# RSIC Newsletter

RADIATION SHIELDING INFORMATION CENTER

OAK RIDGE NATIONAL LABORATORY

OPERATED BY UNION CARBIDE CORPORATION • FOR THE U.S. ATOMIC ENERGY COMMISSION

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No. 112 March 1974

*Education is to man what manure is to the pea.*

*...Reginald C. Punnett (1907)*

## *FOURTH UTILIZATION OF CALIFORNIUM-252 MEETING IN APRIL 1974*

The fourth in a series of AEC-sponsored meetings on the applications of $^{252}Cf$ will be held on April 22-24, 1974, at the Town and Country Hotel, San Diego, California. Previous meetings were held in Augusta, Georgia, in November 1971; Chicago, Illinois, in May 1972; and New York City in March 1973.

The San Diego meeting will review progress made in developing applications of $^{252}Cf$ and will include discussions on basic neutron technology and on the use of the $^{252}Cf$ Demonstration Centers. The principal attraction of the three-day meeting will be six workshop sessions. In each session, a leader and a panel of experts will discuss informally one area of $^{252}Cf$ application. Tentative subjects are:

- Neutron Technology and Safety
- Activation Analysis
- Aerospace Applications
- Nuclear Industry Applications
- Neutron Gauging and Process Control
- Commercialization of Systems Using $^{252}Cf$

The workshops will provide an opportunity for direct and open discussions of the advantages and problems involved in using $^{252}Cf$.

For more details, prospective attendees should write or telephone W. R. Cornman, Californium-252 Information Center, Savannah River Laboratory, Aiken, South Carolina 29801; telephone (803) 824-6331, extension 2252.

## *USAEC REGULATORY GUIDES*

Regulatory Guides are issued to describe and make available to the public methods acceptable to the AEC Regulatory staff of implementing specific parts of the Commission's regulations, to delineate techniques used by the staff in

evaluating specific problems or postulated accidents, or to provide guidance to applicants. Regulatory Guides are not substitutes for regulations and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission.

Published guides will be revised periodically, as appropriate, to accommodate comments and to reflect new information or experience.

Copies of published guides may be obtained by request indicating the divisions desired to the U.S. Atomic Energy Commission, Washington, D.C. 20545, Attention: Director of Regulatory Standards. Comments and suggestions for improvements in these guides are encouraged and should be sent to the Secretary of the Commission, U.S. Atomic Energy Commission, Washington, D.C. 20545, Attention: Chief, Public Proceedings Staff.

The guides are issued in the following ten broad divisions:

1. Power Reactors
2. Research and Test Reactors
3. Fuels and Materials Facilities
4. Environmental and Siting
5. Materials and Plant Protection
6. Products
7. Transportation
8. Occupational Health
9. Antitrust Review
10. General

Guides of special interest include:

Regulatory Guide 2.1 Shield Test Program for Evaluation of Installed Biological Shielding in Research and Training Reactors

This Guide references ANSI Standard N18.9-1972.

Regulatory Guide 1.69 Concrete Radiation Shields for Nuclear Power Plants.

This Guide references ANSI Standard N101.6-1972 "Concrete Radiation Shields."

## *FIRST TOPICAL MEETING ON THE TECHNOLOGY OF CONTROLLED NUCLEAR FUSION*
*April 16-18, 1974, Sheraton Inn, San Diego, Ca 92101*

The first topical meeting on the technology and engineering of controlled nuclear fusion is intended to provide a cross section of current status in this rapidly growing field. The program covers areas primarily related to the development of controlled thermonuclear fusion as a power generation source, and is international in scope. The speakers are persons actively engaged in this relatively new field, and we hope that much personal communication can be enjoyed. All persons interested in this

important subject are cordially invited to attend and participate in this meeting. San Diego promises to be comfortably cool in the spring and a wide variety of activities are available for your enjoyment.

The program includes the following sessions: REACTOR DESIGN; MATERIALS; SYSTEM STUDIES; NEUTRONICS; BLANKETS AND FIRST WALLS; RADIATION TEST FACILITIES; LASER FUSION; INJECTORS, DIVERTORS AND DIRECT CONVERSION; TRITIUM ENERGY BALANCE & FUEL CYCLES; PLASMAS; FUSION-FISSION; ENVIRONMENTAL ASPECTS; PLASMAS I.

Proceedings - Paper summaries will be distributed at the meeting. Full proceedings of the meeting will be published as CONF-740402 by the U. S. National Technical Information Service. These are included with registration and will be shipped about July 1, 1974.

Meeting Officials - Meeting officials can be contacted through G. R. Hopkins, General Chairman, General Atomic Company, P.O. Box 81608, San Diego, California 92138.

Meeting Sponsors - San Diego Section, Technical Group for Controlled Nuclear Fusion and Power Division - ANS; and U.S. Atomic Energy Commission.

Meeting Organizers - G. R. Hopkins, General Chairman; R. W. Werner, Technical Program Chairman; J. R. Gilleland, Local Arrangements Chairman; C. C. Baker, Finance Chairman.

## *CHANGES TO THE CODE COLLECTION*

The following changes were made to the code collection during January. In several instances, changes to existing code packages are described; others are new additions. Unless otherwise noted, requests should be accompanied by a full reel of magnetic tape.

CCC-82C/ANISN — One-Dimensional, Multigroup Discrete Ordinates Code. W.W. Engle, Jr., Oak Ridge National Laboratory, provided a revision to subroutine CLEAR to improve error procedures through ERRSET in the IBM 360 version.

CCC-187/SAM-CE — Monte Carlo Time-Dependent Three-Dimensional Complex (Combinatorial) Geometry Code System. The SAM-X ENDF format cross-section processor has been added. Contributed by Mathematical Applications Group, Inc., Elmsford, New York. CCC-187A, IBM 360; CCC-187B,CDC 6600.

CCC-229/KRONIC — Calculation of Annual Average External (Beta-Ray and Gamma-Ray) Doses from Chronic Atmospheric Releases of Radionuclides, contributed by Occupational and Environmental Safety Department, Battelle Pacific Northwest Laboratories, Richland, Washington. FORTRAN IV; UNIVAC 1108.

CCC-230/TRIPLET Two-Dimensional, Multigroup, Triangular Mesh, Planar Geometry, Explicit Discrete Ordinates Code. A finite element method, in which the angular flux is assumed to be given by a low-order polynomial in each triangle, is used to solve the discrete ordinates equations. Contributed by Los Alamos Scientific Laboratory. Reference: LA-5428-MS. FORTRAN IV. CCC-230A,CDC 7600; CCC-230B, CDC 6600; CCC-230C, IBM 360.

CCC-231/FRCRL2 Calculation of Fission-Product Release in Reactor Accident Analyses, contributed by Battelle Columbus Laboratories, Columbus, Ohio. Reference: BMI-1913. FORTRAN IV; CDC 6400.

PSR-63/AMPX Multigroup Cross Section Processor.
(a) New input cross section data (ENDF format) have been added to the package for the sample problem (C-12, O-16, Al-27, H-1, Ca, U-235, U-238).
(b) A correction has been recommended by John White, Computer Sciences Division, Union Carbide Corporation Nuclear Division. This update increases the storage available for resonance parameters read from ENDF data tapes. The change consists of replacing 2 statements in subroutine NITAWL. Replace statement ISN 0099 by

```
IGMP=MAXO(NCX,50*IGP,50*IPP,3000)
```

and replace statement ISN 0100 by

```
LPNC=LDSN+IGMP .
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FORTRAN IV; IBM 360.

PSR-69/POWER Source Input Data Generator for ANISN. Jim West of Babcock and Wilcox, the contributor, points out that the primary function is to generate a source distribution array for ANISN and to vary interval width for subsequent runs. (See Feb. 1974 Newsletter for original announcement.) FORTRAN IV; CDC 7600.

PSR-74/TECALC Interactive Calculation of Compton, Coherent, and Photoelectric Mass Attenuation Coefficients for Photons ($E<1$ MeV), and the Mass Absorption Coefficient for Known Materials. Contributed by Health Physics Division, Oak Ridge National Laboratory. Reference: ORNL-TM-4451. BASIC; PDP-10.

## CHANGES TO THE DNA WORKING CROSS SECTION LIBRARY

A new evaluation for carbon has been added to the DNA Working Cross Section Library by F. G. Perey, C. Perey, and C. Y. Fu of Oak Ridge National Laboratory. It is designated as DNA MAT 4274 MOD 0 Carbon.

Three partial evaluations, containing gamma-ray production data only, have also been added. P. Young of Los Alamos Scientific Laboratory has supplied DNA MAT 4547 MOD 0 Ag and DNA MAT 4550 MOD 0 Sn. C. Y. Fu and F. G. Perey of ORNL have supplied DNA MAT 4530 MOD 0 Zn.

A modification to the iron evaluation was recently made. The changes are summarized below.

IRON MAT 4180 ORNL
MOD 2 February 1974

This modification supersedes the previous evaluation, DNA 4180 MOD 1, evaluated by S. K. Penny, W. E. Kinney, R. Q. Wright, F. G. Perey and C. Y. Fu, of ORNL.

It was primarily intended for a revision of the secondary neutron distributions as new data have become available and as the pulsed sphere measurements and related calculations indicated that the previous neutron distributions were inadequate, and for a reevaluation of the gamma-ray production cross sections since the recently available data disagree substantially with the data set upon which the previous evaluation was largely based. These purposes were consistently achieved by an extensive calculation for Fe-56 which also led to the refinement of all reaction cross sections except capture. Significant changes are summerized below.

1. Total cross section between 60 keV and 2 MeV -- peaks and valleys were refined based on 3 sets of ORELA measurements using 1-, 4-, and 12-inch samples.

2. Nonelastic -- obtained by compromising 3 pieces of information, namely, the measured nonelastic cross section, the difference between the evaluated total cross section and the available elastic scattering cross section,and the theoretical interpretation of the total gamma-ray-production cross section. The calculated nonelastic cross section was normalized to this result for constraining the subsequent calculations.

3. Inelastic scattering -- 6 more discrete levels were added based on the Hauser-Feshbach and DWBA calculations in order to include the 4.505-MeV collective state as a discrete level. The direct-interaction components were included in 15 of the 26 discrete levels. The direct-interaction

component was included in the continuum using an empirical treatment. MT=51(0.846-MeV level) was changed between 2 and 5 MeV according to an evaluation by Smith which was based on all available data and his recent measurement. MT=52(the 1.408-MeV level in Fe-54) was revised based on recently available data.

4. The Hauser-Feshbach method was used to calculate the ((n,n') and (n,nx)), ((n,p) and (n,px)) and ((n,A) and (n,ax)) cross sections. The ((n,n') and (n,nx)) cross section above 4.531 MeV was split into the (n,n') continuum, (n,2n), (n,np) and (n,na) cross sections using a statistical-empirical model. The empirical aspects include giant-dipole gamma-ray competition, angular-momentum correction for the second outgoing particle and direct or pre-equilibrium emission of the first particle.

5. The (n,p) cross section was well-defined by measurements and that of Fe-56 was well-fitted by calculation up to 13 MeV. The (n,pn) cross section was taken to be the difference between the calculated Fe-56 ((n,p) and (n,px)) cross section and the measured Fe-56 (n,p) cross section. Neutron distribution for the (n,pn) reaction was calculated and was averaged with that of the reaction for MT=28.

6. There is no measured Fe-56 (n,a) cross section. The calculated Fe-56 (n,a) cross section was restrained by the 14-MeV empirical value and the measured fission-spectrum averages. The Fe-54 (n,a) cross section was taken from a calculation by Kirouac and Slavik. The (n,an) cross section and neutron distribution were calculated and were combined with those of the (n,na) reaction for MT=22.

7. The (n,d), (n,t) and (n,He-3) cross sections were based on calculations using the Pearlstein empirical model and parameters.

8. The two major sets of the gamma-ray production cross sections agree within experimental errors for incident neutron energies below 6 MeV and then start to diverge, the GRT set being 1.6 barns larger than the ORNL set in the 11 to 15 MeV region. Other available data were too limited to resolve this discrepancy. The calculated results are intermediate between the two, being closer to the GRT measurement below 12 MeV and closer to the ORNL data above 12 MeV. For a tentative evaluation, the ORNL data were used for the energy distributions (File 15) as these were given in adequate neutron bins and were the only data available from 15 to 20 MeV. The gamma rays below 0.69 MeV were not measured and were filled in using the calculated values. Up to 10 MeV incident neutron energy, the energy conservation law re ating the calculated outgoing particle energy and the evaluated gamma-ray energy was applied to determine the gamma-ray multiplicities. From 10 to 16 MeV the calculated multiplicities were used and were extrapolated to 20 MeV according to the shape given by the ORNL measurement.

## PERSONAL ITEMS

*Alvin M. Weinberg*, who resigned his post as director of Oak Ridge National Laboratory to head the new Oak Ridge Associated Universities-formed Institute for Energy Analysis, effective January 3, was appointed by William E. Simon, administrator of the Federal Energy Office (FEO), as director of the New Energy Research and Development Office, on January 7. Dr. Weinberg is taking a leave of absence from the IEA for a period of at least six months to assume his post in Washington. In the interim, *H. G. MacPherson*, deputy director of IEA, will serve as acting director of the Institute.

*Raphael (Ray) Aronson*, formerly Professor of Nuclear Engineering at New York University, is with Polytechnic Institute of New York, in Brooklyn.

*Theodore Kettler* is now with Automation Industries, Inc., Nuclear Energy Services Division, Danbury, Connecticut.

## VISITORS TO RSIC

Visitors to RSIC during the month of February were: Capt. J. A. Angelo, Jr., Patrick Air Force Base, Florida; W. S. Lyon, Jr., Analytical Chemistry Division, ORNL; J. Silva, KMS Fusion, Ann Arbor, Michigan.

## FEBRUARY ACCESSION OF LITERATURE

The following literature cited has been ordered for review, and that selected as suitable will be placed in the RSIC Information Storage and Retrieval Information System (SARIS). This early announcement is made as a service to the shielding community. Copies of the literature are not distributed by RSIC. They may generally be obtained from the author or from a documentation center such as the National Technical Information Service (NTIS), Department of Commerce, Springfield, Virginia 22151.

RSIC maintains a microfiche file of the literature entered into SARIS, and duplicate copies are available on request. Naturally, we cannot fill requests for literature which is copyrighted (such as books or journal articles) or whose distribution is restricted.

Special bibliographies and selected computer-printed abstracts of the literature in the RSIC system are available upon request. The Selective Dissemination of Information (SDI) Service is available by submitting a list of subject categories defining the recipient's interests.

## REACTOR AND WEAPONS SHIELDING

AI-73-31; NASA-CR-121157; N73-23726
S8DR Shield Examination.
Mason, D.G.; McCurnin, W.R.
January, 1973
NTIS $4.75

AAEC/E-279
Dense Plasma Focus as a Pulsed Deuterium-Tritium Neutron Source.
Hogg, G.R.
April, 1973
AAEC Research Establishment, Lucas Heights

AD-768042
Additional Analysis of Neutron and Secondary Gamma Ray Transport in Concrete Using the SAM-CE Monte Carlo Code.
Cohen, M.O.; Beer, M.
September, 1973
NTIS $3.00

AD-768493/9
Fast-Neutron Transport in Shields Having Limited Transverse Dimensions.
Bass, L.P.; Bozin, G.M.; Germogenova, S.; Degyarev, F.; Suvorov, A.P.
October 9, 1973
NTIS $3.25

AERE-R-7209
Fission Product Chain Yields from Experiments in Thermal Reactors.
Crouch, E.A.C.
January, 1973
Atomic Energy Research Establishment, Harwell

AREAEE-166
Experimental Investigation of Space-Energy Distributions of Slow Neutrons in Water Near Plane Absorber.
Hussein, A.; El-Konsol, S.; Novikov, A.; Zakharkin, I.; Hamouda, I.
1973
NTIS

ANL/CTR-73-2
Accelerator Be-Target Neutron Sources for the CTR Materials Program.
Persiani, P.J.
October, 1973
Dep., NTIS $3.00

BNL-18236
Studies of Fusion Reactor Blankets with Minimum Radioactive Inventory and with Tritium Breeding in Solid Lithium Compounds: A Preliminary Report.
Powell, J.R.; Miles, F.T.; Aronson, A.; Winsche, W.E.
June, 1973
Dep., NTIS $8.50

BNL-18248
Preliminary Neutron Foil Dosimetry Characterization of the Blip Neutron Facility.
Parkin, D.M.; Dudey, N.D.; Heinrich, R.R.; Fluss, M.J.
August,1973
NTIS

CEA-N-1607 (In French)
Fitting of a Statistical Law to the Spatial Distribution of Slowing-Down Neutrons.
Buxerolle, M.; Zaborowski, H.L.
February,1973
Dep., NTIS (U.S. Sales Only) $3.75

CONF-710567, pp.72-75
General Principles for Operational Radiation Protection.
Jain, M.P.
August,1971
Institute of Nuclear Medicine and Allied Sience, Delhi (India)

CONF-711026; RISO REPORT No.249, Part III, pp.880-1229
Proceedings of the Third International Conference on Luminescence Dosimetry, Held at the Danish AEC Research Establishment, Riso 11-14 October 1971.
Mejdahl, V. (Ed.)
December,1971
Jul. Gjellerup, 87, Solvgade, DK-1307 Copenhagen K. Denmark

CONF-721018
Proceedings of the Fourth International Conference on Reactor Shielding (Paris 9th-13th October 1972).
Volume 1. Pages 1-233
Volume 2. Pages 234-624
Volume 3. Pages 625-1189
Volume 4. Pages 1190-1340
Volume 5. Pages 1341-1514
Bergonzi, Mme. J. (Ed.)
1972
OECD

CONF-721018, pp.41-57
Investigation of Intra-Vessel Shielding for the BN-600 Reactor of the BFS-2 Rig.
Bakov, A.T.; Bondarenko, V.V.; Voropaev, A.I.; Efimov, E.P.; Kazansky, Yu.A.; Kulakovsky, M.Ya.; Savitsky, V.I.; Troyanov, M.V.; Fomenko, V.P.; Khokhlov, G.N.
October,1972
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CONF-721018, pp.58-67 (In French)
Conception Des Protections du Reacteur Phenix.
Culambourg, J.; Devillers, C.; de Ville, A.L.D.; Nimal, J.C.; Rastoin, J.
October,1972
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CONF-721018, pp.68-86
Shielding of Commercial Fast Reactors.
Brindley, K.W.
October,1972
OECD

CONF-721018, pp.87-102
Shielding Problems in the PEC Reactor.
Cicognani, G.; Tavoni, R.; Travaglini, N.; Corbo, M.; Farinelli, U.; Tondinelli, L.; Canali, U.; Nicks, R.; Penkuhn, H.; Ponti, C.
October, 1972
OECD

CONF-721018, pp.103-121 (In French)
Problemes Poses par la Definition des Protections d'Un Grand Reacteur Rapide de Puissance.
Moreau, J.
October, 1972
OECD

CONF-721018, pp.122-135
Gas-Cooled Fast Reactor Shield Design.
Haschke, D.; Ganapol, R.; Henniger, C.; Maeder, C.
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OECD

CONF-721018, pp.136-148
An Engineering Approach to Shield Design.
Van Halem, H.; Kuiken, R.
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CONF-721018, pp.149-165 (In French)
Protection Contre les Rayonnements et Ingenierie Nucleaire.
Jegu, J.; Maire, D.; Poitou, M.; Prouillac, C.
October, 1972
OECD

CONF-721018, pp.186-197
Shielding Characteristics of a Nuclear Powered Manned Space Station.
Keshishian, V.
October, 1972
OECD

CONF-721018, pp.198-214
Shielding Problems in Fusion Reactors.
Blow, S.; McCracken, G.M.
October, 1972
OECD

CONF-721018, pp.215-226
The Radiation Shielding Information Center (RSIC) - A Unifying Force in the International Shielding Community.
Maskewitz, B.F.; Trubey, D.K.; Roussin, R.W.; Clark, F.H.
October, 1972
OECD

CONF-721018, pp.226-233
ESIS: A European Shielding Information Service.
Nicks, R.; Ponti, C.
October, 1972
OECD

CONF-721018, pp.236-252
"Anatomy" of Radiation Transmission, Developed by a Semi-Analytical Simulation of Radiation Transport.
Biron, G.C.
October, 1972
OECD

CONF-721018, pp.268-283
Moments Method Calculations of Neutron and Gamma-Ray Penetration in Bulk Media.
Simmons, G.L.; Eisenhauer, C.; Spencer, L.V.
October, 1972
OECD

CONF-721018, pp.284-301 (In French)
Programme de Monte Carlo a Trois Dimensions Polycinetique Tripoli.
Bourdet, L.; Katz, S.; Nimal, J.C.; Vergnaud, T.
October, 1972
OECD

CONF-721018, pp.302-338
A Solution of the Coupled Integral and Differential Form of the Transport Equation Using a Legendre Polynomials Source Density Approximation in Angle and Space for Slab Geometry.
Stepanek, J.; Ligou, J.
October, 1972
OECD

CONF-721018, pp.370-395
New Modifications of the Monte-Carlo Method for Use in Calculating Differential Characteristics of the Gamma Ray Field Outside Reactor Shielding.
Viktorov, A.A.; Zolutukhin, V.G.; Klimanov, V.A.; Kochanov, V.A.; Ksenofontov, A.I.; Mashkovich, V.P.; Orlov, N.Yu.; Panchenko, A.M.; Sinev, V.N.
October, 1972
OECD

CONF-721018, pp.409-433
The Use of Modular Codes in a Unified Scheme of Shield Design Procedures for Commercial Reactor Plant.
Miller, P.C.; Grimstone, M.J.; Packwood, A.; Puch, I.G.
October, 1972
OECD

CONF-721018, pp.434-441
SHIELD - A Monte Carlo Program to Study Neutron Transport in Slab Geometries Using a PDP-9 Computer.
Adams, R.J.; Henry, L.C.; Lokan, K.H.
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OECD

CONF-721018, pp.442-455
Interactive Computer Graphics Applied to Shield Design.
Goddard, A.J.H.; Wrigley, M.F.
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CONF-721018, pp.488-500 (Figs.482,484-487)
Spatial, Energy and Angular Distributions of Radiations in Reactor Shielding.
Abagyan, A.A.; Bass, L.P.; Germogenova, T.A.; Degtyarev, S.F.; Dorofeev, A.A.; Dubinin, A.A.; Zhuravlev, V.I.; Kobozev, M.A.; Kostin, E.I.; Orlov, V.V.; Pupko, V.Ya.; Suvorov, A.P.; Tarasov, V.V.; Utkin, V.A.; Khmylev, A.N.; Tsypin, S.G.; Yarmakhov, I.G.
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CONF-721018, pp.501-537
A Modified Form of Diffusion Theory for Use in Calculating Neutron Penetration in Practical Shields.
Bendall, D.E.; Cripps, S.J.
October, 1972
OECD

CONF-721018, pp.557-578 (Figs.551-556)
Characteristics of Secondary Gamma Radiation from the Heavy Component of Nuclear Reactor Shielding.
Abagyan, A.A.; Vrazhnov, Yu.N.; Germogenova, T.A.; Degtyarev, S.F.; Dubinin, A.A.; Zhuravlev, V.I.; Kazansky, Yu.A.; Utkin, V.A.
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CONF-721018, pp.579-595
Application of the Invariant Imbedding Method to the Penetration Problems of Neutrons.
Nakai, M.; Komatsu, I.; Sato, Y.; Noguchi, F.; Shimizu, A.; Hada, M.
October, 1972
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CONF-721018, pp.596-610 (In French)
Attenuation en Ligne Droite des Particules Traitee par Une Methode de Monte-Carlo - Programme Tridimensionnel MERCURE-4.
Devillers, C.; Dupont, C.
October, 1972
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CONF-721018, pp.611-624
The Development of the Two-Dimensional Shielding Code "RASC-2D" and Its Application.
Tanaka, Y.; Suzuki, I.; Ohtani, N.
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CONF-271018, pp.628-646
Practical Analysis of Primary Concrete Shield for a PWR.
Su, S.-D.; Patti, F.
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CONF-721018, pp.647-663
Radiation Flux and Heating Levels in the HTGR.
Engholm, B.A.
October, 1972
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CONF-721018, pp.664-681 (In French)
Comportement Sous Rayonnements Neutroniques de Materiaux Destinés a la Protection des Reacteurs Nucleaires.
Yvars, M.; Degas, P.; Quetier, M.; Rousseau, G.
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CONF-721018, pp.682-701 (In French)
Propriétés Particulieres des Betons de Protection Influencant le Processus d'Attenuation des Rayonnements.
Dubois, F.
October, 1972
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CONF-721018, pp.702-716
Choice of Characteristic Element Components in Concrete for Shielding Calculations on Biological Shields for Power Reactors.
Schultz, H.; Vogt, H.G.
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CONF-721018, pp.731-740
Heat Generation in Shield Layers of High-Temperature Reactor.
Hofmann, K.; Wahsweiler, H.G.; Weicht, U.
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CONF-721018, pp.757-770 (Figs.751-756)
Reactor Shielding as a Source of Radiation.
Bolshakov, A.V.; Madeev, V.G.; Rudakov, AN.N.; Uksusov, E.I.
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CONF-721018, pp.771-787
Some Problems Associated with Calculating the On Load Dose Rates on the Pile Cap of an Advanced Gas Cooled Reactor.
Dutton, L.M.C.; Palit, A.
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CONF-721018, pp.812-827
In Situ Analysis of Shielded, Gamma-Active Material Using a Ge(Li) Spectrometer.
Spiers, E.W.; Lakey, J.R.A.
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CONF-721018, pp.828-859 (In French)
Calculation of the Circuits Activity in a High Temperature Reactor.
Brisbois, J.; Beslu, P.; L'Homme, A.
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CONF-721018, pp.860-880
Access to the Pressure Vessel of a Direct Cycle BWR for Inspection and Maintenance.
Hock, R.; Seifferth, L.
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CONF-721018, pp.881-897 (In French)
Activity and Shielding of the Blanket Gas Circuit of a Fast Power Reactor.
Costa, L.; Fremont, R. de
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CONF-721018, pp.898-913 (In French)
Activite des Produits de Corrosion dans l'Eau Primaire du Reacteur de Chooz.
Prejaville, G.; Lott, M.; Marchal, A.
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CONF-721018, pp.914-922 (In French)
Determination Continue du Taux d'Endommagement des Cuves de Reacteurs a Eau Pressurisee.
Devillers, C.; Dupont, C.
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CONF-721018, pp.923-936 (In French)
Protection des Installations de Manutention et des Circuits Auxiliaires du Reacteur Phenix.
Culambourg, J.; Godot, B.; Jegu, J.
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CONF-721018, pp.937-952
Investigations into the Gamma-Ray Attenuation in Concrete Shielding with Pipes.
Futtermenger, W.
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CONF-721018, pp.953-972 (In French)
Recent Methods for the Testing of Gamma Shields.
Fourcade-Cancelle, N.; Papot, L.
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CONF-721018, pp.973-995
Investigations of Shielding Problems Arising from Modifications to Refuelling Machinery at Trawsfynydd Power Station.
Pugh, I.G.; Hoverd, R.S.; Avery, A.E.
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CONF-721018, pp.996-1008
Comparison of DINE Neutron Attenuation Calculations with Experimental Results for a Wide Variety of Simulated Concrete Configurations for a Fast and Thermal Reactor Spectrum.
Verschuur, K.A.
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CONF-721018, pp.1009-1025 (In French)
Application des Methodes de Deplacement et Diffusion a des Problemes Concrets de Blindage de Reacteurs.
Herrnberger, V.; Tai, A.S.; Zuber, J.F.
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CONF-721018, pp.1026-1046
Control of Shield Calculation by Test Measurements in the Biological Concrete Shield of Three German Power Reactors.
Futtermenger, W.H.; Schultz, H.G.; Hecht, G.; Weber, H.; Hock, R.
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CONF-721018, pp.1047-1063
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