETY
JPNRSR-70 + JWES-AE-7606 + REPORT 51-STAE-114 +. 181 PPS, TABS, FIGS, OCT. 1976

OBJECTIVE IS TO INVESTIGATE INTERGRANULAR STRESS CORROSION CRACKING (IGSCC) OF SENSITIZED STAINLESS STEEL IN OXYGENATED HIGH TEMPERATURE WATER. A COMPUTER CODE WAS DEVELOPED TO CALCULATE THE LOCALIZED ELASTIC-PLASTIC STRESS USING A FINITE ELEMENT METHOD. TWO TESTS WERE CARRIED OUT ON ELBOWS AND TAPERED PIPES WHICH WERE INTERNALLY PRESSURIZED AND BENT BY EXTERNAL LOAD. THE RESULTS SHOWED THAT GEOMETRICAL DISCONTINUITY EASILY DROVE THE STRAIN TO THE PLASTIC REGION WHILE GROSS STRUCTURAL BEHAVIOR REMAINED IN ELASTIC REGION.

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

PIPES AND PIPE FITTINGS + SAFETY ANALYSIS + STEEL, STAINLESS + STRESS CORROSION + CRACK + WATER + COMPUTER PROGRAM + JACOBS

121013
LEACHING OF RADIOACTIVE NUCLIDES FROM CEMENT-SOLIDIFIED RADIOACTIVE WASTE IN SALINE (PART 2)
CENTRAL RESEARCH INSTITUTE OF ELECTRIC POWER INDUSTRY, JAPAN
JPNRSR-32 + CRIEPI-275003 +. 30 PPS, 20 FIGS, NOV. 1975

THE LEACHING OF 60CO AND 137CS FROM A SOLIDIFIED WASTE INTO THE ENVIRONMENTAL WATER HAS BEEN STUDIED TO EVALUATE THE SAFETY OF SEA DISPOSAL OF THE RADIOACTIVE WASTE CEMENT COMPOSITES. LEACHING TEST WAS CARRIED OUT FOLLOWING THE METHOD RECOMMENDED BY IAEA. THE SAMPLES TESTED WERE PREPARED BY INCORPORATING SIMULATED LIQUID WASTES FROM BWR TO PORTLAND BLAST-FURNANCE SLAG CEMENT

121013 *CONTINUED*
WITH AGGREGATES. THE LEACHING RATIO WAS MEASURED AS FUNCTIONS OF LEACHANT, SETTING TIME AND RADIOACTIVE NUCLIDE. THE RESULTS ARE GIVEN.

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

LEACHING + CONTAINMENT + WASTE DISPOSAL + OCEAN AND SEA + COBALT + CESIUM + CONCRETE

119274
KUWAHARA K + NITTA A
PREDICTION OF THE ENDURANCE LIFE OF A HEAT EXCHANGER TUBING USED AT ELEVATED TEMPERATURES FOR A LONG TIME (IN JAPANESE)
CENTRAL RESEARCH INSTITUTE ELECTRIC POWER INDUSTRY, JAPAN
CRIEPI: 275012 +. 16 PPS, 10 FIGS, JAN. 1976

INTERNAL PRESSURE AND CREEP-RUPTURE STRENGTH OF HIGH-TEMPERATURE HEAT EXCHANGER TUBING DECIDES ITS ENDURANCE LIFE. THE METALLURGICAL STRUCTURE IN THE SURFACE LAYERS CHANGES AND ITS VARIATION CAUSES A DECREASE IN TIME TO CRACK INITIATION AND AN INCREASE IN CRACK PROPAGATION RATE, THAT IS, A DECREASE IN THE CREEP-RUPTURE STRENGTH OF THE TUBING, UNDER THE INTERNAL PRESSURIZED CONDITION.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

STRESS RUPTURE + CREEP + TUBING + GROWTH/DEVELOPMENT + HEAT EXCHANGERS + JACOBS

119271
TOMODA Y + KINOSHITA M
AN EXAMINATION OF MECHANICAL PROPERTIES RELATING TO FAST DUCTILE FRACTURE IN STEELS (IN JAPANESE)
CENTRAL RESEARCH INST. ELECTRIC POWER INDUSTRY, JAPAN
CRIEPI: 275013 +. 88 PPS, FIGS, APRIL 1976

A METHOD OF STRESS ANALYSIS IS INTRODUCED TO DETERMINE THE STRESS IN THE NECK OF A FLAT TENSILE BAR. USING THE METHOD, STRESS DISTRUBTIONS ARE CALCULATED TO EXAMINE MECHANICAL PROPERTIES RELATING TO FAST FRACTURE IN A FEW STEELS INCLUDING A CARBON STEEL, A LOW ALLOY AND AN AUSTENITIC STAINLESS STEEL. IT IS SHOWN THAT THE FAST FRACTURE PROPERTY OF STEELS RELATES TO BOTH THE PARAMETER (PHI) FOR THE NECKED PROFILE LINE AND THE STRESS RATE AFTER UNIFORM ELONGATION OF THE MATERIAL.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

STEEL + TENSILE PROPERTY + STEEL, STAINLESS + PROPERTY, PHYSICAL + STRESS RUPTURE + DUCTILITY + JACOBS

120072
ONCHI T
A SURVEY OF OPERATING EXPERIENCE AND PERFORMANCE FOR LIGHT WATER REACTOR FUELS (IN JAPANESE)
CENTRAL RESEARCH INST. ELECTRIC POWER INDUSTRY, JAPAN
CRIEPI: 275004 +. 113 PPS, FIGS, APRIL 1976

THE PURPOSE OF THIS INVESTIGATION IS TO REVIEW CHANGES OF FUEL DESIGN AND SPECIFICATIONS AS WELL AS OPERATING EXPERIENCES IN COMMERCIAL LWR FUELS FOR THE PAST 15 YEARS. ALSO, THE PRESENT STATUS OF PELLET(FUEL)-CLAD-INTERACTION (PCI) PHENOMENA WHICH IS ONE OF THE MOST IMPORTANT CAUSES OF FUEL FAILURE IS SUMMARIZED. THE EFFECT OF FUEL DESIGN AND REACTOR OPERATING MODE ON PCI FAILURE BEHAVIOR ARE PRESENTED. THE PRESENT STATUS OF PLASTIC DEFORMATION AND STRESS CORROSION CRACKING INDUCED BY IODINE VAPOURS AS FUEL FAILURE MECHANISMS ARE INVESTIGATED AND DISCUSSED IN DETAIL.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

*FUEL, NUCLEAR + FUEL, PELLET TYPE + *REACTOR, LWR + REACTOR, BWR + REACTOR, PWR + *FUEL CLAD INTERACTION + DESIGN CRITERIA + FUEL ROD + STRESS CORROSION + CRACK + PLASTICITY + SURVEY + IODINE + FAILURE, CLADDING + *OPERATING EXPERIENCE + JACOBS

118916
FUKUMOTO H + OKAMOTO H + FIJIMOTO J
REMOTE ULTRASONIC TESTING SYSTEM FOR NUCLEAR POWER PLANTS STRAIGHT PIPING (PART I) (IN JAPANESE)
CENTRAL RESEARCH INST. OF ELECTRIC POWER INDUSTRY, JAPAN
CRIEPI:275022 +. 24 PPS, FIGS, JULY 1976

A SYSTEM HAS BEEN DEVELOPED TO MINIMIZE PERSONNEL RADIATION EXPOSURE DURING INSERVICE INSPECTION IN NUCLEAR POWER PLANTS USING REMOTELY OPERATING ULTRASONIC TESTING SYSTEM. THE SYSTEM USES A FLEXIBLE PLASTIC BELT ATTACHING ON STRAIGHT PIPING, AND DOES NOT REQUIRE THE USE OF ANY PERMANENTLY ATTACHED GUIDE OR TRACK. THIS REPORT, THE INTRODUCTORY ONE, DEALS WITH THE DESIGN CONCEPTS AND TESTING RESULTS OF REMOTE ULTRASONIC TESTING SYSTEM.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

*ULTRASONICS + *TEST, NONDESTRUCTIVE + PIPES AND PIPE FITTINGS + *REMOTE MANIPULATING AND VIEWING + EQUIPMENT DEVELOPMENT + INSERVICE INSPECTION + JACOBS

118915
OKAMOTO H + FUKUMOTO H + KITAMI T
DEVELOPMENT OF REMOTE OPERATION SYSTEM FOR CONTROL ROD DRIVES REMOVAL AND INSTALLATION IN BOILING WATER REACTORS - DEVELOPMENT OF THE FUNDAMENTAL SYSTEM (IN JAPANESE)
CENTRAL RESEARCH INST. OF ELECTRIC POWER INDUSTRY, JAPAN
CRIEPI:275023 +. 21 PPS. FIGS. JUNE 1976

THE REMOTE OPERATION SYSTEM FOR MAINTENANCE WORKS IN NUCLEAR POWER PLANTS IS VERY EFFECTIVE FOR THE REDUCTION OF THE RADIATION EXPOSURE OF MAINTENANCE WORKERS AND THE SPEED-UP OF MAINTENANCE WORKS. HAVE DEVELOPED THE REMOTE OPERATION SYSTEM FOR THE MAINTENANCE WORK OF THE CONTROL ROD DRIVE (CRD) IN BOILING WATER REACTORS (BWR) FOR THE PURPOSE OF APPLYING TO EXISTING PLANTS. THE WORK IS THOUGHT TO BE ONE OF THE MORE IMPORTANT WORKS OUT FOR MAINTENANCE WORKS. THE SCOPE OF REMOTE OPERATION OF THE SYSTEM IS THE FUNDAMENTAL PROCESS OF CRD REMOVAL AND INSTALLATION PROCESS IN THE BELOW-VESSEL AREA. THIS PAPER DESCRIBES THE DESIGN THOUGHT, THE OUTLINE AND THE PERFORMANCE OF THE SYSTEM.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

*EQUIPMENT DEVELOPMENT + REACTOR, BWR + *CONTROL ROD DRIVES + *MAINTENANCE AND REPAIR + REMOTE MANIPULATING AND VIEWING + INSTALLATION + JACOBS

119128
FAKAKU H + TOKIWAI M
BASIC STUDY ON SOME METALLURGICAL FACTORS AND SURFACE TREATMENT EFFECTS OF STAINLESS STEELS FOR BWR COOLING PIPES (IN JAPANESE)
CENTRAL RESEARCH INST. OF ELECTRIC POWER INDUSTRY, JAPAN
CRIEPI:275038 +. 41 PPS. TABS. FIGS. MAY 1976

RESIDUAL STRESS MEASURED BY X-RAY TECHNIQUE SHOWS LARGE RESIDUAL STRESS IS CAUSED BY MACHINING AND GRINDING. HOWEVER, SHOT PEENING EASILY GIVES RISE TO ABOUT 70 KG/MM(SQUARED) COMPRESSIVE RESIDUAL STRESS, AND IT IS ALSO PROVED THAT ABOUT 50 KG/MM (SQUARED) OF COMPRESSIVE STRESS IS RETAINED EVEN AFTER HEATING UP THE SHOT-PEENED SPECIMEN TO 700 C X 5 MIN.. FROM THE ABOVE RESULTS IT IS SUGGESTED THAT SHOT PEENING WILL PROTECT THE STAINLESS STEEL WELDMENTS AGAINST STRESS CORROSION CRACKING.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX, 62, OAK RIDGE, TENN. 37830

STEEL, STAINLESS + WELDING + STRESS CORROSION + CRACK + JACOBS

119993
IWAHORI T + MIZUNO T
DEPOSITION OF SUSPENDED CORROSION PRODUCTS IN BWR'S (PART III) - THE ROLE OF SURFACE CHEMICAL NATURE IN DEPOSITION OF HEMATITE PARTICLES (IN JAPANESE)
CENTRAL RESEARCH INST. OF ELECTRIC POWER INDUSTRY, JAPAN
CRIEPI: 275027 +. 29 PPS, 25 FIGS, JUNE 1976

CRUD (SUSPENDED IRON OXIDES) DEPOSITS HAVE THE POTENTIAL FOR FORMATION OF INDUCED RADIOACTIVITY AND OVERHEATING OF THE FUEL CLADDING. IN THIS PAPER, THE DEPOSITION OF HEMATITE PARTICLES ON A BOILING HEAT TRANSFER SURFACE AND ON A ROTATING CYLINDER SURFACE AT ATMOSPHERIC PRESSURE WERE STUDIED. THE EFFECTS OF SURFACE CHEMISTRY ON DEPOSITION WERE ALSO INVESTIGATED. RESULTS INDICATED THAT: (1) CRUD PARTICLES IN THE PRIMARY COOLANT OF A BWR HAVE A POSITIVE ELECTRIC CHARGE AND A LYOPHOBIC COLLOIDAL BEHAVIOR, WHICH ARE MOSTLY GREATER THAN 0.45 MICRON IN SIZE. (2) THE ELECTROSTATIC RELATIONSHIP BETWEEN THE CHARGE OF A HEMATITE.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

JAPAN + R AND D PROGRAM + REACTOR, BWR + *CRUD + *CORROSION + IRON + OXIDE + *DEPOSITION + COLLOID + HEAT TRANSFER ANALYSIS + CLADDING + COOLANT CHEMISTRY + HEAT TRANSFER, BOILING + JACOBS

118529
HATTA H + MIMURA M + HORIE Y
SOME PROBLEMS ON A LOCA ANALYSIS OF LARGE PWR BY RELAP-4 CODE (IN JAPANESE)
CENTRAL RESEARCH INST. OF ELECTRIC POWER INDUSTRY, JAPAN
CRIEPI-275028 +. 62 PPS. FIGS. JULY 1976

A PARAMETER SURVEY OF A LOCA IN A LARGE BWR WAS DONE BY THE RELAP-4 COMPUTER CODE AND THE BASIC FEATURES AND APPLICABILITY OF THE CODE WERE INVESTIGATED. RELAP-4 MAY BE USED FOR THE ANALYSIS OF BLOWDOWN PHENOMENA IN A BWR LOCA BUT ITS TWO-PHASE FLOW CALCULATION MODEL SHOULD BE IMPROVED BEFORE IT CAN BE APPLIED TO THE REFLOODING CALCULATION.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

REACTOR, BWR + COMPUTER PROGRAM + ACCIDENT, LOSS OF FLOW + ACCIDENT, LOSS OF PRESSURE + REACTOR, SAFETY RESEARCH + JAPAN + BLOWDOWN + SUBCHANNEL ANALYSIS + JACOBS

120041
TASHIRO H
SLUDGE REMOVAL FROM A BWR (IN JAPANESE)
CENTRAL RESEARCH INST. OF ELECTRIC POWER INDUSTRY. JAPAN
CRIEPI:275045 +. 27 PPS. 29 FIGS. JULY 1976

THIS REPORT INVESTIGATES THE REMOVAL OF ACTIVATED SLUDGE WHICH ACCUMULATES AT VARIOUS STAGNANT POINTS WITHIN THE REACTOR COOLING SYSTEM. A METHOD FOR REMOVING THE SLUDGE USING WATER JETS OR THE EQUIVALENT IS DISCUSSED. ALSO, THE SELECTION OF THE APPROPRIATE REMOVAL POINT FOR SENDING THE WATER TO THE CLEANUP SYSTEM IS DISCUSSED.

AVAILABILITY - ERDA TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830

JAPAN + *REACTOR, BWR + *CRUD + WATER TREATMENT + CORROSION + DEPOSITION + COOLANT CHEMISTRY + JACOBS

124994
STUDY ON SEPARATION PERFORMANCE FOR RADIOACTIVE GASEOUS WASTE BY PERMSELECTIVE MEMBRANES (IN JAPANESE)
TOKYO-SHIBAURA ELECTRIC CO. LTD., JAPAN
JPNRSR-68 + REPORT 51 STAE-58 +. 60 PPS. TABS. FIGS. SEPT. 1976

THIS PAPER PRESENTS, FOR THE STEADY AND TRANSITION SEPARATION PERFORMANCE TEST ON THE 9 STAGE RECYCLE CASCADE GAS SEPARATION EQUIPMENT WITH TWO KINDS OF MEMBRANE, THE EVALUATION OF THE ANALYTICAL VALUE AND EXPERIMENTAL VALUE AND THE STUDYING OF THE APPLICABILITY TO NUCLEAR PLANTS OF THIS SYSTEM AND OF THE FEASIBILITY OF THIS MEMBRANE METHOD. IT IS FOUND THAT THE RECYCLE RATE AT EACH STAGE AND THE REFLUX RATIO AT THE FINAL ENRICHMENT STAGE AFFECT THE RECYCLE CASCADE SEPARATION PERFORMANCE. THE SEPARATION EFFICIENCY PER STAGE IN THE RECYCLE CASCADE BECAME APPROXIMATELY 97% ON AN AVERAGE.

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

PERMSELECTIVE MEMBRANE + FILTER, MEMBRANE + WASTE TREATMENT + DIFFUSION + JAPAN + R AND D PROGRAM + JACOBS

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.)

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