Research Reactor Utilization, Safety, Decommissioning, Fuel and Waste Management

Proceedings of an international conference
10–14 November 2003
Santiago, Chile

IAEA
International Atomic Energy Agency

The photograph, on which the cover is based, was taken by George Munro.

RESEARCH REACTOR UTILIZATION, SAFETY, DECOMMISSIONING, FUEL AND WASTE MANAGEMENT

The following States are Members of the International Atomic Energy Agency:

AFGHANISTAN
ALBANIA
ALGERIA
ANGOLA
ARGENTINA
ARMENIA
AUSTRALIA
AUSTRIA
AZERBAIJAN
BANGLADESH
BELARUS
BELGIUM
BENIN
BOLIVIA
BOSNIA AND HERZEGOVINA
BOTSWANA
BRAZIL
BULGARIA
BURKINA FASO
CAMEROON
CANADA
CENTRAL AFRICAN REPUBLIC
CHILE
CHINA
COLOMBIA
COSTA RICA
CÔTE D'IVOIRE
CROATIA
CUBA
CYPRUS
CZECH REPUBLIC
DEMOCRATIC REPUBLIC OF THE CONGO
DENMARK
DOMINICAN REPUBLIC
ECUADOR
EGYPT
EL SALVADOR
ERITREA
ESTONIA
ETHIOPIA
FINLAND
FRANCE
GABON
GEORGIA
GERMANY
GHANA
GREECE
GUATEMALA
HAITI
HOLY SEE
HONDURAS
HUNGARY
ICELAND
INDIA
INDONESIA
IRAN, ISLAMIC REPUBLIC OF
IRAQ
IRELAND
ISRAEL
ITALY
JAMAICA
JAPAN
JORDAN
KAZAKHSTAN
KENYA
KOREA, REPUBLIC OF
KUWAIT
KYRGYZSTAN
LATVIA
LEBANON
LIBERIA
LIBYAN ARAB JAMAHIRIYA
LIECHTENSTEIN
LITHUANIA
LUXEMBOURG
MADAGASCAR
MALAYSIA
MALI
MALTA
MARSHALL ISLANDS
MAURITANIA
MAURITIUS
MEXICO
MONACO
MONGOLIA
MOROCCO
MYANMAR
NAMIBIA
NETHERLANDS
NEW ZEALAND
NICARAGUA
NIGER
NIGERIA
NORWAY
PAKISTAN
PANAMA
PARAGUAY
PERU
PHILIPPINES
POLAND
PORTUGAL
QATAR
REPUBLIC OF MOLDOVA
ROMANIA
RUSSIAN FEDERATION
SAUDI ARABIA
SENEGAL
SERBIA AND MONTENEGRO
SEYCHELLES
SIERRA LEONE
SINGAPORE
SLOVAKIA
SLOVENIA
SOUTH AFRICA
SPAIN
SRI LANKA
SUDAN
SWEDEN
SWITZERLAND
SYRIAN ARAB REPUBLIC
TAJIKISTAN
THAILAND
THE FORMER YUGOSLAV REPUBLIC OF MACEDONIA
TUNISIA
TURKEY
UGANDA
UKRAINE
UNITED ARAB EMIRATES
UNITED KINGDOM OF GREAT BRITAIN AND NORTHERN IRELAND
UNITED REPUBLIC OF TANZANIA
UNITED STATES OF AMERICA
URUGUAY
UZBEKISTAN
VENEZUELA
VIETNAM
YEMEN
ZAMBIA
ZIMBABWE

The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

PROCEEDINGS SERIES

RESEARCH REACTOR UTILIZATION, SAFETY, DECOMMISSIONING, FUEL AND WASTE MANAGEMENT

PROCEEDINGS OF AN INTERNATIONAL
CONFERENCE ON
RESEARCH REACTOR UTILIZATION,
SAFETY, DECOMMISSIONING, FUEL
AND WASTE MANAGEMENT
ORGANIZED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
AND HOSTED BY THE GOVERNMENT OF CHILE
THROUGH THE
NUCLEAR ENERGY COMMISSION OF CHILE
AND HELD IN SANTIAGO, CHILE, 10–14 NOVEMBER 2003

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2005

Printed by the IAEA in Austria
June 2005
STI/PUB/1212

IAEA Library Cataloguing in Publication Data

International Conference on Research Reactor Utilization, Safety, Decommissioning, Fuel and Waste Management (2003 : Santiago, Chile)
Research reactor utilization, safety, decommissioning, fuel and waste management : proceedings of an International Conference on Research Reactor Utilization, Safety, Decommissioning, Fuel and Waste Management / organized by the International Atomic Energy Agency and hosted by the Government of Chile through the Nuclear Energy Commission of Chile and held in Santiago, 10–14 November 2003. — Vienna : The Agency, 2005.
p. ; 24 cm. — (Proceedings series, ISSN 0074-1884)
STI/PUB/1212
ISBN 92-0-113904-7
Includes bibliographical references.

1. Radioactive waste disposal — Congresses. 2. Hazardous wastes — Management — Congresses. 3. Nuclear reactors — Safety measures — Congresses. 4. Nuclear reactors — Decommissioning — Congresses. I. International Atomic Energy Agency. II. Comision Chilena de Energia Nuclear. III. Series: Proceedings series (International Atomic Energy Agency).

IAEAL 05-00407

FOREWORD

The IAEA has, since its inception, been promoting the exchange of scientific and technical information related to the safety and utilization of research reactors, as well as issues surrounding the research reactor fuel cycle. A major avenue for this exchange has been the periodic organization of seminars, symposia and conferences.

There are always a number of new and significant issues being faced by the research reactor community, not the least being the issues of ageing materials, equipment, structures and even staff faced by a mature technology. In addition, the community has recently had to deal with the issue of weapons grade high enriched uranium (HEU) in many facilities. This has diverted attention from the important research work, production of isotopes for medicine and industry, modification of materials and training of scientists and engineers routinely carried out by the research reactor community. The conversion of research reactors from HEU to low enriched uranium and the non-proliferation issues involving research reactors are adequately dealt with in annual international conferences.

The objective of this conference — 'Research Reactor Utilization, Safety, Decommissioning, Fuel and Waste Management' — was to focus instead on scientific and technical activities of the research reactor community. Held from 10 to 14 November 2003 in Santiago, Chile, the conference was organized by the IAEA and hosted by the Government of Chile through its Nuclear Energy Commission. It provided a forum for reactor operators, designers, managers, experimenters and regulators to share their experience, evaluate their findings and present recommendations for future activities in this area.

The conference provided comprehensive and up to date information on the current status of the research reactors. It is hoped that these proceedings will serve as a valuable source of information for specialists involved in research reactor work and for regulatory authorities in IAEA Member States. These proceedings contain a summary of the conference, the major findings, the opening addresses and the technical papers.. The poster presentations are provided on a CD-ROM that accompanies this volume.

The IAEA wishes to express its appreciation to all chairpersons and co-chairpersons of sessions and to those who presented the papers and posters for their contributions to the technical success of the conference, as well as to all of the authors of the papers compiled in these proceedings. Special thanks go to the Government of Chile for hosting and co-sponsoring the conference through the Nuclear Energy Commission of Chile, for their hospitality, and to the local coordinator, R. Muñoz and his team.

The organization of the conference involved the following IAEA scientific secretaries: S. Paranjpe, Division of Physical and Chemical Sciences; T. Hargitai, Division of Nuclear Installation Safety; and R. Burcl; Division of Nuclear Fuel Cycle and Waste Technology. The officer responsible for the conference was I. Ritchie, Cross-cutting Coordinator for Research Reactors. F.N. Flakus compiled and prepared the proceedings in close cooperation with P. Adelfang, Division of Nuclear Fuel Cycle and Waste Technology.

EDITORIAL NOTE

CONTENTS

CONFERENCE SUMMARY 1

OPENING SESSION

Opening Address 13
A. Dulanto Rencoret
Opening Address 15
R. Hojman
Opening Address 17
T. Taniguchi

RESEARCH REACTOR SAFETY (Session 1)

Safety features of the replacement research reactor
(IAEA-CN-100/32) 25
M. Summerfield, V. Garea
A risk perspective for the replacement research reactor in Australia
(IAEA-CN-100/17) 45
J. Barón, J. Núñez Mc Leod, S. Rivera
Assessment of the safety options for the Jules Horowitz Reactor (RJH)
(IAEA-CN-100/70) 57
D. Rive, L. Bertrand, J. Couturier
Safety aspects of cold neutron sources in research reactors
(IAEA-CN-100/72) 71
K.H. Gobrecht, J.-M. Astruc
Research reactors that look similar but are quite different
(IAEA-CN-100/14) 79
C.M. Mazufri, A.S. Doval
Transient thermal-hydraulic modelling of ETRR-2 for off-site power loss
(IAEA-CN-100/64) 91
A.H. Mariy, S.E. El-Morshdy, H.A. Khater, A.A. Abdel-Raouf
Regulatory control of research reactors and critical assemblies
in Argentina (IAEA-CN-100/07) 107
C.D. Perrin
Focus on defence in depth in regulatory assessment
(IAEA-CN-100/33) 119
B. Lawrence, V. Diamond, D. Macnab

Safety requirements applied to research reactors in France
(IAEA-CN-100/69) 131
H. Abou Yehia, G. Bars
Regulatory aspects and experience with Russian research reactors
(IAEA-CN-100/123) 145
S.I. Morozov
Regulatory practice in a country operating one nuclear research reactor
(IAEA-CN-100/108) 151
A.T. Mikulski
Safety re-evaluation of research reactors:
A new trend in license requirements (IAEA-CN-100/104) 167
J.P. Boogaard, H.T. Wiersema
Challenges in updating a generic safety analysis report after 22 years of
technological and regulatory progress (IAEA-CN-100/53) 177
G.G. Kennedy, D.J. Winfield
ISO 9001 and ISO 14001. An integrated quality management system
for an MTR facility: SAFARI-1 research reactor
(IAEA-CN-100/129) 185
J.F. du Bruyn, C.S.B. Piani
Reactor protection systems: Diverse approaches
(IAEA-CN-100/25) 193
C.A. Verrastro, D.S. Estryk, J.C. Carballido
Thermohydraulic design and safety analysis of research reactors
(IAEA-CN-100/137) 205
A. Hainoun
Experience with probabilistic safety assessment for the BR2 reactor
(IAEA-CN-100/39) 217
S. Declercq, P. Gubel, F. Joppen, E. Koonen
Engineering considerations and experience related to
refurbishment and safety upgrading of the 40 MW(th)
CIRUS research reactor (IAEA-CN-100/87) 227
A.V. Kharpate
Comprehensive in-service inspection of Kartini reactor tank liner and
fitness for continued operation (IAEA-CN-100/89) 243
S. Syarip, P. Stathers
Development of modern safe systems of work at the
Imperial College Reactor Centre and their application
to neutron detector testing and nuclear training courses
(IAEA-CN-100/132) 249
D.S. Bond, L. Howard, H.J. Phillips, S.J. Franklin, N.J. Chapman

Safety Culture Assessment Programme:
Statistical analysis of a survey conducted at the
IEA-R1 Brazilian research reactor (IAEA-CN-100/48) 263
P.S.P Oliveira, R.N. Saxena
Improving the safe operation and new open policy at the VR-1 reactor
(IAEA-CN-100/58) 279
L. Sklenka, K. Matejka, M. Kropík

RESEARCH REACTOR UTILIZATION (Session 2)

Indian experience in neutron beam utilization: In-house and
collaborative programmes (IAEA-CN-100/85) 291
M. Ramanadham, S.K. Paranjpe, S.L. Chaplot, R. Mukhopadhyay
Geochronology by the $^{40}Ar/^{39}Ar$ method at the Sernageomin
Laboratory, Santiago, Chile (IAEA-CN-100/55) 299
C. Pérez de Arce, S. Matthews, J. Klein
FiR 1 reactor in service for boron neutron capture therapy (BNCT)
and isotope production (IAEA-CN-100/67) 313
I. Auterinen, S. Salmenhaara
Role of the TAPIRO fast research reactor in neutron capture therapy
in Italy: Calculations and measurements (IAEA-CN-100/97) 325
G. Rosi, S. Agosteo, K.W. Burn, L. Casalini, E. Nava, M. Sarotto,
M. Cortesi, G. Gambarini, S. Gay, F. D'Errico
A coupled calculational system for optimal in-core fuel management in
research reactors (IAEA-CN-100/02) 339
H. Mazrou, M. Hamadouche
Neutronic design of a cold neutron source with MCNP
(IAEA-CN-100/12) 353
D.F. Hergenreder, C. Lecot, O. Lovotti
Statistical processing of calculated reaction rates (IAEA-CN-100/61) ... 365
Z. Lahodova, J. Ernest, S. Flibor, M. Marek, J. Rataj, L. Viererbl
Thermal neutron self-shielding factor in foils: A universal curve
(IAEA-CN-100/111) 375
I.F. Gonçalves, E. Martinho, J. Salgado
A set of programs for experimental data processing of
focusing configurations (IAEA-CN-100/117) 383
I. Ionita, N.C. Popa
Core management and calculation tools for the WWR-M
research reactor in Ukraine (IAEA-CN-100/131) 395
Y.P. Mahlers

Neutronic design of the first core of the replacement research reactor (IAEA-CN-100/18) 403
E. Villarino, S. Gomez, V. Ishida
Optimization calculations at TR-2 (IAEA-CN-100/141) 415
M.H. Turgut, A. Aytekin
Core management of the Dalat nuclear research reactor (IAEA-CN-100/134) 429
Pham Van Lam, Le Vinh Vinh, Huynh Ton Nghiem, Nguyen Kien Cuong, Luong Ba Vien
Reactor simulator development facility for operating personnel training (IAEA-CN-100/15) 441
A. Etchepareborda, C.A. Flury, F. Maciel, D. Alegrechi, G. Ibarra, M. Muguiro, F. Lema, M. Giménez, M. Schlamp, A. Vertullo
Utilization programme of Hanaro: Present and future (IAEA-CN-100/102) 457
Y.J. Kim, B.J. Jun, K.B. Park, I.H. Kuk
Enhancement in the utilization of IEA-R1 research reactor for radioisotope production and research (IAEA-CN-100/43) 467
R.N. Saxena, R. Fulfaro
New IPEN–CNEN/SP neutron powder diffractometer (IAEA-CN-100/45) 477
C.B.R. Parente, V.L. Mazzocchi, Y.P. Mascarenhas
Eight years of neutron activation analysis in Ghana using a low power research reactor (IAEA-CN-100/80) 491
B.J.B. Nyarko, E.H.K. Akaho, Y. Serfor-Armah, E.K. Osae
Extensive utilization of training reactor VR-1 (IAEA-CN-100/59) 499
K. Matejka, L. Sklenka
Teaching and training at RA-6 reactor and their contribution to the research facilities development (IAEA-CN-100/20) 507
O. Calzetta Larrieu, H. Blaumann
The contribution to international safeguards and security by a small university research reactor (IAEA-CN-100/34) 515
H. Böck, M.Villa
Current tendencies and perspectives of the development of research reactors in the Russian Federation (IAEA-CN-100/125) 527
N.V. Arkhangelsky, B.A. Gabaraev, V.V. Kchmelschikov, Yu.N. Kuznetsov, I.T. Tretiyakov

RESEARCH REACTOR DECOMMISSIONING AND WASTE MANAGEMENT (Session 3)

International radiation safety recommendations on decommissioning (IAEA-CN-100/84) 545
L. Jova Sed, D. Reisenweaver

Implementation of Stage-3 decommissioning of the Triton facility (IAEA-CN-100/75) 557
E. Lopes, L. Pillette-Cousin

Experience with partial decommissioning of PARR-1 (5 MW) for core conversion and power upgrade (IAEA-CN-100/107) 569
M. Israr, Q.D. Shami, S. Pervez

Decommissioning Russian research facilities (IAEA-CN-100/121) 573
S.I. Morozov

SAFARI-1: Planning for MTR decommissioning according to IAEA guidelines (IAEA-CN-100/128) 579
C.S.B. Piani, L.J. Shayi

International decommissioning strategies (IAEA-CN-100/83) 593
L. Jova Sed, D. Reisenweaver

'Green Vinca' — Vinca Institute nuclear decommissioning programme (IAEA-CN-100/135) 603
M. Pešic, K. Subotic, V. Ljubenov, O. Šotic

Innovation in radioactive wastewater stream management (IAEA-CN-100/62) 613
A. Karameldin

FUEL MANAGEMENT (Session 4)

Russian Research Reactor Fuel Return Programme (IAEA-CN-100/143) 631
T. Dedik, I. Bolshinsky, A. Krass

Effect of coupon orientation on corrosion behaviour of auminium alloy coupons in the spent fuel storage section of the IEA-R1 research reactor (IAEA-CN-100/44) 637
O.V. Correa, R.M. Lobo, S.M.C. Fernandes, G. Marcondes, L.V. Ramanathan

Study of the effect of sedimented particles on the corrosion behaviour of aluminium clad spent fuel during storage in water (IAEA-CN-100/10) 645
S. Rodríguez, L. Lanzani, A. Quiroga, R. Haddad, E. Silva

Encapsulation of nuclear spent fuel for semi-dry storage at the Budapest research reactor (IAEA-CN-100/81) 659
S. Tözsér, T. Hargitai, I. Vidovszky
Corrosion of aluminium-clad spent fuel at RA research reactor (IAEA-CN-100/136) 677
M. Pešic, T. Maksin, R. Dobrijevic, Z. Idjakovic

Chairpersons of Sessions and Secretariat of the Conference 687
Programme Committee 689
List of Participants 691
Author Index 715
Index of Papers by Number 717

CONFERENCE SUMMARY[1]

BACKGROUND

More than 250 research reactors are operating at present around the world. The level of their operation, utilization, safety and management differs substantially, depending also on the individual situation in the particular country. With the aim to promote discussion and information exchange on technical and safety aspects of research reactor utilization, safety, decommissioning, management of spent fuel and radioactive waste generated during their operation and utilization, the Agency is regularly organizing technical meetings on research reactors and the international Conference at Chile was a major milestone in this series of technical meetings.

PARTICIPATION AND PROGRAMME

The conference attracted a wide spectrum of participants: scientists, designers and operators of nuclear research reactors, and representatives of decision making organs and regulatory authorities. This combination of scientists, technologists and regulators provided a unique opportunity to discuss various aspects from both the theoretical as well as the technological point of view.

At its opening, the conference was addressed by Alfonso Dulanto Rencoret, Minister of Mining, Government of Chile, Roberto Hojman, President of the Nuclear Energy Commission of Chile (CCEN) and by Tomihiro Taniguchi, Deputy Director General for Nuclear Safety and Security of the International Atomic Energy Agency (IAEA). Many officials attended the Inaugural Session from the Government, senior members of the Nuclear Energy Commission of Chile, academia and the press.

The Minister referred to the importance placed by the Government of Chile on Science and Technology for improving the quality of life, and to the role of the IAEA in ensuring nuclear safety for the world. There was a mention of appreciation for the nuclear contributions to humanity, but with due concern regarding waste management and final disposal of radioactive waste. The President of CCEN enumerated the beneficial applications of radioisotopes

[1] The views and recommendations expressed in this summary are those of the conference participants, and do not represent those of the IAEA.

and radiation technology, and the respective utilization of research reactors. Mr. Taniguchi underlined the fact that the IAEA is authorized to encourage and assist in the development and practical application of research related to the peaceful uses of nuclear science and technology, stressed the need for consideration of how to better utilize these technological tools, that the IAEA can assist national authorities in developing realistic strategic plans but that the research reactor community must take the initiative and generate creative solutions.

A total of 153 participants from 39 countries attended the conference. Over 100 technical papers and posters were presented during the conference in four topical sessions on: (1) research reactor safety; (2) research reactor utilization; (3) research reactor decommissioning and waste management; and (4) research reactor fuel management.

The conference concluded with a panel discussion to explore the main findings, and to prepare recommendations to the research reactor community and to the IAEA to help guide future support.

SESSION 1: RESEARCH REACTOR SAFETY

Scope

Papers presented during the Session 1 on research reactor safety covered a variety of aspects: safety features of research reactors, risk perspectives, safety options of planned facilities, regulatory framework, thermal hydraulic calculations, safety assessment and evaluation, safety of experiments, in-service inspection, refurbishment experience, new structure of reactor protection system. In particular, several papers addressed the very high standard of safety requirements being applied for new and modem research reactors. Although considered to be very important, only one paper addressed safety impacts of experiments and experimental set-ups, safety culture, and physical protection.

Conclusions

From the presentations related to the design, safety analyses and safety features of new reactors under construction or planned, it could be seen that mechanical effects and fire hazards resulting from aircraft crashes have to be considered in the design base accidents caused by external events.

The presentations dealing with probabilistic safety analysis (PSA) techniques showed that these are more frequently used, especially for research reactors with higher power levels. It was stressed that PSA is a valuable tool to be used also at research reactors, but that precautions needed to be taken in the interpretation of the results: complete sets of reliable data are not yet available; most research reactors are unique and consequently it is not easy to obtain reliable data; human behaviour plays a dominant role in the safe operation of a research reactor facility. It is not questioned that PSA can facilitate a better understanding of the behaviour of the installation and that it is a powerful tool to indicate the strengths and weaknesses of a facility and its operation. Using PSA in addition to deterministic safety analyses can indicate unknown deficiencies, which could cause unexpected safety problems. To ensure a more reliable outcome it is desirable to improve the reliability of data in the available databases.

Although verified codes had been used, some presentations showed conflicting results of thermal-hydraulic calculations. The development of a benchmark programme in order to compare thermal hydraulic analyses and to prepare detailed guidelines for modelling and performing thermal hydraulic analyses was strongly recommended.

Especially for research reactors with higher power levels the regulatory body has recently required a periodic safety review. Due to the systematic evaluation of all safety related technical, operational, personnel and administrative requirements, and the implementation of improvement measures, the safety of those facilities comply with the state of art regulations. To assist smaller research reactors in performing periodic safety reviews it is recommended that guidance should be developed, e.g. based on draft guidance used for Integrated Safety Assessment of Research Reactors (INSARR) missions.

SESSION 2: RESEARCH REACTOR UTILIZATION

Scope

Under this topic, several very impressive research reactor utilization records were presented, covering utilization for training, research and development, radioisotope production and neutron based research. Extensive possibilities in the use of neutrons from research reactors were described, including capacity buildup for associated instrumentation.

A number of papers dealt with configuration and optimization of core design so as to suit specific applications. It was evident that the major thrust of application will dictate the choice of core configuration in the future. Neutron activation analysis, including the technique of prompt gamma ray usage, is attractive for wide ranging applications, e.g. in the mining industry. A number of papers dealt with theoretical calculations of reactor physics and building up of beam line instrumentation.

Conclusions

There was a general consensus that the utilization of a number of research reactors could be considerably improved. This could be achieved on a national, regional and international level. Sustained interest in enhanced utilization of medium powered research reactors was obvious. Training, teaching and education constitute a major part of the utilization of low powered research reactors. Expertise in developing and constructing beam line instrumentation was highlighted; such experience should be shared, at least on a regional basis. Low power research reactors could be used for benchmark experiments, which could later be transferred to specialized medium to high power research reactors. Emerging trends indicate that in the future there will be essentially three categories of research reactors, the core configuration of which will, in turn, be dictated by the respective application area:

(1) Educational research reactors (academic);
(2) Commercial research reactors (services);
(3) Advanced application research reactors (high tech).

Support was needed through regional cooperation efforts in promoting reliability and self-sufficiency in the utilization of research reactors. This would facilitate regional and interregional thematic collaboration for enhanced utilization of research reactors, aid in carrying out regional strategic planning for the utilization and promotion of regional *centres of excellence,* and for resource sharing for regional self-sufficiency.

SESSION 3: DECOMMISSIONING AND WASTE MANAGEMENT

Scope

An international perspective on decommissioning was provided. For over 20 years the IAEA has been developing guidance and technical information relating to the decommissioning of nuclear facilities. During this time, the international concept of decommissioning strategies has changed. Three basic decommissioning strategies are envisaged as possibilities for nuclear installations:

(1) Immediate dismantling;
(2) Deferred dismantling;
(3) Entombment.

The basic approach of these three strategies was discussed, immediate dismantling being the generally preferred option.

Many research reactors that were built and put into operation more than 30 years ago have approached the end of their life, are being decommissioned or need to be partially or completely decommissioned. Owing to the lack of financial resources, many research reactors are still waiting to be decommissioned. Broad practical experience was reported on this issue, on step-wise decommissioning of nuclear research reactors and facilities, on partial decommissioning and on dealing with problems of ageing equipment and corrosion.

An extensive decommissioning programme, including decommissioning of nuclear facilities of the front end and the back end of the nuclear fuel cycle, and experimental reactors was presented. A selected dismantling scenario was provided, taking into account a number of significant non-radiological hazards. Preliminary dismantling operations performed on concrete structures and the decontamination of hot cell walls was described and it was stated that significant experience feedback has been acquired.

Work accomplished under a decommissioning concept was outlined whereby a plant is being completely dismantled in ten steps of which the first eight steps have been completed: the fuel elements and sodium coolant were removed; facilities and systems no longer required were shut down; the cooling towers and machine hall were demolished; secondary and primary sodium circuits were completely disassembled; and the rotary lid of the vessel was dismantled. Only the reactor vessel with its internals, the primary shield and the biological shield are still inside the plant and will be dismantled as part of the ninth step.

Aluminium alloys and stainless steels have been used as cladding materials for nuclear fuel in research reactors. A surveillance programme developed for a TRIGA reactor to obtain a fundamental understanding of the related corrosion problems was described, consisting of in-pool tests using corrosion surveillance coupons made of aluminium alloys and stainless steel.

Conclusions

Decommissioning of research reactors and related facilities is a complicated process, requiring the equal involvement of operator and regulator. It also needs proper strategic planning. The decommissioning plan and relevant safety and quality assurance documentation must be prepared well in advance as a solid basis for the development of a detailed work plan for particular operations. The decommissioning plan needs to be regularly revised and updated to reflect the real situation at the facility and recent development of technologies and safety rules. Proper funding needs to be assured.

Various waste flow streams are generated during research reactor decommissioning; in some cases chemical toxicity of waste should also be considered (e.g. beryllium, asbestos, etc.). Waste minimization is a key issue; careful segregation of inactive bulk material (mostly construction materials) ensures for significant savings.

The operator needs to ensure proper, safe and feasible management of all types of waste, generated in the operation and utilization of research reactors, considering the potential utilization of waste handling and processing technologies in a decommissioning phase. All waste lifetime steps (in particular, decontamination, sorting, segregation, processing, storage and eventually final disposal) need to be considered in an integrated manner.

Safe and sound technologies exist for the management of all types of radioactive waste, generated in research reactor operation, utilization and decommissioning. Obviously, more advanced technologies provide for the reduction of the conditioned waste volume for storage and disposal., However, their implementation increases the price for waste processing. The research reactor operator needs to select an optimum composition of equipment to cover all actual and foreseen requirements, well harmonized with other waste management capabilities and the waste management policy of the country.

The IAEA provides regular technical support in all the above mentioned activities through the dissemination of technical information, as well as through direct technical assistance. It has made available a full set of safety documents for decommissioning and waste management, starting from the safety requirements (the top level) through to safety guides (implementation level). Complementary sets of technical reports and documents, dealing with relevant decommissioning and waste management techniques, technologies and strategies, are also available and provide sufficient information to perform all activities.

A great deal of experience has been collected in the last decade with respect to decommissioning and refurbishment at several research reactors. The IAEA is capable of playing a special role in the dissemination and transfer of information and knowledge, and in organizing seminars and workshops at the facilities where operations are performed.

SESSION 4: RESEARCH REACTOR FUEL MANAGEMENT

Scope

This session on fuel management was opened with a keynote presentation on the return of Russian origin research reactor fuel, in which the IAEA's role was highly appreciated. Also presented was a machine developed to automatically dry and encapsulate WWER type spent fuel, which could be adapted to deal with other fuel types. Further presentations highlighted the work carried out in the monitoring of water chemistry and the minimization of corrosion in research reactor spent fuel pools within the frameworks of the IAEA's regional technical cooperation (TC) project in Latin America and the Coordinated Research Project on the corrosion of aluminium-clad research reactor fuel in water.

Conclusions

The session very effectively underlined the need for a solution to the back end of the research reactor fuel cycle and endorsed the efforts of the IAEA in trying to arrange the return of research reactor fuel to the country where it was originally enriched as the best possible option for most countries.

Furthermore, the session endorsed the ongoing activities of the IAEA to help ensure that the fuel is safely stored under the best possible conditions to minimize physical degradation until such time as a final solution can be arranged.

PANEL DISCUSSION AND OVERALL FINDINGS

In a final panel discussion session, the highlights of all topical sessions held during the conference were presented. A consensus outcome was elaborated and subjected to comment by all participants. The recommendations and overall findings resulting from this process are summarized below.

Recommendations to the research reactor community

The recommendations made to the research reactor community, with advice and assistance from the IAEA, are as follows:

— It is highly desirable for all countries with at least one research reactor to adopt the ‘Code of Conduct for the Operation and Utilization of Research Reactors’;
— States should have a strong, independent regulatory body and the relevant legal infrastructure;
— States should strengthen physical security at research reactors and research reactor fuel cycle facilities;
— States should consider life-cycle issues and how to improve the utilization of research reactors through the formulation and periodic updating of:
  - strategic plans for the utilization of research reactors;
  - fuel management plans;
  - ageing management plans;
  - refurbishment or modernization plans;
  - decommissioning plans;
  - operational, utilization and decommissioning waste management plans.
— Carry out regional strategic planning for the utilization of research reactors and the promotion of ‘centres of excellence’;
— Consider resource sharing for regional self-sufficiency (e.g. in radioisotopes).

Overall findings

(1) The feasibility of regional strategic planning for the utilization of research reactors and the promotion of regional ‘centres of excellence’, especially in resource sharing for regional self-sufficiency (e.g. radioisotopes) should be carefully investigated.

(2) Support of RERTR and its non-proliferation goals including:
— Core conversion from HEU to LEU;
— Target conversion from HEU to LEU;
— Development and qualification of new, high-density fuels
should be continued and strengthened.

(3) The repatriation of research reactor fuels to the country of origin will require new donor countries to support the “Tripartite Initiative” (involving the IAEA, Russian Federation and the USA) to return Russian origin fuel, referred to as the Russian Research Reactor Fuel Return (RRRFR) Programme, and to broaden its scope to include experimental and exotic fuels.

(4) The feasibility of regional and international solutions to the back end of the research reactor fuel cycle needs to be carefully assessed.

(5) Regional and international networks for knowledge preservation and sharing of experience and expertise should be set up, specifically for research reactors.
(6) The issue of ageing staff at research reactors and how younger replacements can be encouraged and trained needs to be addressed.
(7) A database on regulatory resources with the aim of sharing regulatory know-how, particularly among the smaller regulatory bodies, should be compiled and made available on the Internet.
(8) The projected needs for research reactors in the long term, for example over a 2025 or 2030 horizon, should be assessed and the results made available to planners and decision makers.

Overall conclusions

It was concluded that the Conference provided comprehensive and up to date information on the current status of the subject. Nevertheless, it was concluded that in any future international conference on research reactors, physical security and safety culture issues be addressed in some detail in the sessions on safety. A relatively large number of participants indicated that the conference represented a valuable source of information for operators and also regulatory authorities in many Member States. The IAEA demonstrated its role as an international coordinator and promoter of research reactor issues. Its activities in the coordination and implementation of several difficult tasks were appreciated. A strong requirement to continue these activities was clearly voiced in the closing discussion at the conference.

BLANK

OPENING SESSION

BLANK

OPENING ADDRESS

A. Dulanto Rencoret
Minister of Mining,
Santiago, Chile

On behalf of the Government of Chile, I wish to cordially welcome the authorities and scientists involved in the field of nuclear science and technology who have gathered today in Chile from 50 countries to discuss the use of experimental nuclear reactors and their contribution to scientific and technological development. Naturally, I also wish to convey to the IAEA our thanks for selecting Chile, through its Commission on Nuclear Energy, to be the host of this important international conference.

I would like to begin by explaining the context in which this meeting is taking place.

The Government which I have the honour of representing places great importance on scientific and technological development, referring to it in its programme as one of the fundamental pillars for national development, since it is impossible to maintain a competitive productive and exporting sector, or to raise the quality of life of our fellow citizens, without the assistance of science and technology. Conscious of this fact, President Ricardo Lagos, while addressing national business executives last week, issued a call for renewal of the effort being made in research, urging the private sector to increase investment in this area.

Investment in science and technology is part of the action being taken by our Government to enable Chile to reach the status of a developed country. Another action measure has been the major increase in State and private investment in education and infrastructure, to mention only some of the priority areas for national development. In the field of education alone, public expenditure has more than tripled during the past decade.

The level of investment achieved in infrastructure, on the other hand, is unparalleled in our country, thanks to private investment by major agreements with those countries in which there is greater demand for their use.

Chile values the efforts of the IAEA in all fields, especially in terms of implementing safeguards to prevent the proliferation of nuclear weapons in the world. I consider this a timely opportunity to reiterate the Government of Chile's iron commitment to exclusively peaceful uses of nuclear energy, as well as our firm opposition to any deviation of this form of energy towards military ends.

We also acknowledge the important work achieved through technical cooperation, of which our country has been a beneficiary ever since the initiation of nuclear activity in Chile almost three decades ago.

I cannot keep from mentioning the fact that even though nuclear energy represents a major contribution to humanity through the use of isotopes and ionizing radiation, as well as through the generation of electricity throughout the world, our country recognizes that there are some issues pending which must be resolved, taking into account the perspective of developing nations. I refer to the management and final disposal of highly radioactive waste, and also to the cross-border transport of radioactive materials. Both of those issues are rejected by public opinion, making social validation of this form of energy difficult.

Esteemed scientists, in closing I would like to convey to you my wishes that the conversations held and agreements entered into during this week may lead to major progress in the use of nuclear research reactors, and that this in turn may contribute to improving the quality of life worldwide, as well as to reinforcing international cooperation.

I also hope that your stay in our country will be an enjoyable one, that you may have the opportunity to become a little more acquainted with its natural beauty and culture, and that through the focus on the particular scientific issues which brings you here, it may become possible to achieve progress towards the genuine integration of different peoples and cultures sharing a common ideal: that of putting nuclear science and technology at the service of humankind.

Thank you very much.

OPENING ADDRESS

R. Hojman
President, Council of Directors,
Nuclear Energy Commission of Chile,
Santiago, Chile

I have the high honour of addressing you in order to express on behalf of the Nuclear Energy Commission of Chile (CCHEN), and also on my own behalf, our pleasure in having the opportunity to welcome to our country this select group of authorities and professionals of the IAEA, and more than 100 representatives of nuclear institutions from some 50 countries.

We are certain that this conference will facilitate a valuable exchange of knowledge in areas as important as the use of research reactors, safety related aspects of such use, the fuel cycle, and dismantling and management of radioactive waste.

We also value the opportunity provided by this forum to share experiences, exchange opinions, and discuss options and priorities during the five technical sessions scheduled as part of the conference and, moreover, to deepen existing and create new bonds of international cooperation.

I would like to mention the fact that our country has experience in the operation of its experimental reactors and has managed to make contributions to major applications in various fields including medicine, agriculture, industry and mining, and environment and metrology.

Health applications have been given priority among the CCHEN's activities, and this is why the use of the reactor at La Reina Nuclear Studies Center has focused on the production of radioisotopes and radio-pharmaceutical applications which have short half-lives and are used in the diagnosis of the dynamic functioning of various organs and as therapeutic agents in some types of cancer.

In agriculture, various isotope techniques have been investigated and fine tuned in areas such as soil fertility, fertilizing sources, optimum use of nutrients, rationalization and economy in the use of water, quantification of the degree of soil erosion, and irrigation with fertilizers. More recently, a study of residuality and mobility of agro-chemicals in soil and water was initiated through a project developed jointly with the IAEA and with our country's Agricultural and Livestock Raising Service, for the purposes of contributing regulations and standards in the use of pesticides.

In the area of mining, the CCHEN has been using radioactive tracers in mining and metallurgical processes for over two decades in determination of residence times, flow rates, fluid velocity, and characterization of runoff in in situ leaching processes, among other techniques, all of which has made it possible to optimize processes and achieve cost reductions.

In the environmental field, isotope techniques have been developed for the study of both surface and underground aquifer resources. Work has also been carried out in determining contaminant sources and, more recently, there has been participation in multi-disciplinary research in order to collaborate in the control of marine toxins responsible for red tides through the use of isotope techniques.

In the area of chemical metrology, the CCHEN renders major assistance services to the national export system in terms of sanitary certification, giving support and technical assistance to national laboratories so that the latter may raise their standards and undertake measurements in compliance with the growing requirements imposed by international markets. Worth pointing out are both the organization of aptitude drills and inter-comparison rounds at the national and international levels, and the CCHEN's capacity to prepare reference and control materials in natural matrices and also control materials for chemical analyses, all of which has enabled the CCHEN's laboratories to attain formal recognition as the reference laboratory for determining trace elements in biological samples.

I have mentioned only some of the technological developments associated with the operation of our reactors for the purposes of contextualizing the importance that various topics we are to discuss during the current week have for us.

I want to end by reiterating that we feel honoured and experience great pleasure in having a select group of researchers, scientists and professionals with us who have travelled from neighbouring as well as from remote countries in order to meet in Chile and focus the highest level of discussion in the world on dealing with matters related to nuclear research reactors.

Esteemed colleagues and friends, I wish your stay in Santiago to be the source of much benefit and also, to the extent allowed by the schedule, of much enjoyment and pleasure.

Thank you very much.

OPENING ADDRESS

T. Taniguchi
Deputy Director General,
Department of Nuclear Safety and Security,
International Atomic Energy Agency,
Vienna

On behalf of the Director General of the IAEA, it is my pleasure and privilege to welcome you to this International Conference on Research Reactor Utilization, Safety, Decommissioning, Fuel and Waste Management. I would like to offer my sincere thanks to the Government of Chile, and the Nuclear Energy Commission of Chile, for hosting this Conference in the beautiful and historic city of Santiago de Chile. I would also like to thank you, the 180 or so registered delegates from around the world, for participating in this conference. I trust that you will have an interesting and enjoyable week.

For more than 50 years, research reactors have been one of the locomotives of nuclear science and technology. To date, approximately 670 research reactors have been built, and some 270 of these reactors, in 59 countries, continue to operate today. Altogether, over 13 000 reactor years of operational experience have accumulated during this period. Just as important, however, is the fact that those reactors have operated in a remarkably safe manner.

The IAEA's statutes charter it to promote the contributions that atomic energy can bring to the health and prosperity of people throughout the world. Thus, the IAEA is authorized to encourage and assist in the development and practical application of research related to the peaceful uses of nuclear science and technology.

From its inception in 1957, there has been a broad interest at the IAEA in the benefits to be derived by Member States from the safe operation of research reactors. The multi-disciplinary research that a research reactor can support has led to the development of numerous capacities within Member States, many of which have been realized under the umbrella of the IAEA's technical cooperation programme. These benefits have been realized in a wide variety of areas within science and technology: nuclear power, radioisotope production, neutron beam research and analysis, nuclear medicine and personnel training, and more recently, materials development, component testing, computer code validation and environmental pollution control. One can cite numerous Member States, for example, Argentina, Brazil and Mexico in the Latin American region, which have developed high quality nuclear programmes indigenously, thanks in no small part to the growth and effective utilization of their research reactor programmes. Thanks to the recognition of these great benefits for human health, welfare and social development, new research reactors are being planned and built. Pre-eminent to the pursuit of these universal gains, however, is the precondition that any and all reactor operations be conducted with a commitment to and an assurance of safety: safety for the operators, safety for the public and safety for the environment.

The research reactor community has had a long and successful history of both productive and safe operations; however, nearly two thirds of the world's operating research reactors, i.e. 63% are now over 30 years old. Many of them have been refurbished to meet today's technological standards and safety requirements; however, there are challenges associated with ageing components and materials — and even members of staff — at these facilities. They continue to be serious issues, and are receiving increased attention, worldwide. Likewise, worldwide attention is focused on the serious erosion in governmental support, management commitment and available resources to the infrastructure necessary for effective research reactor operations. Robust utilization plans are not always an inherent part of the decision making process for determining whether a research reactor should be built, in the first place, or should continue to operate, in the long run. This is compounded by the fact that the use of these reactors is no longer an attractive research vehicle for university students and academic researchers. From these facts, it can be seen that there is a need to infuse vitality into this critical part of the nuclear industry and its infrastructure, thereby allowing for broader beneficial applications.

And that brings us to the challenges that stand before this conference. I am confident that each of you has come here in the hope of offering stimulating contributions and in the hope of helping to promote a renewed vision within the research reactor community. I know your discussions will be both insightful and probing. Hopefully, the findings and recommendations will carry on this vision of far and broad sightedness. We all know that if the safe operations, as a hallmark, are to be continued; if the scientific research and discoveries are to be continued; if the benefits for humankind are to be maintained, then the premises upon which research reactors are built must be reconsidered and brought into the technical, economic and social realities of today. The question is, "How is this to be done?" Here are some of my thoughts relating to the IAEA's activities.

First, in response to a resolution from the General Conference of the IAEA in 2000, the IAEA is developing a Code of Conduct concerning the utilization and operation of research reactors, relying on the technical, legal and political expertise of many of the Member States. This Code of Conduct, as a non-binding international legal instrument, offers guidance to Member States for the development and harmonization of policies, laws and regulations, and includes recommendations for 'best practices' for safety management of research reactors. The final draft version of the Code is being distributed to all Member States for comment this week and is expected to be resubmitted for consideration by the IAEA Board of Governors in March 2004. It is expected that the Code will have the unequivocal support of all States with research reactors.

The acceptance of such a Code will carry forward the initiatives that have been pursued under the IAEA's programmes for Safety Standards and Safety Review Missions. The concept of providing internationally accepted standards for the safe construction, operation, shutdown and decommissioning of a research reactor has been in place for over a decade. Likewise, the IAEA has commissioned numerous safety and security missions, such as Integrated Safety Assessment of Research Reactors (INSARRs), International Regulatory Review Teams (IRRTs) and, more recently, International Physical Protection Advisory Services (IPPASs) to ensure that the totality of the research reactor infrastructure is effective and properly focused. These activities must continue in the future and must be supported by all concerned stakeholders, as a mutually enhancing learning processes.

My second point addresses the need to effectively deal with the unique issues associated with fuel management in research reactors. In today's world, converting existing reactors from highly enriched uranium (HEU) to low enriched uranium (LEU), and designing new reactors to burn LEU, are urgently needed. The IAEA supports the Reduced Enrichment for Research and Test Reactors (RERTR) programme that aims at reducing and eventually eliminating all commerce in HEU for research reactors. High density fuels that will support the operation of high flux reactors must be developed and qualified. Fuel conversion leads directly to the next element within fuel management, that being the need for ensuring the repatriation of research reactor spent fuel to the country of origin. The United States of America has an Acceptance Programme in place for fuel discharged of US origin, up to May 2006, which is supported by the IAEA. A Tripartite Initiative (involving the IAEA, the Russian Federation and the USA) is working on developing a similar programme for research reactors with Russian origin fuel. The research reactor community, at large, must also address this issue. If only from economical and human resource viewpoints, safe and secure operations cannot be ensured in the future if each individual end user must deal, by themselves, with the issue of long term storage. Creative solutions, such as regional initiatives, to problems with the back end of the fuel cycle, must be considered and undertaken if the balance between technological advance, safety and security is to be maintained. This particular issue has been a point of emphasis by the Director General, as noted by his statement at the General Conference of the IAEA in 2003, and by his recent article in *The Economist*.

My third point addresses the threat to public safety and security posed by some form of nuclear terrorism. In the wake of recent highly organized terrorist attacks, the international community has to come to recognize that new and stronger measures must be taken to protect against and prepare for a diverse range of terrorist scenarios. One of the key priorities must be the provision of adequate physical protection for all nuclear materials, radioactive materials and facilities. This includes the transportation challenges inherent in their use. The concern is no longer limited to a specific country or geographic region. All of the activities associated with research reactor design, operations and utilization must include security implications. All strategies must consider the implications of theft, sabotage or other malevolent activities.

My final but, nonetheless, most important point, addresses the need for us to look at all aspects of life cycle management as part of our consideration of how to better utilize these technological tools. Many research reactors that are in operation, or are being proposed for operation, today seem to have neither realistic utilization plans nor solid decommissioning plans. The IAEA can assist operating organizations and national authorities in developing realistic strategic plans, focusing on implementing and utilizing these reactors in a manner that is consistent with the facility's and host country's capabilities and objectives, and that will make them more viable in the next decade and more attractive to future generations. However, the research reactor community must take the initiative and generate creative solutions to this problem.

In addition, special consideration should be given to research reactor facilities that have become or are developing into regional 'centres of excellence' in order to maintain a critical mass to create adequate benefits. Likewise, we must also consider regionally focused solutions to issues, such as the development and maintenance of the necessary regulatory infrastructures, the production of radioactive sources and the long term disposal of spent fuel and radioactive waste. The IAEA can assist Member States in the decommissioning of reactors by providing guidance documents, training and expert advice. Whether it is a point of national status or institutional pride, some countries and reactor operators are reluctant to take on the challenges of decommissioning. As a minimum, preliminary decommissioning plans should be prepared for all research reactor facilities. Only then can options be assessed, informed decisions made and appropriate actions undertaken. This challenge must be dealt with.

Now is the time to pursue provocative and innovative solutions to problems. Creative solutions, such as the recent initiation of the Asian Nuclear

Safety Network and the Iberian-Latin American Nuclear Safety Network, as vehicles to address the problem of knowledge management for sharing expertise and experience, must be pursued. The vision of the research reactor community must be extended. Perhaps, the chartering of a new international group, similar to the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) in the power reactor community, that would set its horizon on the 2020 or 2030 time frame, and would look to see where and how research reactors could and should evolve, is appropriate.

This conference was designed to provide us with a forum for considering far ranging issues. In that regard, you have a very full programme ahead of you. I encourage you to be provocative and innovative and to think outside the box. I encourage you not to be satisfied with 'good enough' when 'excellence' is within your grasp. Thank you for your attention, and I look forward to your findings.

BLANK

RESEARCH REACTOR SAFETY

(Session 1)

Chairpersons

J.P. BOOGARD
Netherlands

J.R. GRANADA
Argentina

S. MOROZOV
Russian Federation

H. ABOU YEHIA
France

Co-Chairperson

T. HARGITAI
IAEA

BLANK

SAFETY FEATURES OF THE REPLACEMENT RESEARCH REACTOR

M. SUMMERFIELD
Australian Nuclear Science and Technology Organisation (ANSTO),
Lucas Heights Science and Technology Centre,
Lucas Heights, Australia
Email: msx@ansto.gov.au

V. GAREA
INVAP S.E.,
Bariloche, Argentina

Abstract

The paper presents a general description of the development and application of basic safety criteria and the implementation of specific safety features in the design of the 20 MW pool-type research reactor currently being built by INVAP for the Australian Nuclear Science and Technology Organisation (ANSTO). A summary of the results of the preliminary deterministic safety analysis and the probabilistic safety assessment prepared by INVAP on ANSTO's behalf are presented as part of demonstrating the robustness of the design to the wide range of postulated initiating events considered. The paper also briefly describes the licensing process with respect to the way in which the licensing and regulatory regime within Australia influenced the design of the replacement research reactor (RRR). In particular, the reasoning for safety design features that have been incorporated as a result of the specific requirements of ANSTO and the Australian regulator is described.

1. INTRODUCTION

The replacement research reactor (RRR) is a 20 MW pool-type research reactor currently being built by INVAP for the Australian Nuclear Science and Technology Organisation (ANSTO) at their site in Lucas Heights, Australia. A description of the overall design of the facility and the status of the project to construct the facility have been provided elsewhere [1]. This paper considers the safety aspects of the facility, including the following:

— The basic safety criteria and design safety requirements applicable to the design.
— The specific safety features of the design, particularly those that were incorporated as a result of the specific requirements of ANSTO and the Australian regulator.
— A summary of the deterministic and probabilistic safety assessments that have been prepared by INVAP on ANSTO's behalf.
— A description of the licensing of RRR, including the way in which the licensing process is integrated into the overall project programme.

2. BASIC SAFETY DESIGN CRITERIA

The basic safety criteria applicable to the facility are discussed in the following sections.

2.1. Defence in depth

The 'defence in depth' principle has been applied in the design of the facility through the use of multiple barriers and levels of protection to prevent the unintentional release of radioactive material. Each level provides adequate protection in the event that the previous one fails. The five successive levels of defence in depth are:

— Prevention of deviations from normal operations and of failures.
— The detection and interception of such deviations and failures so as to prevent anticipated operational occurrences from escalating into accident conditions.
— Controlling the consequences of any resultant accident conditions in the unlikely event of such an escalation occurring.
— Controlling severe conditions, including the prevention of further escalation and the mitigation of the consequences of severe accidents.
— Mitigation of the radiological consequences of significant releases of radioactive materials.

In accordance with internationally accepted practice and the regulatory assessment principles (RAPs) of the Australian regulator, the Australian Radiation Protection and Nuclear Safety Agency (ARPANSA), the design emphasis is on the first two levels of defence in depth to carry the burden for nuclear safety. In particular, significant effort has been expended in minimizing the potential for initiating events to actually occur.

An example of this is the design of the control rods control logic, which incorporates a hard-wired interlock that prevents more than one control rod being withdrawn at any one time. In addition, the stepping motors that actually move the control rods incorporate a hard-wired filter that prevents them moving out of the core faster than intended. Both these features contribute to minimizing the potential for reactivity insertion accidents to occur.

2.2. Conservative design

As far as is possible, the technology used in the design, construction, commissioning and operation of the facility is based on conservative engineering practices that have been proven by previous experience. This is supported by the application of appropriate national and international codes and standards together with other relevant documented sources of information, such as IAEA guidance.

The use of novel design features and new technologies has been minimized as far as possible. Where such features have been incorporated into the design of the facility, appropriate research, development and testing, including the construction of prototypes, is being performed to validate the design feature. An example of this includes the construction of a full scale mock-up of the Second Shutdown System to experimentally determine the rate of draining of heavy water from within the reflector vessel that surrounds the reactor core. The results of the testing of this mock-up were utilized in the safety analysis of the facility, providing a basis for the assumed performance of the Second Shutdown System.

2.3. Inherent safety characteristics

Wherever practicable, inherent safety characteristics have been incorporated into the design of systems important to safety, particularly those that fulfil the three basic safety functions of reactor shutdown, decay heat removal and the containment of radioactive materials. An example of this is ensuring that all the reactivity coefficients associated with the reactor are negative. This is typical practice for modern nuclear reactors (both research and power) and is also consistent with the requirements of the Australian regulator, ARPANSA.

2.4. Passive and simple safety systems

Wherever practicable, passive safety features have been incorporated into the design of systems important to safety, particularly those that fulfil the three basic safety functions of reactor shutdown, decay heat removal and the containment of radioactive materials. In addition, the design has been kept as simple as possible, subject to the operational and availability requirements imposed on the facility.

Again, an obvious example is the water in reactor pool itself, which provides a passive heat sink for the decay heat following a reactor shutdown in conjunction with the natural circulation flow through the core initiated by the passive flap valves. This is a very simple design in that the only movement required is the opening of at least one of the four flap valves. This does not require any outside power, being reliant simply on the absence of discharge head from the Primary Cooling System (PCS) pumps (i.e. when the pumps stop and forced circulation is lost, the flap valves open as a result in the change in the pressure differential across them).

2.5. The ALARA principle

The 'as low as reasonably achievable' (ALARA) principle has been taken into consideration in all aspects of the design of the facility with respect to minimizing the radiation dose to operating personnel and members of the public that may arise as a result of normal operation and accident conditions. One feature of the application of the ALARA principle is the layout of the reactor building that utilizes zoning and access routing so as to separate, as far as practicable, users areas from operational areas. For example, the main reactor process systems (such as the PCS pumps and heat exchangers) are located within the reactor containment boundary in the basement area, completely separate to the areas accessed by research scientists and those involved in the production of radioisotopes.

Another example is the provision of a hot water layer at the top of the reactor pool. This layer reduces the radiation levels above the pool surface, thus reducing the dose to operating personnel ALARA, by preventing radioactive materials suspended in the reactor pool water actually reaching the surface of the pool.

3. SPECIFIC SAFETY FEATURES

A number of specific safety features have been incorporated into the design of the facility, the four most significant of which are identified as the First Reactor Protection System, the First Shutdown System, the Containment Energy Removal System and the Second Reactor Protection System.

3.1. Reactor protection and shutdown systems

The facility incorporates two independent and diverse reactor protection systems, each of which actuates independent and diverse shutdown systems.

The First Reactor Protection System (FRPS) is a software based protection system that generates a reactor shutdown through the First Shutdown System (FSS), which inserts the control rods into the core. The FSS acts by interrupting the electric power supplies to electromagnets that connect the control rods to their associated drive motors when a reactor shutdown signal is generated by the FRPS. It also incorporates a compressed air system that reinforces the insertion of the control rods, ensuring insertion under all design basis circumstances.

The FRPS also incorporates a number of protection interlocks that inhibit operator actions in certain circumstances. For example, reactor startup is not permitted unless both reactor shutdown systems are available. The FRPS also initiates reactor containment isolation in the highly unlikely event of a release of radioactive material within the containment and controls the associated Containment Energy Removal System (CERS).

The Second Reactor Protection System (SRPS) is a hard-wired protection system that generates a reactor shutdown through the Second Shutdown System (SSS), which partially drains the heavy water from the reflector vessel that surrounds the reactor core. A single dump line is routed from the reflector vessel to the heavy water storage tank located in the reactor building sub-basement. Six fail-open ball valves located in parallel open when a reactor shutdown signal is generated by the SRPS and allow the heavy water to drain under gravity from the reflector vessel.

3.2. Core cooling and decay heat removal

As indicated previously, core cooling and decay heat removal following reactor shutdown is achieved by natural circulation through the core that is initiated by the opening of flap valves on the PCS. The water within the reactor pool is used as the heat sink and no operator intervention is required for up to 10 days in the case of loss of the ultimate heat sink.

3.3. Reactor containment

The facility incorporates a reactor containment that forms a barrier to the release of radioactive material to the environment in the highly unlikely event of a reactor accident involving fuel damage. In the highly unlikely event of a release of radioactive material within the reactor containment, the FRPS initiates containment isolation. As indicated previously, a Containment Energy Removal System (CERS) is provided. This system removes heat from within the containment, minimizing any pressure increase following isolation and thus minimizing leakage from the containment to the environment.

A Containment Pressure Relief and Filtered Venting System (CPRFVS) ensures that the containment's structural integrity is maintained in the event of the failure of the CERS and over-pressurization of the reactor containment. It also provides a means of manually controlled filtered venting of the reactor containment as an accident management measure in the highly unlikely event of a beyond design basis severe reactor accident. This system discharges through the ventilation stack, which is approximately 47 m high.

A Containment Vacuum Relief System (CVRS) is also provided. Similar to the CPRFVS, the CVRS ensures that the containment's structural integrity is maintained in the event of the under-pressurization of the reactor containment. Such an event could arise as a result of changes in the external atmospheric pressure simultaneous with the containment being isolated.

3.4. Protection against external events

The facility is designed to take into account external events, the most significant of which are seismic events and aircraft impacts. Designing for seismic events is standard practice for all nuclear reactors and the approach adopted for the facility is generally the same as that adopted elsewhere in the world.

In response to specific commitments made by ANSTO, the facility has been specifically designed to withstand the effects of a light aircraft crash. The design solution adopted is the provision of two layers of protection by an aircraft impact grillage external to the reactor building and the reactor building structure itself. This grillage works in conjunction with the reactor building structure to ensure that in the unlikely event of a light aircraft crash, the reactor can be shutdown and maintained in a safe condition. In practice, there is no actual penetration of the reactor building and no scabbing on the internal surfaces as a result of a light aircraft crash. As such, there is considerable margin available in the design to allow for impacts larger than a light aircraft.

4. DETERMINISTIC SAFETY ASSESSMENT: METHODOLOGY AND RESULTS

Following the guidance of the IAEA [2], a set of postulated initiating events (PIEs) were assembled for assessment against the design. The set covered all aspects of the design, operation and utilization of research reactors, and was obtained by means of a systematic comparison of the IAEA generic list with the design of the facility, together with the application of engineering judgement and a review of the specific design features of the facility. The assessment of the specific design resulted in the identification of other initiating events, such as those involving the Cold Neutron Source. In all, 13 accident groupings were identified (see listing below). Each PIE was then assessed to determine whether or not it fell within the design basis. Exclusion from the design basis was on the grounds of inapplicability to the design, or eliminated by inherent design provisions, or sufficiently unlikely to occur as to allow it to be considered beyond the design basis. All PIEs remaining after this sorting were considered to lie within the design basis and designated design basis initiating events (DBIEs). This set constituted all credible accidents that were considered to have the potential to challenge plant safety, as follows:

— Loss of normal electric power;
— Insertion of excess reactivity;
— Loss of flow;
— Loss of heat sink;
— Loss of coolant from the primary cooling system;
— Loss of coolant from the reactor and service pools cooling system;
— Loss of heavy water;
— Erroneous handling or failure of equipment or components;
— Special internal events;
— Reactor utilization malfunctions;
— Spurious triggering of safety systems;
— External events;
— Human errors.

The DBIEs were then assessed, both within and across groups, to identify those whose consequences bounded other DBIEs. The identification of bounding DBIEs reduced the amount of necessary analysis while still allowing demonstration of a robust design of plant.

Having identified the bounding DBIEs, representative transients were defined for detailed analysis. The response of the reactor to the transients was analysed and evaluated to demonstrate that the design met the safety objectives and was acceptable to ASNTO and the regulatory body.

The transients were analysed by calculating the evolution of the main reactor parameters using the PARET-PC [3] and RETRAN02 [4] codes. Conservative modelling assumptions were made regarding the response of the reactor and the actuation of the safety systems. These included the neglecting of the negative reactivity inserted by the FSS for those DBIEs in which it was considered to fail ('failure' being defined as two or more control plates not reaching the bottom within the necessary time). In addition, no credit was taken for the actuation of any control systems to bring the plant into a safe shutdown state.

The design philosophy was that no significant damage to either the reactor core or the radioisotope production targets should occur for any DBIE. The intent was that such damage should be restricted to accidents having a very low likelihood of occurrence. Minor damage was tolerable, such as might arise from mechanical damage to a fuel assembly, on the basis that the consequences would be minor. The aim of the analysis was to show that the core and rigs were brought to a safe shutdown state, cooled by natural circulation. As part of the demonstration of this, all safety systems were assumed to function at their minimum design values.

Although highly unlikely, failure of one safety system, the FSS, was considered to lie within the design basis for frequent DBIEs. Therefore, for some transients, failure of the FSS with operation of the SSS was analysed. For those DBIEs deemed infrequent, failure of the FSS was considered to render the combined event sequence beyond the design basis and that particular event sequence was not considered further. Of particular interest in this instance was the potential for failure of both shutdown systems. This was considered to be so unlikely as to render it beyond the design basis and thus the failure to shutdown the reactor was considered not credible.

Other results obtained from the deterministic analysis showed that:

- The core is capable of being cooled by natural circulation without operator intervention for over 10 days before requiring coolant make-up.
- The reactor is capable of coping with a full break in the 350 mm PCS line.
- The passive siphon effect breakers ensure the cessation of any pipe work leak.
- The reactor can cope with severe losses of flow in the PCS and Reactor and Service Pool Cooling System (RSPCS) with no damage to either the fuel or the rigs.
- Reactivity insertion transients are moderate.
- Failure of the Cold Neutron Source (CNS) does not challenge the safety of the core.
- The reactor can cope with the continuous extraction of a control plate during reactor startup with failure of the FSS.
- Losses of heat sink lead to benign transients with no adverse impact on the core or rigs.
- Resistance to human-induced errors.

5. PROBABILISTIC SAFETY ASSESSMENT: METHODOLOGY AND RESULTS

In parallel to, but independent from, the deterministic analysis, a probabilistic assessment of the design was undertaken by INVAP. The purpose of the probabilistic analysis was to assess the residual risk posed by the facility and demonstrate that it met the dose-frequency criteria stipulated by the regulator for new nuclear facilities. It also had as objectives the identification of systems, components and human actions important to the overall risk as well as the estimation of the impact of dependent failures on the overall risk.

The scope of the probabilistic analysis was essentially that of a Level 1 PSA with certain Level 3 considerations. Accident sequence models were developed to the point of determining whether or not damage was expected to the structures containing radioactive material. The sequence frequency was then estimated. A number of fault sequences were then selected for consequence assessment based on their having a significant contribution to the overall frequency of core/rig damage. The consequences of these fault sequences were analysed conservatively in order to lead to a bounding determination of risk for the installation.

The identification and selection of initiating events was by using source and event analysis (SEA), a method similar to the Master Logic Diagram approach. The sources of radiation in the plant were identified together with the barriers that separate them from personnel and members of the public. The identification of the primary failure mechanisms of these barriers allowed the definition of the initiating events with the potential to bring about the failures.

The SEA method can generate a long list of failure mechanism and initiating events. To manage this list, a screening process is performed at each step in order to eliminate those events considered to make a negligible contribution to the overall risk. Eight categories of initiator were defined for detailed analysis.

Once the initiating events were established, an interference matrix was developed whereby each initiator, and the possible accident sequences derived from it, were mapped against the safety systems and functions that may be required along those sequences. The matrix essentially showed which systems and functions were relevant to each initiator and therefore needed to be modelled in event and fault trees. The safety systems modelled consisted of the following:

— First Reactor Protection System;
— First Shutdown System;
— Second Reactor Protection System;

— Second Shutdown System;
— Flap valves and siphon effect breakers;
— Emergency Make-up Water System;
— Standby power supply.

For each of the event tree headings identified in the interference matrix, a success criterion was established, allowing a fault tree to be developed. These fault trees were simplified on the assumption that the active components make a larger contribution to system failure than the passive components. In generating the event trees, a screening process was undertaken to eliminate those sequences that had no physical meaning or that were considered irrelevant.

The SAPHIRE code [5] was used to quantify the likelihood of the release of radioactive material. The IAEA database on component failure was used to obtain the necessary component failure rates.

At the time that this probabilistic study was carried out, detailed engineering of the plant had not been completed. This, together with the lack of plant-specific operating data, made assessment of dependent failure preliminary. Following a review of the causes of dependent failures, three generic types of common cause effects were identified and modelled as human errors. These were design errors, maintenance errors and test, calibration and inspection errors. These were then identified as human errors in the fault trees. The results of the probabilistic assessment gave a predicted frequency of severe core damage of 3.7×10^{-7}/a.

As a consequence of the very low frequency of severe core damage, those accidents with a credible potential for radioactive release to the environment do not involve significant damage to the core but instead involve damage to the irradiation rigs or minor core damage. Again, as the detailed engineering of the plant had not been completed, a conservative assessment of the consequences of the sequences needed to be made in determining overall operational risk. The accident release categories, their estimated frequencies of occurrence and their consequences to members of the public at the edge of the buffer zone are shown in Table 1.

6. LICENSING OF THE REPLACEMENT RESEARCH REACTOR

This section presents a summary of the licensing of the replacement research reactor (RRR) with particular emphasis on the way in which the licensing process is integrated into the overall project programme and how the licensing and regulatory regime within Australia influenced the design of the RRR.

TABLE 1. ACCIDENT RELEASE CATEGORIES, ESTIMATED FREQUENCIES AND CONSEQUENCES

Release category RC-B5	Local blockage of two flow channels
Mean annual frequency	1.3×10^{-5}/a
Maximum individual effective dose	0.0077 mSv
Release category RC-E3	Fall and subsequent break up of a fuel element in transit under water
Mean annual frequency	3×10^{-3}/a
Maximum individual effective dose	0.00053 mSv
Release category RC-G1	Melting during irradiation of one U-Mo irradiation rig containing 3 targets
Mean annual frequency	2×10^{-5}/a
Maximum individual effective dose	0.0056 mSv
Release category RC-G2-FRPS	Melting during irradiation of all twelve U-Mo irradiation rigs containing, in total, 36 targets
Mean annual frequency	2.8×10^{-6}/a
Maximum individual effective dose	0.0677 mSv
Release category RC-G3-RMI	Melting in air of one U-Mo irradiation rig containing 3 targets
Mean annual frequency	6.9×10^{-5}/a
Maximum individual effective dose	0.0127 mSv

6.1. The site licence

The application for the facility licence, site authorization was submitted to ARPANSA in April 1999. This licence was required in accordance with the ARPANS Act in order to demonstrate the suitability of the proposed site for the construction of the RRR.

One of the principal documents supporting the site licence application was the environmental impact statement (EIS). The EIS had been prepared previously in accordance with the Environmental Protection (Impact of Proposals) Act 1974 by specialist consultants as part of the environmental impact assessment, during which it was subject both to review by Environment Australia (EA) and to public review. EA prepared an environmental assessment report for the Minister of the Environment that also addressed the public submissions. The Minister for the Environment subsequently made recommendations to the Minister for Industry, Science and Tourism (the Commonwealth Minister responsible for ANSTO). The Government of Australia then considered the proposal and made the decision to proceed with the project.

The other principal documents supporting the site licence application were the siting safety assessment—site characteristics and site related design bases and the reference accident submission. These documents were prepared in accordance with ARPANSA draft criteria for the siting of controlled facilities.

The assessment of the suitability of a site for a proposed controlled facility involves the determination of the consequences of a hypothetical accident at the facility, called the reference accident, and comparing those consequences to the siting criteria. In general terms, the reference accident is a beyond design basis initiating event that also assumes degraded performance of one or more safety systems which leads to a release of radioactive material to the environment together with assumptions about prevailing meteorological conditions. The radiological consequences of the accident would be very unlikely to be exceeded by any actual accident, and are determined using conservative assumptions similar to those that would be used for design basis accidents. For site approval, these consequences must meet the siting criteria defined in the ARPANSA draft criteria.

The reference accident proposed by ANSTO and agreed by ARPANSA was based on a generic reference design for a pool type research reactor. It made a number of conservative assumptions about the actual design of the RRR. For example, the retention of fission products and other radionuclides in the reactor pool water was based on an assumption of a pool water volume of 80 m^3. In practice, the actual reactor pool water volume in the RRR is approximately 200 m^3.

It should be emphasized that the ARPANSA review of the site licence application was a separate but complementary review process that was performed subsequent to the environmental assessment process.

6.2. Specification of safety objectives and design safety requirements

During the first six months of 1999, ANSTO devoted significant resources to the preparation of a request for tender (RFT) that specified in detail ANSTO's requirements for the RRR. Integral within the team that prepared this RFT were a number of safety and licensing experts whose functions were to:

— Prepare a section of the RFT covering the overall safety objectives and the design safety requirements that the tenders must meet.
— Prepare a section of the RFT covering the content and format of a safety statement to be submitted by tenders.

Review of other sections of the RFT to ensure that safety issues were adequately and appropriately addressed in a consistent manner throughout all sections of the RFT.

The principal sources used in the identification and development of the safety objectives and the safety design requirements were various IAEA codes, standards and guidelines and the ARPANSA regulatory assessment principles (RAPs) and their associated regulatory guides (RGs). Other sources included the recommendations identified by the Minister for the Environment as part of the environmental assessment process and licence conditions imposed by the CEO of ARPANSA when granting the facility licence, site authorization.

The intent of the RFT was to translate all the requirements and guidelines contained in these sources into a single, comprehensive set of requirements that could be readily understood by the tenders without their having to try to interpret the IAEA guidance and the ARPANSA RAPs and RGs. This was particularly important since all four of the pre-qualified tenders were overseas organizations who, although experienced suppliers of research reactors, were unfamiliar with the Australian regulatory regime.

The RFT was released in July 1999 with the call for tenders from the pre-qualified tenders. Submissions were received by the end of 1999, including a safety statement that would form the basis of the preliminary safety analysis report (PSAR) to be prepared by the successful tender. The purpose of requiring the safety statement as part of the tender documentation was as follows:

— To confirm that there would be no significant problems associated with licensing the tender proposed design within the context of the Australian regulatory regime.
— To assess the ability of the tender to prepare a suitable safety case in accordance with the standard content and format of a Safety analysis report (SAR) as defined by the IAEA.

The same safety and licensing experts who were involved in the preparation of the RFT were also involved in the review of the tenders received as one of the tender evaluation working groups, with particular emphasis on reviewing the safety statements with respect to the two aspects above. The safety and licensing working group also reviewed other parts of the tender documentation to confirm consistency with the safety statement, as well as provided support to other working groups with respect to any safety or licensing issues identified.

As such, the safety and licensing of the RRR was considered as an integral part of the overall RRR project, with the same experts being involved in both the preparation of the RFT and the evaluation of the tenders' submissions received from the tenders. This also ensured that the evaluation of the various tenders' submissions was conducted in a consistent and even manner in accordance with the principles of 'due diligence'.

6.3. The construction licence

The application for the facility licence, construction authorization was submitted to ARPANSA in May 2001. This licence was required in accordance with the ARPANS Act in order to commence construction of the RRR.

The principal document in the construction licence application was the PSAR prepared by INVAP and based upon the safety statement prepared and submitted as part of INVAP's tender. The purpose of the PSAR is to demonstrate the safety of the reactor facility design. It also serves the following purposes:

— To aid the designer in confirming that individual systems are integrated correctly, since the reactor design and the development of the PSAR are complementary, interactive processes.
— To ensure that the safety analysis has properly identified the safety issues relevant to the design and that safety analysis and design are consistent.
— To aid in the appreciation of the relevant design criteria, their limitations and requirements, and in the evaluation of the hazards posed by the facility.
— To provide the basis for development of the draft final safety analysis report (FSAR) which will be submitted to ARPANSA in support of the application for the facility licence, operations authorization.
— As an aid in operator training and familiarization with the RRR during the construction and commissioning phase of the RRR project.

The PSAR was prepared by INVAP in accordance with the guidelines of the IAEA Safety Guide SS 35-G1 [2]. It demonstrates that the RRR design complies with the requirements of the ARPANSA RAPs for controlled facilities and the associated RGs. At the time of the submission of the PSAR with the construction licence application, the preliminary engineering stage of the project had been completed and all the significant design decisions had been made.

The construction licence application as a whole, and the PSAR in particular, were subjected to extensive review by ARPANSA and 1159 formal reactive review comments were received that required formal responses. There were also a significant number of areas where more detailed discussions were required and additional analysis and clarification had to be provided. Examples of these include the detailed thermohydraulic calculations upon which the safety analysis presented in the PSAR was based, as well as additional analyses of some beyond design basis events not considered in the PSAR. At the end of this review, a regulatory branch assessment report (RAR) was prepared that contained a summary of the ARPANSA review. This identified over 100 recommendations and was presented to the CEO of ARPANSA for his consideration when determining whether to grant a construction licence.

The ARPANSA Nuclear Safety Committee also performed a review of the application in relation to the following three key safety issues:

— Seismicity;
— Spent fuel and radioactive waste;
— Safety analysis.

This committee is an independent body that advises the CEO of ARPANSA. It is made up of experts and representatives of the community. It not only reviewed the application but also sought additional information from both ARPANSA regulatory branch (the part of ARPANSA with the lead responsibility for reviewing and assessing the safety of controlled facilities) and from ANSTO. The Committee prepared and presented its own report, containing various comments and recommendations, to the CEO of ARPANSA for his consideration when determining whether to grant a construction licence.

The PSAR was also subjected to an independent peer review by a team of six experts nominated by the IAEA. They considered 22 key safety issues and their final report identified 25 recommendations and 18 comments. This report, including its recommendations and comments, was also presented to the CEO of ARPANSA for his consideration when determining whether to grant a construction licence. In addition, the PSAR was reviewed by the Autoridad Regulatoria Nuclear (ARN), the Argentine regulator, who provided a report of their review to ARPANSA.

The PSAR was also subjected to a public review with submissions being made to ARPANSA. To facilitate this, copies of the complete application, including the PSAR, were made available to interested parties. These included Sutherland Shire Council (the local council covering the area in which the RRR would be located) and to various libraries for public viewing. In addition, a summary version of the PSAR was prepared and made available on the Internet and in hard copy. The ARPANSA reactive review comments and the ANSTO response to them, together with the final report of the peer review by the group of IAEA nominated experts, were also made available to the public before the end of the public review period. This enabled the public not only to present their own submissions but also to consider the issues raised by the 'experts'. Approximately 1150 submissions were made from which 177 issues were identified, virtually all of which had been previously identified in public submissions on the EIS during the environmental assessment process. A public consultation report was prepared, summarizing the public submissions. This report was also presented to the CEO of ARPANSA for his consideration when determining whether to grant a construction licence.

Finally, in December 2001, the CEO of ARPANSA chaired a public forum in Sydney, Australia, to receive oral submissions from interested parties. He was assisted in this by a three-member panel of experts, each of whom prepared and presented a report of the public forum to the CEO of ARPANSA.

On the basis of the evidence provided in the various submissions and reports identified above and in accordance with the requirements of the ARPANS Act and associated Regulations, the CEO of ARPANSA granted the facility licence, construction authorization in April 2002. This licence incorporated 18 licence conditions and contained a number of specific references to ANSTO demonstrating that recommendations identified in the RAR have been taken into consideration to the satisfaction of the CEO of ARPANSA.

6.4. The ongoing interface with ARPANSA

The RRR project is currently in the construction stage. The principal safety and licensing activities during this stage are as follows:

— The review of detail engineering (DE) design deliverables (principally design documentation) prepared by INVAP and their sub-contractors to ensure compliance with the safety and licensing requirements and the safety case as presented in the PSAR. Note that all DE design deliverables are subject to an ANSTO review, verification and acceptance process in accordance with an appropriate project procedure and that the safety and licensing review is just part of this.

— Ensuring compliance with the licence conditions imposed by the CEO of ARPANSA when granting the facility licence, construction authorization together with the conditions inherent in the ARPANS Act and associated Regulations and demonstrating such compliance to ARPANSA.

— Obtaining the CEO of ARPANSA approval to construct items important to safety in accordance with ARPANS Regulation 54 and Licence Condition 4.6. Due to the number of items that require such approval and the need to demonstrate compliance with Licence Condition 4.6, a project procedure has been developed and the submission process formalized. In addition, ARPANSA have set up an Assessment Committee to which ANSTO reports on a weekly basis to facilitate their review of these submissions.
— Obtaining the CEO of ARPANSA approval for changes that will have significant implications for safety in accordance with ARPANS Regulation 51 and Licence Condition 4.11. Note that at this time, there have been numerous changes as a result of the development of the detailed engineering of systems and components. A number of changes have been identified that constitute a change that will have significant implications for the safety of the RRR, although none is considered to be detrimental.
— Managing the interface with ARPANSA in relation to the ongoing procurement, manufacture, construction and installation of the RRR, with particular emphasis on items important to safety as identified in the PSAR. This principally involves ensuring that ARPANSA is aware of the ongoing status of the project and facilitating their inspections and audits.
— Updating of the PSAR in accordance with Licence Condition 4.8 and developing it into the draft FSAR. This also includes the revision of various nucleonics, thermohydraulic and transient analyses in line with the development of the detailed engineering of the RRR, as well as the validation of the computational modelling used for such analyses in accordance with Licence Condition 4.10.

Ensuring compliance with the conditions imposed by the Minister for the Environment, arising from the environmental impact assessment and demonstrating such compliance is also an ongoing licensing activity. However, this is being done as part of ANSTO's site-wide environment management plan.

From the point of view of the project, it is the third of these items that involves a significant amount of effort. This is because it has meant that every Safety Category 1 and 2 structure, system and component (as defined in the PSAR) requires a separate regulatory approval prior to procurement and/or manufacture, construction, and installation. The need to integrate this Reg54 approval process with both the schedule for the review, verification and acceptance of DE design deliverables, and with the construction schedule for a whole new facility has resulted in a complex process requiring careful control. In particular, it has become necessary to make multiple submissions for approval for some systems in order to comply with ARPANSA Regulation 54, the availability of accepted DE design deliverables and the construction schedule. To date, approximately 90 submissions have been made to ARPANSA for Reg54 approval and it is anticipated that the final number of such submissions will be between 110 and 120.

For example, the PCS was split across three submissions, one covering the decay tank, one covering the other main components with a significant manufacturing lead time (i.e. the main PCS pumps and heat exchangers), and one covering the remainder of the system. The submissions were done this way because the decay tank needs to be installed very early in the construction schedule, prior to the pouring of the heavy concrete for the reactor block and the manufacture of the decay tanks involves a significant lead time. As such approval to manufacture the PCS decay tank was required before all the DE design documentation for the remainder of the PCS completed the ANSTO review, verification and acceptance.

The setting up by ARPANSA of the Reg54 Assessment Committee to facilitate the review process and provide a working level forum planning and progressing these submissions has been of considerable benefit.

6.5. Future licence applications

The application for the facility licence, operating authorization will be submitted to ARPANSA in anticipation of the authorization to operate the RRR. As identified previously, the principal document in the operating licence application will be the FSAR that will be prepared jointly by ANSTO and INVAP. It is anticipated that this application will be submitted to ARPANSA mid-2004, since an operating licence will be required to enable fuel to be loaded and hot commissioning (Stage B and Stage C commissioning) to be performed.

The purpose of both versions of the FSAR is to demonstrate the safety of the ‘as-built’ reactor facility, with the difference between these versions being the extent of the commissioning results available. The FSAR will also incorporate details of the operating regime for the RRR, including the operational limits and conditions (OLCs). This approach is consistent with the ARPANSA RAPs and internationally accepted practice for nuclear facilities.

7. CONCLUSIONS

ANSTO’s replacement research reactor (RRR) has been designed to comply with a number of basic safety criteria that were derived from best international practice and the specific requirements of the Australian regulator, ARPANSA. It incorporates a number of specific safety features that ensure compliance with these basic safety criteria.

Both a deterministic and a probabilistic safety assessment have been performed for the facility and presented in the PSAR. Both these assessments demonstrate that the design complies with all statutory limits and objectives.

The PSAR was submitted to the Australian regulator, ARPANSA, and after a thorough review process, including an international peer review, a public review and forum, and various external consultants, an authorization to construct the facility was granted in May 2002.

Safety and licensing aspects have been fully integrated into the project management structure and during all phases of the project.

REFERENCES

[1] CAMERON, R.F., HORLOCK, K., ROBINSON, R., The Australian Replacement Research Reactor, PBNC13 (October 2002).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report, Safety Series No. 35-G1, IAEA, Vienna (1994).

[3] OBENCHAIN, C.F., PARET—A Program for the Analysis of Reactor Transients, IDO-17282 (1969).

[4] RETRAN 02, A program for transient thermal-hydraulic analysis of complex fluid flow systems, Volume 1: Theory and Numeric (Revision 1), Chapter 11.

[5] IDAHO NATIONAL ENGINEERING LABORATORY, “Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) Version 6.54”, Lockheed Martin Idaho Technologies Company Inc., USA.

BLANK

A RISK PERSPECTIVE FOR THE REPLACEMENT RESEARCH REACTOR IN AUSTRALIA

J. BARÓN, J. NÚÑEZ MC LEOD, S. RIVERA
CEDIAC Institute, Facultad de Ingeniería,
Universidad Nacional de Cuyo,
Mendoza, Argentina
Email: jbaron@uncu.edu.ar

Abstract

In July 2000, the Australian Nuclear Science and Technology Organisation (ANSTO) signed a contract with the company INVAP S.E. of Argentina for the design, construction and commissioning of a replacement research reactor (RRR). INVAP contracted CEDIAC to prepare the probabilistic safety assessment (PSA) for the RRR in support of the ANSTO application for the construction licence. The PSA is complementary to the safety analysis, in the sense that it asks questions such as "What if the postulated initiating events were to occur and more than one piece of equipment were to fail? What if several things were to go wrong?" The PSA attempts to determine all the possible combinations of how the plant could respond to an initiating event, group all the possible outcomes, obtain conservative estimates of the frequency, and bounding estimates of the consequences (i.e. doses to the worst exposed individual of the public). The frequency and consequence constitute the risk, and when evaluated for all possible events, can be compared against the safety objectives set out in the regulatory principles. Besides the basic objective of the PSA, which is the quantitative evaluation of the risks associated with the RRR, and its comparison to the regulatory objectives, the PSA studies have been performed in parallel with the basic engineering phase of the project. Therefore, preliminary results from the 'risk point of view' were used as input to the design process, thus permitting improvements to be made to the design, and resulting in an effective reduction of the residual risk. To perform the PSA studies, several methodological developments were made in order to obtain a representative list of internal and external initiating events, to treat component and human related failures, to consider common-cause failures, and to consider some specific aspects of the design (i.e. fail safe components, passive systems, and lack of need for support systems). The PSA studies were performed to obtain not only quantitative estimations of the risk, but also quantitative estimations of the uncertainty associated with them. The overall results of the PSA indicate a very low residual risk for the RRR, and provide a valuable tool to analyse detailed engineering alternatives. From the regulatory point of view, the safety objectives have been fulfilled. Several design characteristics of the RRR contribute to the very low risk estimations (e.g. two completely independent shutdown systems, the lack of need for support systems for the safety functions, the absence of in-core experiments, the redundancy on the cooling modes, etc.). The PSA proved to be a valuable tool to increase the safety level of the RRR, and this was possible because the interaction between the PSA performers and the designers was straightforward.

1. INTRODUCTION

The proposed replacement research reactor (RRR) is an open pool-type reactor, that is, a reactor where the core sits in a deep pool of water which provides cooling of the core, and protection against the effects of radiation. The metallic pool is inserted in a high integrity reinforced concrete block. The RRR will provide facilities for irradiating targets for the production of radiopharmaceuticals and silicon, and for experiments, as well as high quality neutron beams for specialized research.

2. OBJECTIVES OF THE PSA

Probabilistic safety assessment (PSA) is a valuable tool for the quantification of risks arising from the operation of nuclear reactors and other complex installations. By means of this risk quantification, the whole plant is analysed from the safety point of view, and the features that govern the risk in the plant can be identified and ranked by importance. Furthermore, individual safety issues can be analysed and their impact on the risk estimated. This procedure involves not only the existing issues, but the PSA can also be used as a tool to estimate the expected risk reduction benefits from proposed changes to plant design, operating and maintenance practices. In an overall sense, the PSA performed on a specific plant constitutes a realistic measure of its safety by quantitative means, which can follow the safety improvements of the plant in what is known as a living PSA.

The basic objective of the present PSA for the RRR is the quantitative evaluation of the risks associated with the operation of this reactor, according to the present PSAR and located in the proposed siting at Lucas Heights, in New South Wales, Australia.

As part of this basic objective, the following particular objectives were pursued:

(a) The identification of internal and external events that may lead to accident conditions;

(b) The identification and analysis of the plant systems responses to the initiating events identified to pose a relevant risk to the public and operators;
(c) The identification of systems, components and human actions—considered important to the overall risk;
(d) The estimation of the impact of dependent failures in the overall risk;
(e) The estimation of the containment response and associated source terms for a few representative accident sequences;
(f) The comparison of the representative accident sequences risks with the regulatory objectives.

Besides these objectives, the following two operative objectives were also pursued:

— The development of a living PSA that will allow for the inclusion of any minor design change to be produced after the detailed engineering is finished, and preparation of an updated PSA for the FSAR.
— The preparation of a comprehensive PSA document that will allow for the audit of the hypothesis, methods and assumptions included in it.

Moreover, since this PSA was developed in parallel with the basic engineering phase of the RRR, the preliminary results were used as input to the design process, thus permitting improvements to be made to the design.

The scope for the present PSA is a Level I PSA with certain Level III considerations. These considerations include the selection of a few accidents that are considered to be representative of the risk of the installation. These accidents were selected on the basis of having a significant contribution from the frequency point of view, understanding as significant contribution a frequency higher than the most stringent frequency set in the safety objective of the Australian Radiation Protection and Nuclear Safety Agency (ARPANSA) regulation. This means a yearly frequency higher than 10^{-6}. The comparison is made taking into account the 95% confidence level provided by the uncertainty analyses.

For these accidents, their release fractions were derived, based on conservative assumptions, and the containment response was analysed, in order to obtain representative source terms for the installation. The dose expected for these source terms on the public was also calculated, taking into account conservative weather conditions.

The corresponding dependent failure analyses helped to verify the segregation and redundancy of safety functions. Several lessons learned in these analyses were retrofitted to the designers.

The PSA includes consideration of all envisaged operation modes of the RRR, and all expected radiation sources that may exist in the plant. However, as the detailed engineering and the specific operating and maintenance manuals are not yet available, several conservative hypotheses were made. In a later revision of the present PSA, more realistic assumptions may be made. It is expected that there will be a corresponding reduction in the risk estimations from the present PSA. In this sense, the present results are considered to be a bounding estimation of the risk.

3. ANALYSES METHODS

3.1. Method for identification and selection of initiating events

The identification and selection of initiating events always poses a question about the completeness of the PSA, and up to now there is no method that can guarantee this completeness. At some level, every method requires engineering judgement.

For this purpose, a specific systematic method was developed, known as source and event analysis (SEA). This method is quite similar to the Master Logic Diagram (MLD) proposed in NUREG/CR-2300 [1] and has been derived from the approach indicated in IAEA-TECDOC-517 [2]. It was used in Ref. [3].

The SEA method is a multi-step, bottom-up approach that consists of:

— Identification of the relevant radiation sources in the plant (e.g. core, fuel in the spent fuel pool, fuel in the shipping cask, etc.);
— Identification of the barriers that separate the radiation sources from the public and/or plant personnel;
— Identification of the primary failure mechanisms of these barriers;
— Identification of the initiating events that may cause the identified failure mechanisms;
— Systematic selection and grouping in a set of representative initiating events.

The SEA method can generate a large list of failure mechanisms and initiating events (which is an advantage). To manage this list, a screening process is performed at each step, in order to eliminate those events that make a negligible contribution to the overall risk. For example, if a certain source of radioactive material (e.g. an ion-exchange resin) is assumed to pose a negligible risk due to its small inventory, it can be excluded in the first step. If a certain failure mechanism is known to occur very slowly (e.g. corrosion), and its status readily identified, it may also be excluded in the third step for the purposes of the PSA.

Specific considerations are used for the definition of the external events to be analysed. For this purpose, a screening of possible events is performed according to the IAEA guidelines [4]. After this screening process, the local expected external events at the Lucas Heights (Australia) site were analysed based on the experience of the HIFAR PSA [5].

A review of incidents in research reactors was prepared independently in chapter 16 of the PSAR and no additional initiating events were identified.

The list of initiating events identified for this PSA is indicated in Table 1.

3.2. PSA models and data

Many sophisticated tools for the quantification of risks arising from the operation of nuclear reactors and other complex installations have been developed to perform this analysis. SAPHIRE (INEL) is one such package and was used for this study.

Once the initiating events (IE) are established, the following procedure is applied:

— Development of an interference matrix, where each IE is analysed and the possible accident sequences derived from it are mapped against the safety systems and functions that may be required along those sequences. This matrix shows which systems and functions are relevant to each initiating event and, therefore, which are needed to be modelled in fault trees and included in event trees. It also helps in the definition of the success criteria for the event tree headers.
— Development of qualitative system fault trees. For each of the event tree headings identified in the interference matrices, a success criterion is established. Upon this success criterion and taking as a base the process and instrumentation (P&I) and the operational limits and conditions, where known, of each safety system, a fault tree (FT) is developed. These FTs are simplified, based on the criterion that the active components will have a larger contribution to the system failures than the passive components.

The systems analysed with the FTs technique are:

— First and second reactor protection systems (FRPS and SRPS);
— First shutdown system (FSS);

TABLE 1. LIST OF INITIATING EVENTS

Categories	ID	Description
A		Reactivity transients
	A1	Erroneous withdrawal of a control rod during startup
	A2	Erroneous withdrawal of a control rod during normal operation
B		Loss of flow
	B1	Core bypass
	B2	Loss of electric power
	B3	Primary pump failure
	B4	Primary isolation valve undesired closure
	B5	Fuel channel local blockage
C		Loss of coolant
	C1	Primary LOCA caused by a rupture upstream of the primary pump
	C2	Primary LOCA caused by a rupture downstream of the primary pump
	C3	Pool cooling system LOCA
D		Loss of heat sink
	D	Loss of heat sink
E		Mechanical damage to fuel assemblies
	E1	Fuel assembly mechanical damage in the irradiated fuel assemblies pool
	E2	Fuel assembly mechanical damage in the spent fuel storage racks in the reactor pool
	E3	Fuel assembly fall while in transit
F		Heavy water leak
	F	Heavy water spill outside reactor pool
S		Seismic events
	S	Seismic event

- Second shutdown system (SSS);
- Flap valves and siphon effect breakers, which includes five conceptual headings:
  - Siphon effect breakers (SEB);
  - Suction and Impulsion siphon effect breakers (S&I-SEB);
  - Flap valves at Level 6000 (FVL6000);

  - Flap valves at Level 7000 (FVL7000);
  - Flap valves at Level 6000 and 7000 (FV6&7);
- Emergency make-up water system;
- Emergency electrical power supply (EEPS).

(1) Development of qualitative event trees (ET). These trees are developed for each IE, with the corresponding headers obtained from the interference matrices. After the development of each ET, a screening process is taken in order to eliminate those sequences that have no physical meaning or that are considered irrelevant.

(2) When all the ETs and their corresponding headings have been qualitatively delineated, they are programmed in the SAPHIRE [6] code. These codified trees are then quantified at a component level. For the quantification process, the IAEA database on component failure [7, 8] was used. The results are then integrated in order to obtain overall quantification. SAPHIRE provides also tools to perform importance, sensitivity and confidence analyses on the results.

(3) The Level I PSA results are analysed in order to obtain key end state frequencies, such as the core damage frequency for the reactor. At this step, several 'importance measures' are estimated, according to the Fussel-Vessely importance estimation, and sensitivity studies are then performed, on the parameters identified with highest importance. The various importance measures give an indication of how significant each basic event is to the overall top event probability, or to an end state frequency.

(4) Each basic event in the PSA has some uncertainty associated with its probability. This uncertainty is represented by the associated confidence interval. Having obtained 'best estimates' of all of the end state frequencies, it is usually of interest to understand the uncertainty associated with these estimates. This is achieved by 'propagating' the uncertainties through the Level I PSA model. The objective is twofold: to understand the limitations of the numerical results, and to obtain upper bound estimates of the end state frequencies at specific levels of confidence.

(5) In order to estimate the plant behaviour beyond the scope of the Level I PSA, a few representative accidents are analysed from the phenomenological point of view. The overall PSA is screened to obtain a reduced number of accident sequences that represent risk-relevant contributors. The selection criteria are based on the potential for causing relevant doses to the public, and/or criteria based on the expected high frequency

of such an event or sequence of events. In this sense, a few accidents encompass the risk characteristics of the plant.

The selected accidents are quantified in their expected frequencies, according to the frequencies of the accident sequences that lead to them, and source terms are derived for each of them, in order to be used for dose estimation on the public. The results are then compared against acceptance criteria and its compliance is discussed.

3.3. Dependent failure analyses

Actual operating experience with complex plants demonstrate that, although the likelihood of a series of failures is quite small, it is numerically higher than would be estimated solely from a postulated chain of independent failures. This is because physical and human interactions result in dependent failures that increase the conditional probability of each successive failure in the chain.

The treatment of dependencies in the identification and quantification of accident sequences is called ‘dependent-failure analysis’. Dependencies tend to increase the frequency of multiple, concurrent failures. Since essentially all important accident sequences that can be postulated for nuclear reactor systems involve the hypothesized failure of multiple components, systems and containment barriers, dependent-failure analysis is an extremely important aspect of PSA.

A detailed engineering analysis of dependent failures must consider the root causes of component failures and the degree of dependence among component failures with regard to each root cause. This is particularly important for a new reactor, where no operating, maintenance and testing experience exists.

There are three types of dependent failures:

(1) Functional dependencies between systems. These are where the success of one system is dependent on the success of another, for example, due to reliance on a support system or where there is a shared component or subsystem in two systems. In the present PSA, functional dependencies are treated explicitly in the event trees.

(2) Dependencies or common causes between basic events. Traditionally, in PSAs of existing plants, common-cause dependencies (the second type of dependencies) are modelled in as a numeric fraction of the basic event failure probability that occur dependently (e.g. the beta factor, or multiple Greek letter methods). Parameters for these models make use of

generic data, where available, which are then updated with plant-specific data. For the present PSA, in lieu of the more conventional method, common-cause effects were modelled as follows. Components in redundancy sets were identified for inclusion in common-cause groups. Three main types of common-cause effects were modelled as human errors:

— Design errors;
— Maintenance errors;
— Test, calibration and inspection errors.

These errors were developed as human error fault trees.

(3) Dynamic human interactions. In the present PSA, the actuation of the safety systems is automatic, and no credit is taken for manual actions. Therefore, the third class of dependent failures, which corresponds to dynamic human interactions (e.g. inability to act due to operator error in response to) is not relevant because it does not contribute to the failure of safety systems. Where credible operator actions that could jeopardize safety system functions were identified, they were included as basic events in the fault tree models. Furthermore, conservative assumptions were made regarding plant operations (e.g. that the operator prematurely shuts down the primary cooling system pumps, following a trip).

4. PRELIMINARY PSA RESULTS

4.1. Core damage frequency

The core damage frequency (CDF) obtained by the summation over all the frequencies of internal event sequences that may lead to core damage is:

Mean CDF:	6.96×10^{-08}/a
5% percentile CDF:	5.72×10^{-09}/a
95% percentile CDF:	2.51×10^{-07}/a

These values, when compared with the Safety Limits and Objectives, indicate, with a 95% confidence, that the CDF fulfils the more stringent Safety Objective (10^{-06}/a). Therefore, it can be stated that those accidents with the potential to cause a significant damage to the core, pose a negligible risk to the public in the vicinity of LHSTC.

The relative contribution to the mean CDF due to each internal event is indicated in Fig. 1.

It can be seen that the loss of flow and loss of heat sink initiated transients contribute to the overall CDF with more than 99%.

From the consequence point of view, it is important to notice that these transients have the potential to cause core damage with the core covered with water. This means that, despite the overall negligible frequency numbers, those accidents that have the potential to cause a core meltdown without water (e.g. LOCAs), have a contribution to the overall CMF one hundred times smaller than the contribution of the water-covered meltdown accidents.

The seismic contribution to the core damage frequency has been analysed in two different scenarios.

The first scenario considers the contribution to the CDF for those seismic events whose frequency is up to that stated for the SL2 earthquake, that is, with a frequency higher or equal to 10^{-4}/a. This scenario is considered appropriate from the standpoint that the critical systems in the RRR have been designed to withstand this seism, according to IAEA recommendations.

The CDF obtained by the summation over all the frequencies of seismic event sequences (up to SL2 level) that may lead to core damage is:

Mean CDF:	1.91 × 10-10/a
5% percentile CDF:	2.89 × 10-11/a
95% percentile CDF:	7.96 × 10-10/a

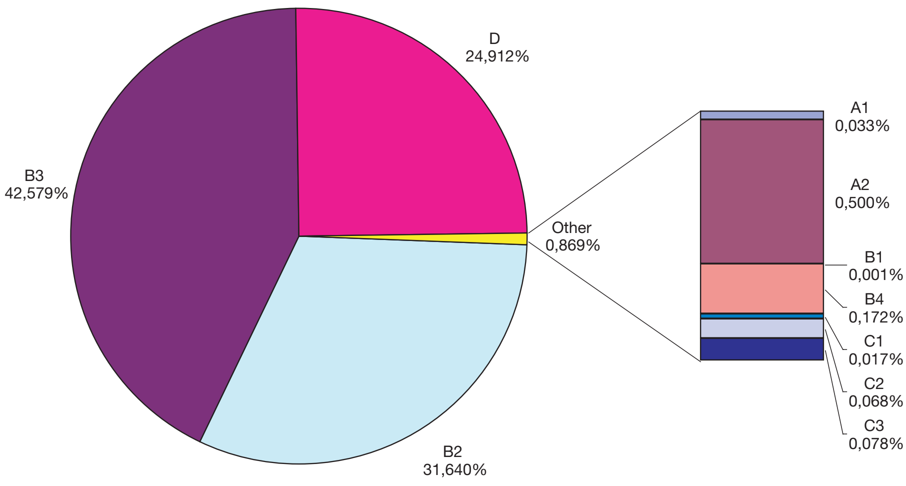


FIG. 1. Relative contribution of each initiating event to the CDF.

These values, when compared with the contribution of internally initiated events, contribute in less than 0.3% to the overall CDF. Therefore, it can be stated that those seismically initiated accidents with the potential to cause a significant damage to the core, up to the SL2 Level, pose a negligible risk to the public in the vicinity of LHSTC.

The second scenario considers the contribution to the CDF for all the seismic events indicated by the Hazard Curve for the site. This scenario is not considered credible, but in any case it was analysed in order to have an estimation of the seismic events beyond SL2. The CDF obtained by the summation over all the frequencies of seismic event sequences (for the full hazard curve) that may lead to core damage is:

Mean CDF:	$3.32 \times 10\text{-}08/a$
5% percentile CDF:	$7.87 \times 10\text{-}09/a$
95% percentile CDF:	$5.53 \times 10\text{-}07/a$

These values, when compared with the contribution of internally initiated events, contribute in 32% to the overall CDF. Therefore, it can be stated that those seismically initiated accidents with the potential to cause a significant damage to the core, for the whole hazard curve (e.g. beyond SL2 Level), pose a risk comparable to that posed by internal initiators to the public in the vicinity of LHSTC.

4.2. Level III PSA results

It is usual that the accident scenarios which contribute the most to the risk of a nuclear reactor are those that involve substantial damage to the reactor core. However, the very robust design of the RRR, with a high degree of redundancy and independence of its safety functions, makes these sequences to be of such a low probability that they are not considered credible.

These very low values fulfil the most stringent ARPANSA regulatory requirements for PSA frequencies. One of the reasons for this is that the PSA was developed at the same time the basic engineering was developed, and the weak points identified during the system's analyses for the PSA were retrofitted to the designers, who in turn improved those system designs.

As a result, for the RRR the risk representative scenarios do not involve significant core damage. These accident scenarios were analysed and quantified both in their frequencies and their associated potential doses on the public.

The release categories (RC) for the representative release events, even when modelled under extremely conservative assumptions, show that the expected risk contribution of the selected accidents is well below the acceptable levels.

It is important to notice also, that the maximum doses do not require any off-site emergency measures (e.g. sheltering).

REFERENCES

[1] NUCLEAR REGULATORY COMMISSION, PRA Procedures Guide, Rep. NUREG/CR-2300, Office of Standards Development, Washington, DC (1983), pp. 3–21.

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Application of Probabilistic Safety Assessment to Research Reactors, IAEA-TECDOC-517, IAEA, Vienna (1989).

[3] BARÓN, J., NÚÑEZ J., RIVERA, Y S., "A Level III PSA for the Inherently Safe CAREM-25 Nuclear Power Station", Probabilistic Safety Assessment and Management 96 (CACCIABUE, PAPAZOGLOU, Eds), Springer-Verlag, London (1996), pp. 553–58.

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Treatment of External Hazards in Probabilistic Safety Assessment for Nuclear Power Plants, Safety Series No. 50-P-7, IAEA, Vienna (1995).

[5] "A Level I + Probabilistic Safety Assessment of the High Flux Australian Reactor", Pickard, Lowe & Garrick (1998).

[6] IDAHO NATIONAL ENGINEERING LABORATORY, Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) Version 6.54, report, Lockheed Martin Idaho Technologies Company, Inc., USA.

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Generic Component Reliability Data for Research Reactor PSA, IAEA-TECDOC-930, IAEA, Vienna (1997).

[8] INTERNATIONAL ATOMIC ENERGY AGENCY, Component Reliability Data for Use in Probabilistic Safety Assessment, IAEA-TECDOC-478, IAEA, Vienna (1988).

ASSESSMENT OF THE SAFETY OPTIONS FOR THE JULES HOROWITZ REACTOR (RJH)

D. RIVE, L. BERTRAND, J. COUTURIER
Département d'Evaluation de Sûreté (DES),
Institut de Radioprotection et de Sûreté Nucléaire (IRSN),
Fontenay-aux-Roses, France

Abstract

After a general overview of the technological irradiation reactors existing in Europe, the paper details the safety approach adopted for a new research reactor to be constructed in France and the main recommendations resulting from IRSN assessment.

1. INTRODUCTION

As shown in Table 1, the existing technological irradiation reactors in Europe are now ageing.

The French Atomic Energy Commission (CEA) implemented a new project of pool-type reactor called the Jules Horowitz Reactor (RJH). Its construction is planned on the Cadarache site with an operational start by 2013. In 2002, the CEA transmitted an application to the General Directorate for Nuclear Safety and Radiation Protection with regard to the safety options selected for this project. The CEA, designer of new nuclear plants, stated its intention to obtain an increased safety level.

TABLE 1. TECHNOLOGICAL IRRADIATION REACTORS IN EUROPE

State	Reactor type	Thermal power (MW)	Year built
Norway	HBWR	25	1959
Sweden	R-2	50	1960
Belgium	BR-2	80	1961
Netherlands	HFR	45	1961
France	OSIRIS	70	1966

The search for improvements should be an ongoing concern with regard to safety. For existing plants, improvements are implemented in accordance with a pragmatic method, considering the design limitations, during scheduled safety reviews. For new plants, a significant safety improvement is required at the design stage.

This significant step, at design stage, is possible in an 'evolutionary' roadmap if the necessary attention is paid to the lessons learned from the operating experience, the in-depth studies conducted for existing plants, and the results of safety researches. However, the introduction of innovating provisions should also be considered, in particular for preventing and controlling major accidents.

The purpose of this paper is to highlight the strong points, as safety improvements, resulting either from options deliberately selected by the CEA, or from recommendations resulting from Institut de Radioprotection et de Sûreté Nucléaire (IRSN) assessment, presented to expert groups, then notified by the relevant safety authorities. This paper deals especially with the case of the Jules Horowitz Reactor, the only planned research reactor in France in the past 20 years.

2. ASSESSMENT OF THE SAFETY OPTIONS FOR THE JULES HOROWITZ REACTOR

2.1. Description and basic design options

The construction of the Jules Horowitz Reactor is planned on the CEA Cadarache site and will be a pool-type research reactor, cooled and moderated with light water; its maximum thermal power will be approximately 100 MW.

The general layout of this installation is illustrated in Fig. 1. This reactor is composed of a reactor building (BR) and a nuclear auxiliary building (BAN), as well as a control building and other installation support buildings. The reactor building, with a cylindrical shape (approximately 36 m in height and diameter), will be partitioned in two geographically distinct zones, one compartment for reactor operation and one for the operation of experimental devices. The nuclear auxiliary building ('parallelepiped') will house, in particular:

— A 'hot zone', designed for the preparation and conducting of experiments. It will include hot cells, pools and laboratories for analysis, on-line or post-irradiation measurements;

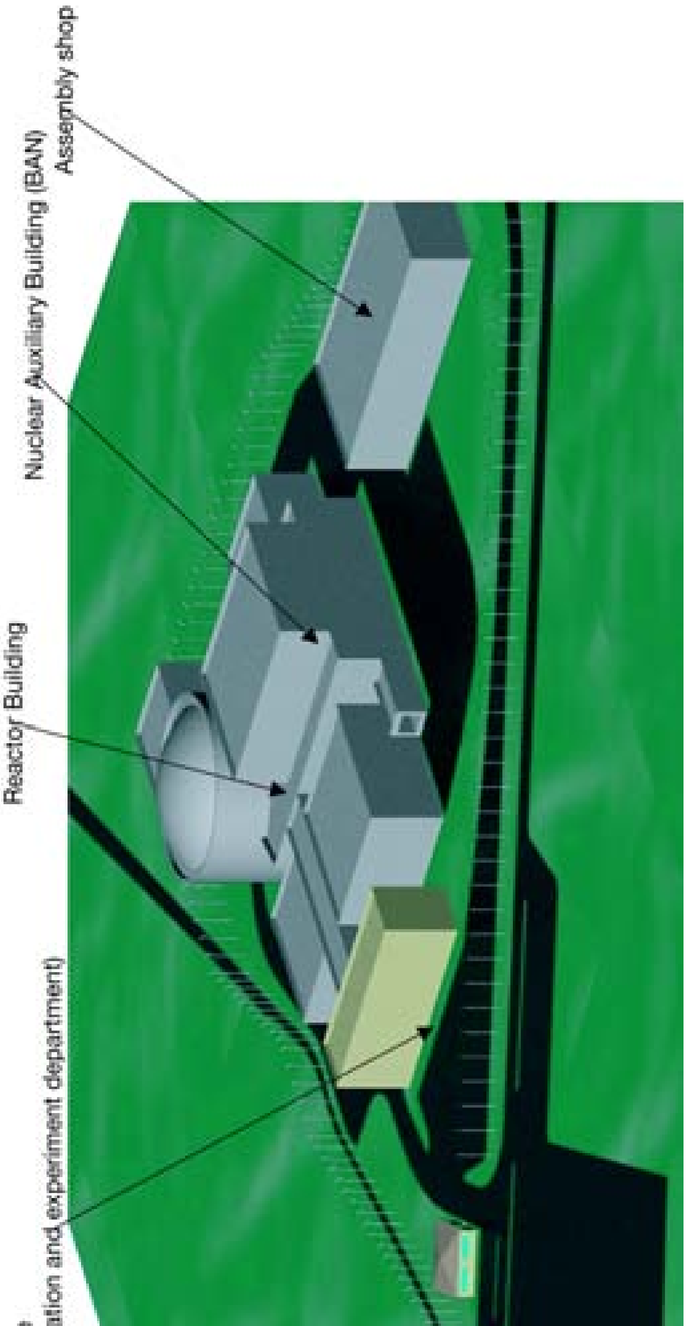


FIG. 1. General installation of the Jules Horowitz reactor.

— A ‘cold zone’, housing the components providing nuclear auxiliary systems required for the operation of the installation.

The nuclear unit (BR and BAN) will include only one civil engineering infrastructure. The containment between BR and BAN will be differentiated in the superstructures. The common base mat will support the ‘water block’, a single block assembly made of reinforced concrete, which will gather the pools (reactor, intermediate storage and work), the primary system bunkers and a transfer channel between the various pools and hot cells. It will rest on an aseismic support system.

A great variety of experiments are considered. In particular, it is planned to conduct, in the Jules Horowitz Reactor installation, experiments intended for irradiating:

— Actinides, or products including a high content of α emitters;
— Samples, located in core or in reflector zone, under environmental conditions (pressure, temperature and coolant fluid), significantly different from the reactor core conditions (via core experimental loops);
— Fuel samples, under degraded thermohydraulic conditions (experiments on broken fuel elements with experimental sequences conducting to sample melting).

The design options selected for this pool-type reactor are closely derived from the existing research reactors (especially OSIRIS and ORPHEE), although they involve significant evolutions. In addition to the reactor building and nuclear auxiliary building aseismic insulation, using elastomer support devices, these evolutions include:

— Composition of the driving core, consisting of low enriched uranium 235 (less than 20%) fuel plates (see Fig. 2); this will be a UMo-AL alloy, selected for non-proliferation and reprocessing possibility;
— Use of closed cooling systems, not only for power removal, but also for residual power removal (primary system and safety system) (see Fig. 3).

The following options should also be noted:

— The ‘water block’ concept aimed at preventing the uncovering of the core;
— The leak recovery zone (‘buffer’ space between BR and BAN containments); the reactor building will thus include no direct outlet to the environment.

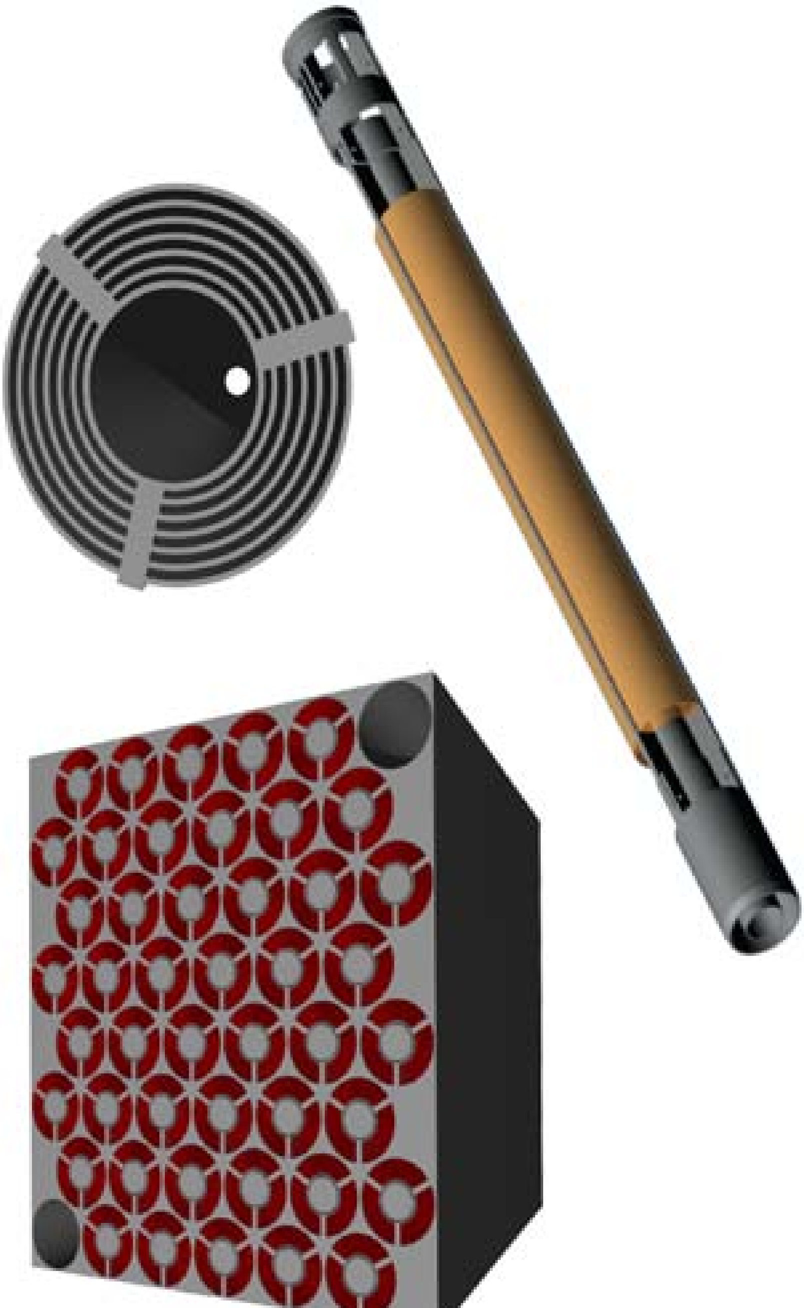


FIG. 2. Core and fuel elements of the Jules Horowitz reactor.

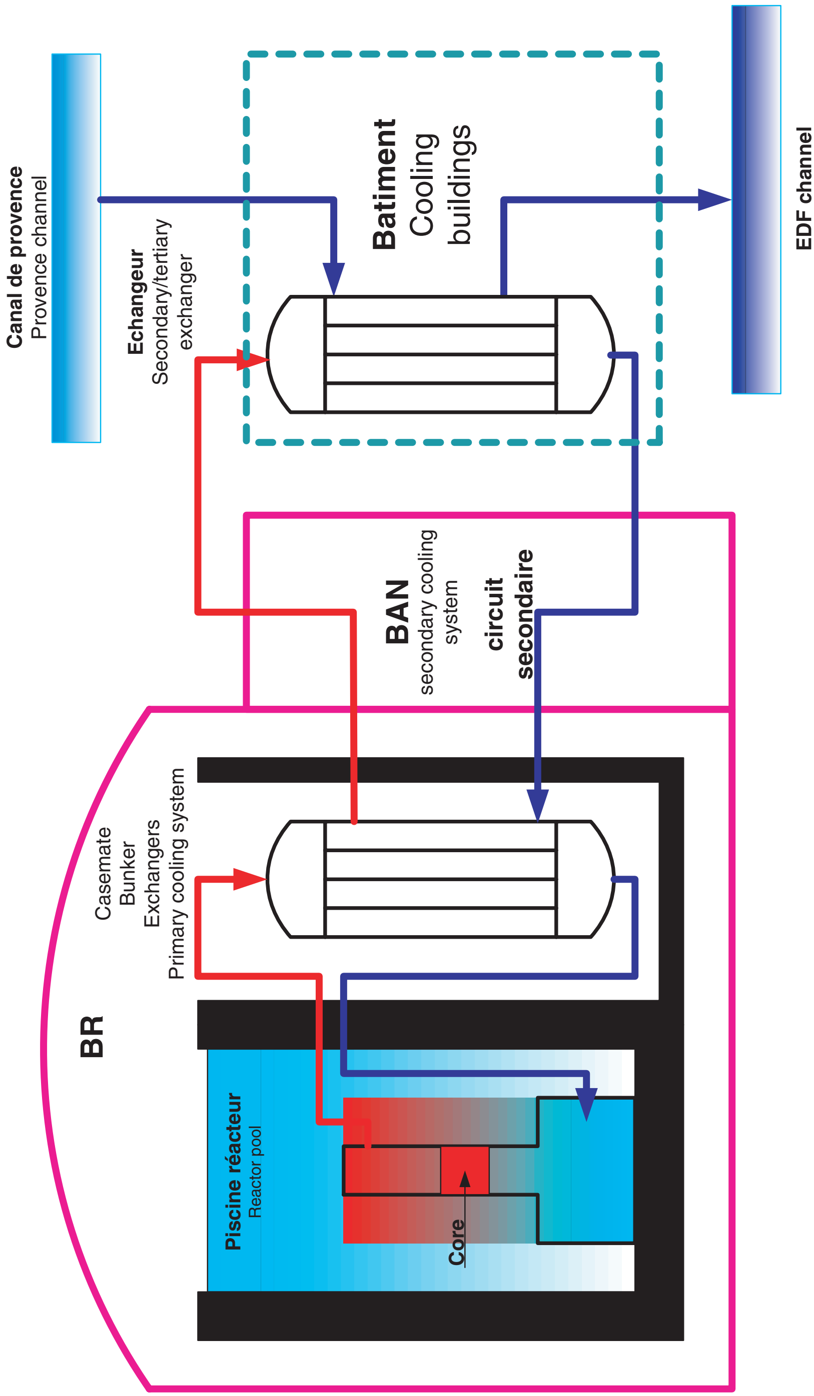


FIG. 3. Functional diagram of the Jules Horowitz Reactor cooling systems.

2.2. Topics reviewed and resulting conclusions

The review requested the following from the permanent group especially concerned:

— Justification for the site selected for the future reactor;
— Safety options;
— Reactor design and sizing basis.

The technical assessment conducted by the IRSN was presented to the Expert Group for reactors during three meetings, on 20 March 2003, 3 April 2003 and 24 April 2003. During this assessment, difficulties were reported in connection with inaccuracies in the safety options file, as well as inconsistent or non-consolidated positions from the operator with regard to certain topics.

2.2.1. General methodology of the safety analysis

The safety analysis approach for the Jules Horowitz Reactor will be aimed at demonstrating that the provisions taken with regard to the various in-depth defence levels are sufficient versus general safety objectives defined for this plant, both in normal operation and after incident and accident situations. Normal operation will include maintenance operations.

Despite all measures taken with regard to design and development quality, the failure possibilities and associated risks will be investigated. This identification of risks that have an impact on the plant condition, or even the environment, will result in a classification of these risks in accordance with the frequency–consequence couple, resulting in a list of operating situations according to the classification specified in Table 2.

TABLE 2. OPERATING SITUATIONS AND RISK PROBABILITY

Definition	Occurrence probability value (/a)
1. Normal operating conditions as defined in the operation technical specifications	$p = 1$
2. Incidents whose consequences should remain limited	$1 > p \quad 10^{-2}$
3. Accidents 3(a) Complex sequences with multiple failures whose consequences should remain acceptable	$10^{-2} > p \quad 10^{-6}$
4. Controlled severe accidents	

The safety approach proposed by the operator is based on design choices that include:

— Definition of a system safety classification, in accordance with six criteria;
— Gradation of requirements applied to safety classified systems.

Taking into account, from the design stage, of:

— Risks of common failures and human errors;
— Internal and external hazards, earthquakes (MHPE and SSE) being combined with operating situations in compliance with Fundamental Safety Rule IV.2.a;
— Severe accidents;
— Risks, for the plant, associated with reactor experimental device failures.

In addition, an overall safety analysis is based on:

— General safety objectives, expressed in terms of radiological consequences;
— Analysis of operating situations, not yet defined, which will be classified according to their estimated occurrence probability after inventory and grouping, in a limited number of families, of initiator events;
— Adding a single failure for this analysis (second and third category situations);
— Analysis of internal hazards (e.g. fire, internal flooding, internal explosion) and external hazards (earthquake, airplane crash), with the aim that a hazard should not result in an accident situation.

With regard to controlled severe accidents, the operator assumes:

— A BORAX-type reactivity accident, in the reactor building, leading to destruction and partial melting of the core;
— A generalized fire in the hot cell, in the nuclear auxiliary building, with the failure of associated filtration systems.

Severe accidents, from the design stage, both through preventive provisions and provisions for managing and attenuating their consequences, are taken into account and should allow for the strengthening of the plant's in-depth defence. Furthermore, the operator provided a list of so-called excluded situations, such as air melting of fuel and pool de-watering, for which he will demonstrate that he can run a sufficient number of robust preventive provisions. Finally, the experimental aspect more specially results in the following objectives:

- Physical separation between reactor operating systems, experimental device operating systems and safety systems;
- Design of the installation, in order to allow implementation of new components;
- Taking into account the failure of experimental devices and associated operations.

The approach, basically similar to the approaches applied in France for pressurized water reactors in the Electricité de France plants and the Superphénix fast neutrons reactor, shows marked progress for research reactors.

2.2.2. Positions and recommendations resulting from the assessment conducted by the IRSN

As a result of the assessment conducted by the IRSN, a number of positions and recommendations were issued, and most of them are examined below. It should be noted that some recommendations (especially the wide application of the 'hazard–event' approach, effects of aircraft fuel fire) represent progress with regard to reactor safety, and not only for research reactors.

2.2.2.1. Selection of the installation site

The selected site for installation of the Jules Horowitz Reactor results from an approach involving three steps:

(1) *Selection*: After collecting and studying the available documents, three potential sites were pre-selected. After field studies, which improved the existing mapping, two of the three pre-selected sites were selected for installing cored and destructive borings;

(2) *Qualification*: This step was associated with a more detailed characterization of both selected sites (hydro-geological monitoring, tectonic, seismotectonic and seismological surveys). On completion of this step, the site with the most appropriate characteristics for the installation of the Jules Horowitz Reactor was selected;

(3) *Confirmation*: The purpose of this step was to conduct a detailed investigation of the selected site, in particular, geotechnical studies in order to define the structure calculation parameters.

The selected site, Bargette East Sector on the Cadarache site, raised no objections; however, the operator was asked to pay particular attention to the risk of water table rise, considering the high water level reported in the past.

2.2.2.2. Selection of an aseismic insulation

It appeared that selection of an aseismic insulation for the reactor building and the nuclear auxiliary building, through elastomer support devices, required the operator to perform an in-depth analysis of the technological control of such a design, especially in terms of ageing, vulnerability to malicious actions and compatibility with the control of the consequences of a major accident.

2.2.2.3. 'Clean reactor' option

The fact that the operator did not state a clear position with regard to the application of the usual 'clean reactor' option (unloading a fuel assembly showing a clad failure, as soon as detected) selected for research reactors was vigorously discussed. Consistent with the rules stated by the IAEA for research reactors, it was requested, during operation, to limit the operating time with a degraded fuel element in the feeding core. In this respect, the operator should describe, in the preliminary safety analysis report (PSAR), the provisions planned in case a broken clad is detected, specifying in particular the selected approach to define the warning and automatic core shutdown thresholds, the surveillance systems used, and the modes of identification and unloading of degraded elements.

2.2.2.4. Safety analysis approach: operating situations

The safety analysis approach described in the safety options file for the design of the Jules Horowitz Reactor, based on the definition of reference situations studied with deterministic rules, was considered as satisfactory in principle. However, the operator was requested to:

— Apply the usual classification of accident situations, corresponding to a single initiator event, in two categories;

— Prepare, as a support to the lists of operating situations to be presented in the RPrS for the various parts of the Jules Horowitz Reactor (reactor, cells, experimental devices, etc.), a file describing the identification of initiator events, how they were grouped in envelope operating situations, and how the situations were classified in the various categories; the interest of using a probabilistic risk assessment (PSA), in particular to confirm the list of operating situations, will be reviewed at a later stage;
— Specify the list of so-called complex situations (multiple failures situations), the associated frequency, and the study rules selected for assessing their consequences;
— Specify the elements that allow considering a situation or a scenario as ‘excluded’;
— Demonstrate, in the PSAR, that controlled major accidents selected for the reactor part and the nuclear auxiliary parts are highly improbable, through the availability of preventive provisions equivalent to two strong lines of defence;
— Finally, transmit a first list of operating situations.

2.2.2.5. Safety analysis approach: taking into consideration acts of aggression

The operator was requested to implement a ‘hazard–event’ approach, not only for earthquakes, but also for hazards such as extreme meteorological conditions. this approach involves the following steps:

— Identify the equipment essential to maintain or restore the safe condition of the installation during such a hazard and the direct consequences;
— Verify that the design of this equipment withstands the initial hazard;
— Check that the failure of installation equipment not sized for the initial hazard would not affect the safety related equipment or, as a minimum, would not prevent the required safety actions from being undertaken;
— Verify, otherwise, that the design of the equipment in question also withstands the hazard and/or the implementation of protective provisions for safety related equipment;
— Specify the approach for incorporating extreme meteorological conditions, especially combinations of operating situations and other hazards;
— Classify the ‘safety’ of those systems whose failure in case of internal or external hazards would result in an accident situation;
— Incorporate the fuel fire effects added to the direct effects of the impact of selected aircraft, as well as the airborne resources used in case of external firebreak for studying the risks of aircraft accidents;

— Propose an argued position with regard to the importance of fitting the reactor with an automatic emergency shutdown in case of earthquake, consistent with the rules states by the IAEA for research reactors;
— Assume a firebreak in any room and do not systematically exclude from the analysis the fire hazard in rooms where ignition sources were not in evidence.

2.2.2.6. Single failure criterion

With regard to the application of the single failure criterion, it was requested that the operator should pay particular attention to the presence of check valves in the design of the various systems, when these components are considered as passive.

2.2.2.7. BORAX-type reactivity accident

In the options considered for the design of the reactor containment system, the operator selected for the controlled severe accident category (AGM), a BORAX-type explosive reactivity accident. The selected accident is characterized by 50% melting of the core fuel, thermal energy of 135 MJ deposited in the fuel during the accident, added to the energy initially contained in the fuel, and a mechanical energy release corresponding to 5% of the thermal energy released during the accident, i.e. 6.75 MJ. These values may be increased by the values of the thermal and mechanical energies that may be released by the experimental devices installed in the reactor and that may be damaged during the accident.

The operator also planned studying the radiological consequences of the accident, assuming the melting of the whole core, as well as the post-accident cooling of the damaged core, with reasonably enveloping hypotheses and considering the various possible effects, particularly on the containment, of mechanical energy release. Although consistent with those selected in the past for certain research reactors (OSIRIS, ORPHEE, etc.), the hypotheses mentioned are arbitrary.

For a new research reactor, such as the Jules Horowitz Reactor, it appeared necessary for the operator to provide the elements to confirm the envelope character of the values of released thermal and mechanical energy and selected hypotheses for assessing the consequences of an accident. In this respect, the operator should, in particular:

— Identify the accident sequences that may result in melting of the core and correspond to the notion of controlled severe accident, in terms of probability;
— Assess for these sequences the energy deposited in the fuel, using the tools currently available;
— Also assess, using the tools available, the mechanical energy developed by the release of the steam bulb generated during the thermodynamic interaction between the melt fuel and water;
— Assess the transfers of radioactive products into the hall, the risks of solid material or existing pool systems projections, etc.

2.2.2.8. Fuel

The UMo-Al alloy is selected as the reference solution for the core fuel (and U3Si2-Al as the fallback solution). This new type of fuel requires a qualification, which should especially demonstrate its compliance with the fuel response requirements selected in the various operating situations (‘service limits’), including accident situations. In this respect, the operator was requested to select as a target, for the feeding core, the absence of clad breaks, not only for first category situations, but also for second category situations. Furthermore, it appeared necessary that the radionuclide release rates selected in case of fuel degradation be confirmed by a sufficient experimental data basis.

2.2.2.9. Experimental devices

With regard to the safety options relating to experimental devices, the operator’s overall target to match the requirements applied to these devices, depending on the safety stakes, was considered satisfactory. However, the operator was requested to define the device failure cases representing situations to be considered when sizing the installation.

The operator was also requested to specify:

— Firstly, the notion of full failure of a device, in particular, how the various events that may generate stress on the device barrier(s) are considered;
— Secondly, the approach planned for designing and sizing the device barriers and the systems associated with these barriers;
— Human and organizational factors;
— The operator was requested to detail, in the PSAR, the action plan implemented to incorporate the human and organizational factors throughout the design process, as well as the initial results of implementation for this action plan.

3. CONCLUSION

As a general conclusion, the safety related provisions taken by the Atomic Energy Commission of France (Commissariat à l'énergie atomique—CEA) for the Jules Horowitz Reactor will allow a safety level to be obtained that is consistent with the future power generating reactors safety level with, marking the most significant progress, the consideration from the design stage, of severe core meltdown accidents from all points of view (control of the removal risk in containments, post-accidental cooling of relocated materials, etc.). Generally, it was considered that these provisions constitute a satisfactory 'evolutionary' basis for the continuation of the project.

However, the assessment conducted by the IRSN led to the issuing of a number of recommendations relating, in particular, to some relatively arbitrary assumptions and approaches (for example, the energies considered for the BORAX accident).

The plans for new facilities implemented by the CEA seem to be moving in the right direction: in an 'evolutionary' way, not a 'revolutionary' one.

SAFETY ASPECTS OF COLD NEUTRON SOURCES IN RESEARCH REACTORS

K.H. GOBRECHT
Grenoble
Email: klaus@ottosix.com

J.-M. ASTRUC
Institut Laue-Langevin,
Grenoble

France

Abstract

There are 16 cold neutron sources (CNS) operating in research reactors (RR), two under construction, about five planned and 11 decommissioned. Some CNS have been built at the same time as the reactor itself, others have been back-fitted, either in the frame of a major power upgrade or without touching general layout of the reactor. The CNS are secondary moderator volumes placed close to the core of the RR, and cooled to cryogenic temperatures. The moderation lowers the energy of the neutrons to below 20 meV, an energy range extremely useful for studying a variety of topics, such as molecular structures, stress in material, magnetics and even the properties of the neutron itself. The safety risks are of three kinds: chemical, nuclear and cryogenic. These risks are considered in detail, and examples are given about how to cope with them. Topics dealt with in the paper are definition and use of a cold neutron source; different design types and safety aspects, in particular, safety philosophy, chemical risks, nuclear risks and cryogenic risks.

1. DEFINITION AND USE OF A COLD NEUTRON SOURCE

A cold neutron source (CNS) is a secondary moderator volume placed close to the core of the research reactor (RR), and cooled to cryogenic temperatures (see Fig. 1).

All CNS are built according the following scheme:

— Close to the reactor core: the moderator chamber with its vacuum jacket;
— Pumping line, filling line, cryogenic lines leading to out-of-pile components;

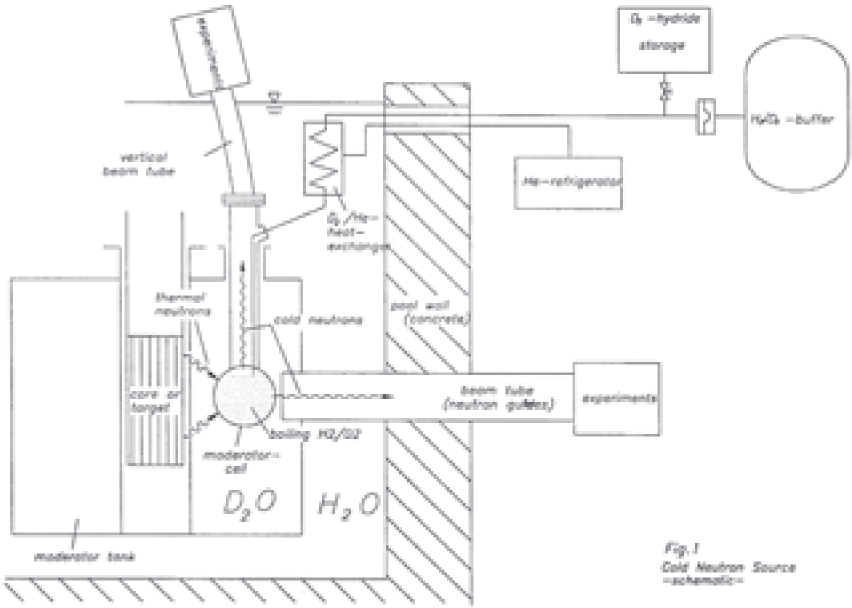


FIG. 1. Cold neutron source—schematic.

— Moderator filling and storage system;
— Vacuum system, refrigerator with compressor;
— Control system.

The moderation lowers the energy of the neutrons to below 20 meV, an energy range extremely useful for studying a variety of topics, such as molecular structures, stress in material, magnetics, chemical composition and even the nuclear properties of the neutron itself.

Efficient moderator substances should have a high inelastic scattering cross section and low absorption. Liquid hydrogen or hydrogen-rich chemical compounds, as well as liquid deuterium, are the best moderators. The cold neutrons escape from the moderator chamber with an energy distribution corresponding to the moderator temperature, usually well below 100 K (see Fig. 2). To get a beam of cold neutrons out of the reactor, one collects them in one or several beam tubes crossing the biological shield. Cold neutrons can be guided over several 10 m to be used on neutron scattering instruments and other experiments with neutrons.

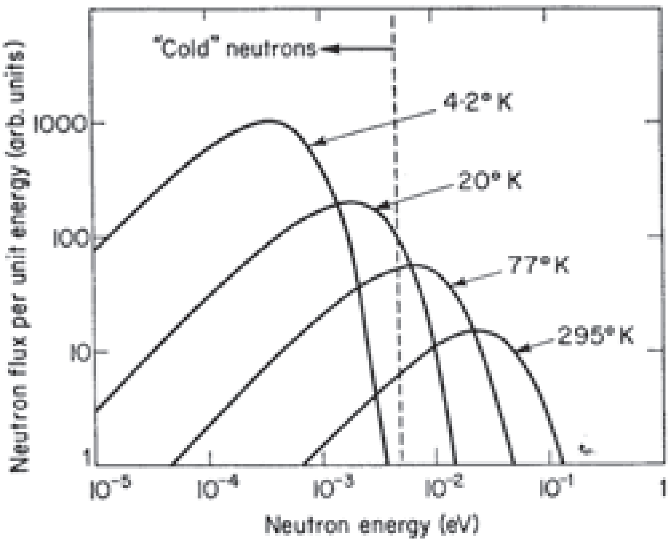


FIG. 2. Theoretical spectral distribution of neutron flux in cold and warm moderators.

2. DIFFERENT DESIGN TYPES

The 29 CNS which are operating, under construction, or decommissioned, can be classified by the following features (see also Table 1):

— Composition and state (solid or liquid or gaseous) of the moderator;
— Process of heat removal.

2.1. Composition and state of the moderator substance

As mentioned before, hydrogen is, together with deuterium, the best moderator. It is kept cool at about 20 K, in order to slow down as many neutrons as possible to velocities below 1 km/s, corresponding in energy to about 5 meV. It is used either as a liquid or, only for hydrogen, also in the supercritical state at a high pressure such that the proton density is close to the one in the liquid. Isotopic mixtures are sometimes used, too. The liquid or supercritical fluid can easily be cooled by contact with the heat exchanger of a refrigerator.

Hydrogenous compounds, such as methane, water, mesitylene or clathrates are used because of their high hydrogen densities, even at higher

TABLE 1. COLD NEUTRON SOURCES, OPERATING, UNDER CONSTRUCTION OR PLANNED

Location	Reactor	Power (MW)	CNS type	Temperature (K)	Heat load (W)
Austin, TX, USA	TRIGA Mark II	1	Mesitylene	40	16
Beijing, China	CARR	60	Project		
Berlin, Germany	BER-2	10	Hydrogen gas	28	1800
Budapest, Hungary	WWR	10	Liquid hydrogen	20	250
Cairo, Egypt	ETRR2	22	Project		
Delft, Netherlands	HOR	2	Liquid hydrogen	Project	
Dubna, Russian Federation	IBR-2 pulsed	2	Solid methane	30–70	150
Gaithersburg, USA	NIST NBSR	20	Liquid hydrogen	20	850
Garching, Germany	FRM2	20	Liquid deuterium*	25	5000
Gatchina, Russian Federation	WWM-R	15	Ld2 + lh2	20	4000
Gatchina, Russian Federation	PIK	100	Project		
Geesthacht, Germany	FRG2 MTR	5	Hydrogen gas	28	1150
Grenoble, France	ILL HFR	58	Liquid deuterium	25	6500
Grenoble, France	ILL HFR	58	Liquid deuterium	25	3000
Juelich, Germany	DIDO	23	Liquid hydrogen	19	700
Kjeller, Norway	Jeep-2	2	Liquid hydrogen	21	60
Kyoto, Japan	KUR	5	Liquid deuterium	25	250
Lucas Heights, Australia	RRR	20	(Liquid deuterium)	Project	(4000)
Mianyang, China	CMRR	20	(Liquid hydrogen)	Project	(1500)
Oak Ridge, USA	HFIR	85	(Hydrogen gas *)	20	3000
Saclay, France	ORPHEE	14	Liquid hydrogen	20	500
Saclay, France	ORPHEE	14	Liquid hydrogen	20	650
Serpong, Indonesia	Siwabessy	30	Project		
Taejon, Korea	Hanaro	30	(Liquid hydrogen)	Project	
Tokai-mura, Japan	JRR-3M, Jaeri	20	Liquid hydrogen	20	350

* Expected in 2004.

temperatures. They suffer from decomposition or polymerization under nuclear radiation (radiolysis, see below).

All moderator substances (besides water) react more or less violently with oxygen, a fact which has to be considered a serious safety hazard.

2.2. Process of heat removal

The great advantage of hydrogen and its isotopes is that they stay fluid at very low temperatures. The heat generated in the moderator by nuclear radiation can, therefore, be removed easily by the moderator fluid itself. The fluid circulates in a closed cycle between the moderator chamber and a heat exchanger, which can be located relatively far away from the reactor core, and which is cooled by an industrial refrigerator (see Fig. 3).

We distinguish between natural convection cooling (gravity driven) and forced convection (driven by a pump or a blower). In the first case (also called thermal siphon), the heat exchanger is placed at a level above the moderator chamber, from which the warm (low density) fluid lifts up, is cooled in the heat exchanger and, being more dense, flows back.

Solid moderators are directly cooled by the refrigerant (e.g. He gas) circulating from and to the refrigerator. The direct cooling limits the heat removal to a few W while the closed loop cooling can handle up to several kW.

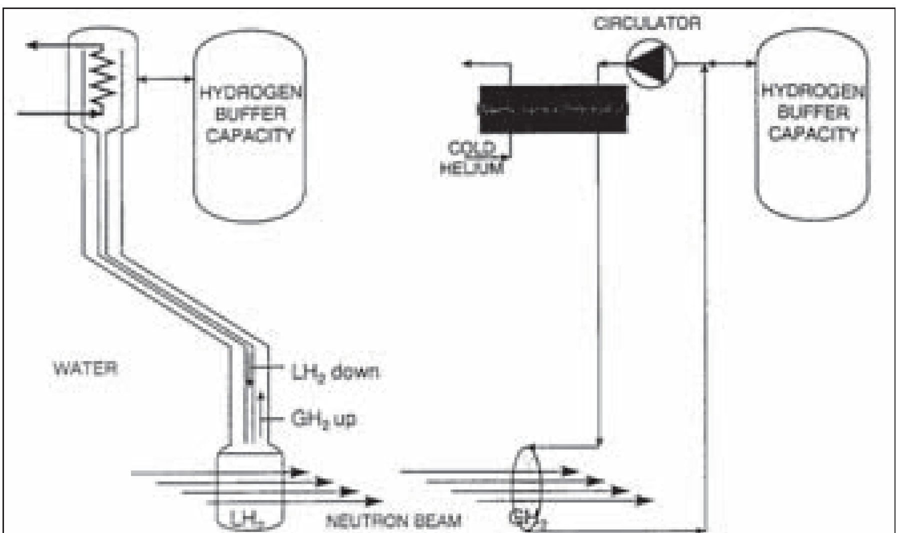


FIG. 3. Methods of heat removal: natural convection (left) and forced convection (right).

3. SAFETY ASPECTS

The safety aspects addressed in the following discussion are safety philosophy, chemical risks, nuclear risks and cryogenic risks.

3.1. Safety philosophy

Practically all research reactors with a CNS adopt the following safety philosophy:

— Any anomalous behaviour of the CNS shall not affect the safety of the reactor itself;
— No explosive mixture with air shall develop in the reactor or on the RR site;
— No radioactive contamination above authorized levels shall affect the environment.

If the reactor is shut down in an emergency because of a dysfunction of the CNS, then only in order to avoid damaging the CNS.

The installation and operation of a CNS in an RR is always subject to a licensing procedure, no matter whether it is constructed in parallel with a new reactor construction, or back-fitted into an existing RR. The Safety Analysis Report (SAR) is best elaborated according to IAEA guidelines (see Refs [1, 2]).

The most recent safety analysis reports have been edited in Oak Ridge, USA (these reports are not public; information can be obtained from D.L. Selby, at yb2@ornl.gov); at NIST (these reports are not public; information from R.E. Williams, at robert.williams@nist.gov), in Garching, Germany at FRM2 [3], and in Sydney, Australia by ANSTO (these reports are not public; information can be obtained from R. Miller, at rmx@ansto.gov.au) (see also Table 1).

3.2. Chemical risks

The risk of explosion of the moderator substance in contact with air can be minimized by creating at least two solid barriers between the moderator and the atmosphere. This double containment concerns the components close to the reactor core and inside the reactor building. Two barriers usually enclose an inert gas liner (helium or nitrogen), the pressure of which is above atmospheric pressure. The integrity of the barriers is ensured by continuous leak testing. In spite of the fact that the occurrence of an explosive mixture with air is extremely improbable, some regulators require that at least one barrier withstand the pressure peak of an atmospheric explosion in the moderator chamber or in the cryogenic vacuum jacket.

For liquid moderators, it is important that the gas has the required purity (absence of oxygen) from the beginning, and that eventual refills are made with high purity gas.

In solid moderators, there are two additional risks, namely, the development of radiolysis products and of lattice defects due to the strong nuclear radiation field. In both cases, the recombination can be sudden and strongly exothermal, leading to explosion-like pressure peaks. The best countermeasure is to anneal the moderator substance from time to time at a high enough temperature.

3.3. Nuclear risks

These are of two kinds: effects on reactivity, activation of the moderator substance and structure.

A sudden evaporation or leakage of the moderator creates a reactivity surge which has to be compensated for by the reactor control system. In practice, this effect is usually negative and never fast enough to provoke a power excursion. Nevertheless, the reactivity change is calculated and the worst case assumption considered in the layout of the reactor control system.

The moderator is activated in the continuous flux of thermal and cold neutrons. Especially in a deuterium moderator, a significant amount of tritium builds up with time (at ILL Grenoble 0.5 ppm/cycle, equivalent to about a GBq). But, since the system is gas tight and surrounded by a containment for the reasons enumerated above, the environmental contamination risk is very low. The activated moderator and the activation of in-pile CNS structure material (in particular, zircalloy) represents a risk, however, if the CNS is decommissioned.

3.4. Cryogenic risks

In the case of a failure of the refrigerator, the moderator no longer cools the moderator chamber, it warms up, and there is a risk of destruction the chamber if the reactor is not shut down in time. Some reactors (BER2, FRG2, RRR, WWR Budapest) continue operation with an emergency cooling of the moderator chamber, others can only restart when the refrigerator is operating again (ILL HFR, ORPHEE). There is a strong demand for availability of the refrigerator, MTBF of >1000 hours typically. Some installations have doubled sensitive components for redundancy.

All fluid moderators solidify if the temperature becomes low enough. If this happens, there is a risk of blockage of the moderator flow followed by a warm-up in the moderator chamber because of lack of cooling. The moderator then evaporizes and eventually destroys the moderator chamber. A good temperature and/or pressure control (usually with 2-of-3 logic) in the moderator circuit is required in order to minimize this risk.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report, Safety Series No. 35-G1, IAEA, Vienna (1994).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety in the Utilization and Modification of Research Reactors, Safety Series No. 35-G2, IAEA, Vienna (1994).

[3] Private communication from FRM2 Garching (klaus.schreckenbach@frm2.tum.de). More references with details of each existing or projected CNS are given in the OTTOSIX homepage (www.ottosix.com).

RESEARCH REACTORS THAT LOOK SIMILAR BUT ARE QUITE DIFFERENT

C.M. MAZUFRI, A.S. DOVAL
Nuclear Engineering Division,
INVAP S.E.,
Bariloche, Argentina
Email: mazufri@invap.com.ar

Abstract

At the end of 1997, the 22 MW ETRR-2 was started up and in November 2001, the 20 MW design, offered by INVAP, was selected by the Australian Nuclear Science and Technology Organisation (ANSTO) to be the new replacement research reactor. Overall, both reactors generate practically the same power and the primary cooling system presents similarities that can be considered as the use of a proven technique. However, the required core configuration introduces some important challenges in the design. The paper is intended to show the experience of INVAP in the thermal-hydraulic design of two 'similar' research reactors specified with different requirements and fulfilling the same safety standards.

1. INTRODUCTION

Sometimes what appears at a glance to be the same design approach is not the same. An example of this is the case of two research reactors generating the same power, with almost the same fuel type, the same coolant and flow rate. However, they could be quite different when their core heat fluxes are look at and, as a consequence of this, their coolant velocities, among other characteristics, is taken.

The importance that arises from what seemed to be a single extrapolation of a similar reactor design becomes a real challenge, as it results in several verifications to confirm a safety design. A methodical review of the different aspects of the reactor core and primary cooling system allows the main design topics and its differences to be identified, and the solutions to be foreseen.

The present work is intended to show the experience of INVAP in the thermal-hydraulic design of two 'similar' research reactors fulfilling the same safety standards.

2. CORE COOLING SYSTEM CHARACTERISTICS

A brief description of the core and the primary cooling system of both similar facilities, ETRR-II and RRR-ANSTO [1, 2] is provided. In both cases, the core consists of fuel plate-type assemblies placed vertically in a reactor grid. A chimney shroud surrounds the core, which is internally divided, by the control plate guide boxes housing the control plates. The external plates of each fuel assembly are longer and in the lower part they have a window allowing a bypass flow for cooling purposes.

Regarding the primary cooling system, two inlet pipes drive the coolant towards the inlet plenum. A perforated plate is used to homogenize the velocity distribution upstream the core and to minimize any possible flow induced vibration on the core structure. From the plenum, the coolant flows upwards through the core and the chimney or riser and reaches the outlet pipe, towards the two primary coolant pumps (Fig. 1).

A closure flow moving downwards inside the chimney plays the role of a hydraulic plug avoiding the release of activated water coming from the core and minimizing the dose rate in the reactor hall. Downstream, the pumps there are two heat exchangers, one on each line, which drive the coolant towards the inlet pipes, closing the loop.

Main data for both reactors are summarized in Table 1.

TABLE 1. MAIN THERMAL-HYDRAULIC DATA FOR BOTH DESIGNS

	ETRR-2	RRR
Fission power (MW)	22	20
Amount of fuel assembly	29	16
Amount of fuel plates	19	21
Core flow (m^3/h)	1900	1900
Average core coolant velocity (m/s)	4.7	8.6
Core pressure drop (kPa)	60	200
Core flow direction	Upwards	

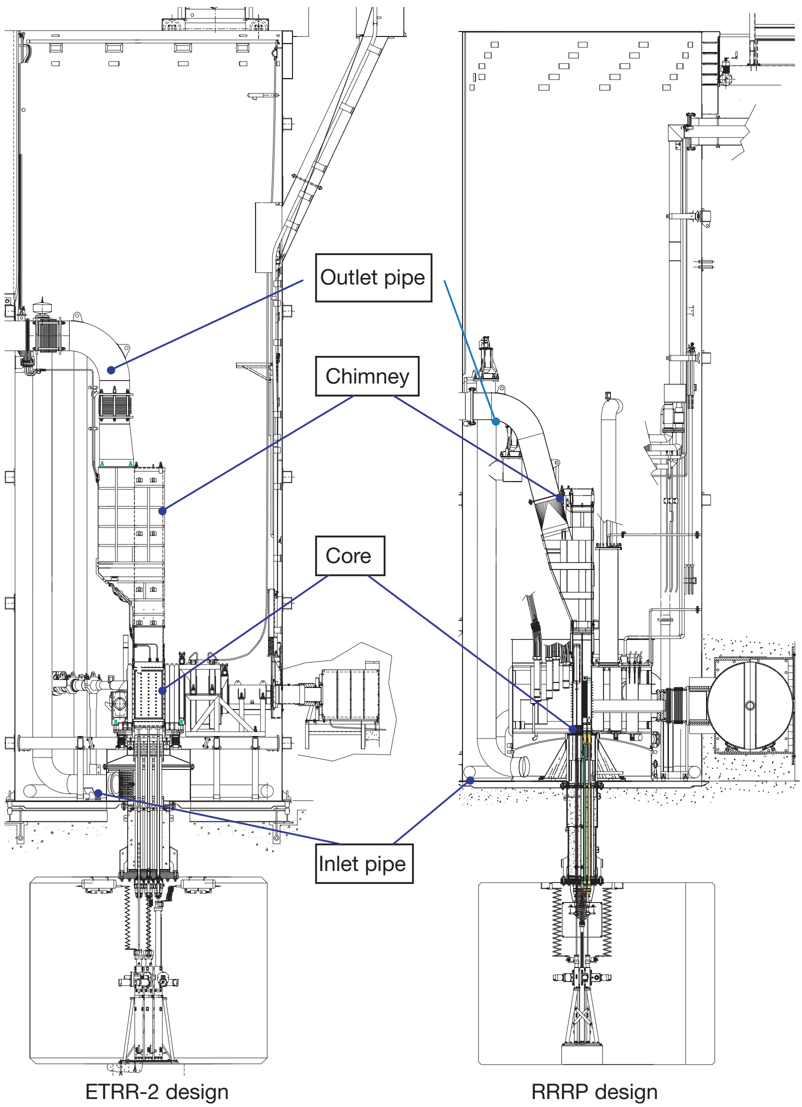


FIG. 1. Core and main components in the reactor pool of both designs.

3. NECESSARY STUDIES ACCORDING TO THE NEW REQUIREMENTS

A design requirement for the RRR was the definition of a more compact core configuration. As a consequence, the coolant velocity in the fuel assemblies increased considerably, giving rise to a series of verifications of the previous design.

Another requirement was the use of heavy water as the reflector. The resulting 'new component', the reflector vessel, underwent particular studies and verifications from the thermal point of view, due to its dimensions and requisites.

These necessary verifications and studies were performed during the design stage in two different ways:

— Experimental tests;
— The use of modern analytical tools.

In the discussion that follows, a description of verifications in those components affecting the safety and performance requirements of the reactor is carried out.

4. VERIFICATIONS BY MEANS OF EXPERIMENTAL TESTS

4.1. Fuel assembly

The requirement of a higher coolant velocity in the fuel channels impacts directly on the structural design of the fuel assembly. In addition to the mechanical design being reconsidered, the hydraulic stability was also verified.

To cope with the high coolant velocities reached during normal operation, a new structural device was introduced to supply more flow stability at the entrance of fuel parallel plates.

Several tests were carried out over a dummy fuel assembly to qualify the structural design, stability and fabrication techniques under new higher hydrodynamic loads, hydrodynamic drag and wear phenomenon. Figure 2 shows a picture of the dummy fuel assembly.

The tests were performed in an experimental loop using a single fuel assembly and they included the verification of the pressure drop curve, not only of the fuel assembly, but of its clamp element and its relative contribution, as well (see Fig. 3). Vibration levels in the fuel and clamp were also verified.

FIG. 2. Dummy fuel assembly.

A comparison of the fuel plate stability without the plate separator device confirmed that for a coolant flow rate of about 170% of the nominal value, the mechanical limit is reached, as shown in Fig. 4.

4.2. Core pressure drop and control plates

It is very well known that control rod plates must be cooled under any condition, either normal, abnormal or accidental, in order to preserve the plate integrity and its safety function.

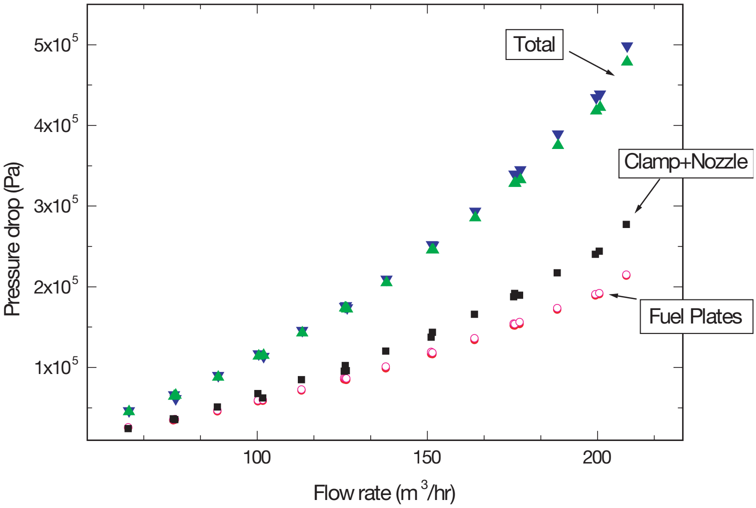


FIG. 3. Measured pressure drops in the fuel assembly and clamp.

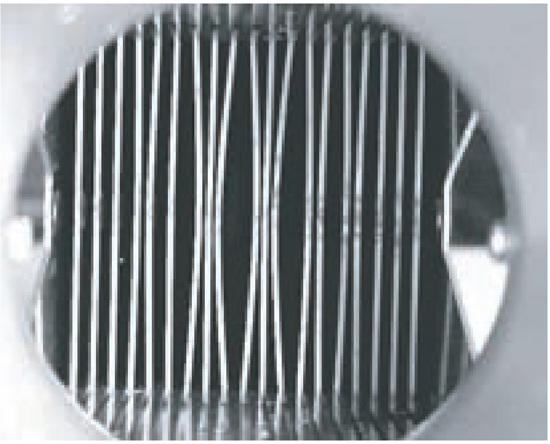


FIG. 4. Structural limit of stability in the fuel plate.

With regard to the two reactors being compared, inside the core there is a control rod (CR) guide box housing the control rod plates. As the fuel assemblies and the control rod guide box are connected to the same inlet and outlet plenum, they should have the same total pressure drop between the plenums. As the coolant velocity differs approximately by a factor of 2 between both designs, the pressure drop differs by a factor of 4.

This different load requires, from the thermal-hydraulic point of view, a cautious analysis of the pressure drop distribution inside the control rod guide box in order to avoid:

— Excessive core coolant bypass;
— Excessive upward dragging force;
— Excessive deformation in the guide box.

The elastic deformation of the guide box, caused by the static and dynamic pressure difference, is conservatively considered adopting a design with a positive higher pressure inside the box in order to prevent a channel reduction.

To fulfil this design criterion, an important fraction of the total pressure drop inside the control rod guide box must be lost at the exit. This is obtained through calibrated restrictions at the exit of the control guide box.

Given the importance in the safety of the plant, the distribution of pressure drops in the RRR core was verified by experimental tests in a special mock-up shown in Fig. 5.

Figure 6 shows the comparison between theoretical and experimental results for partial and total ΔP inside the guide box.

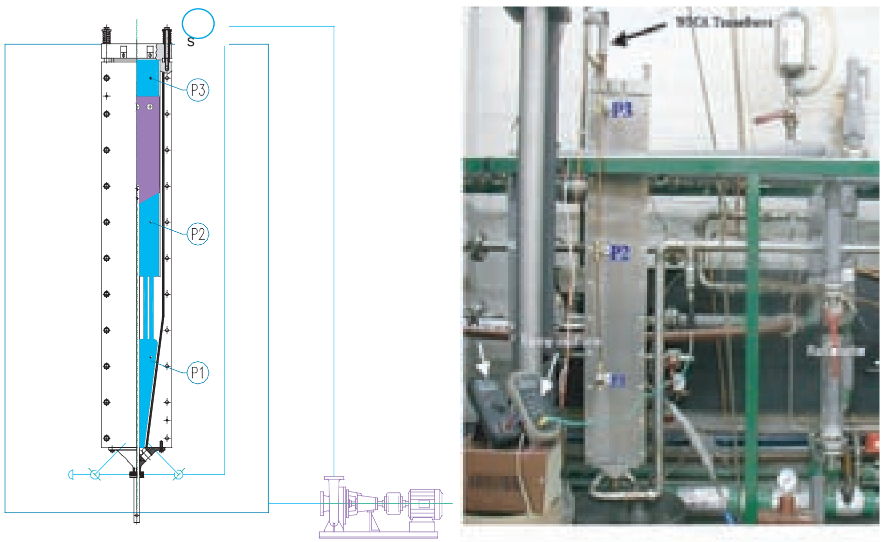


FIG. 5. Scheme and picture of the control rod guide box mock-up.

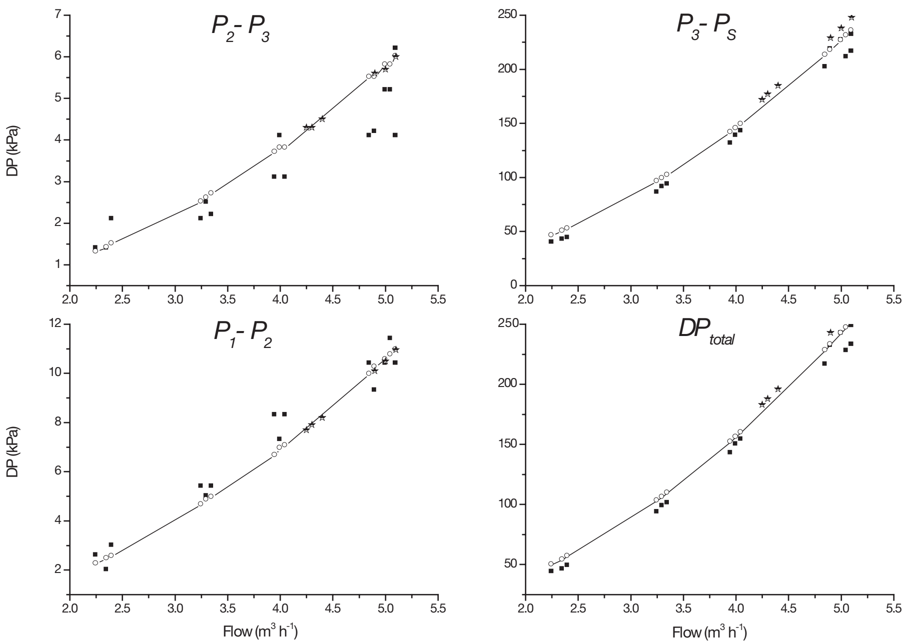


FIG. 6. Comparison of experimental and theoretical results for partial and total ΔP.

5. VERIFICATIONS USING MODERN ANALYTICAL TOOLS

5.1. Closure flow

As mentioned previously, a downward closure flow is specified inside the chimney to avoid hot water coming from the core. As it is not essential for the core cooling, its value should be optimized to fulfil the function.

This downward flow produces a mixed convection flow with inverse stratification in the upper region of the core. A hot liquid stream leaving the core is 'pulled up' by the suction branch, resulting in a hot/cold interface near the top of the connection between the chimney and the suction branch. This interface, inversely stratified, is unstable by nature. This instability causes large-scale turbulent vortices of hot liquid that detach and rise. For design purposes, the question is to define a minimum closure flow able to reverse the direction of movement of these hot liquid patches.

The efficiency of this hydraulic plug depends, among other parameters, on the ratio between the coolant velocity of flow coming from the core and the downward velocity so that a flow representing a percentage of the core flow was specified.

As it is definitively a three-dimensional thermal-hydraulic problem, the analysis was performed using a CFD code with the full set of Navier-Stokes equations. The applications of these types of engineering tools during the design stage can be considered as an important innovation in this field and of great assistance to the designer [3]. Figure 7 shows some results of the flow pattern structure for the limit flow.

5.2. Reflector vessel/decay tanks

Neutron irradiation of the heavy water and structures inside the reflector vessel system result in heat deposition. This heat must be removed by the heavy water to avoid the existence of regions of high temperature.

Unfortunately, since the heat source exhibits strong radial gradients, the flow is prone to buoyancy-generated vortical structures. These vortical structures enhance mixing, so that in the limit of perfect mixing, practically all of the heavy water volume would be at uniform temperature. Also, as buoyancy-driven flows tend to be unstable, intermittent flows, and undesirable fluctuations, may appear in the thermal neutronic flux.

On the other hand, since the reflector system does not have a decay tank, it is important to quantify the specific activity of ^{16}N in order to verify the shielding in pipes, pumps and heat exchangers of this system.

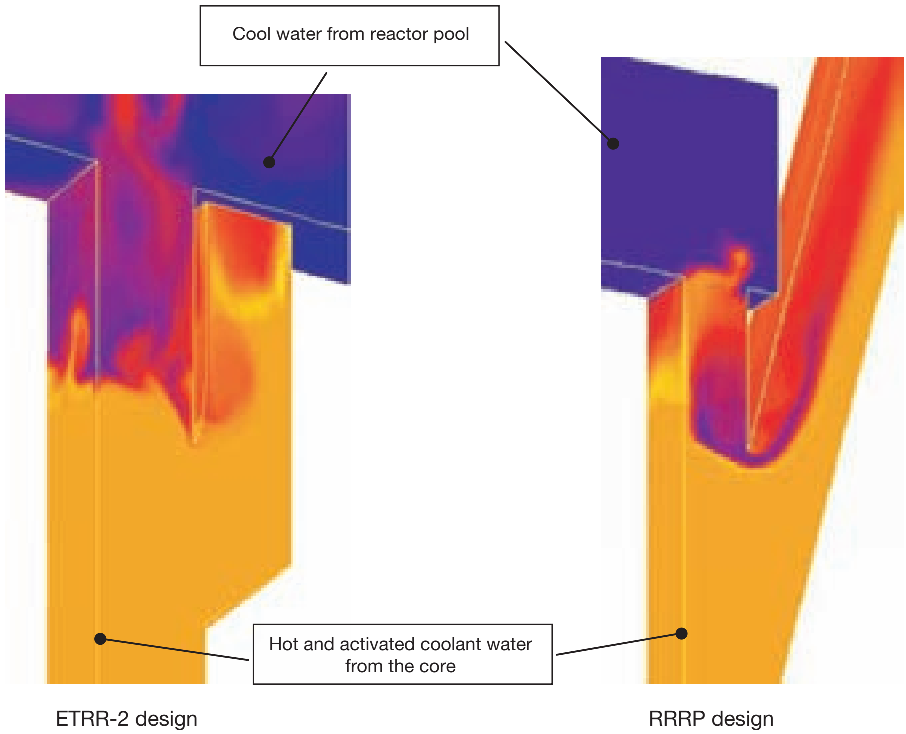


FIG. 7. Closure flow effectiveness of both designs.

Given the complex flow structure of the heavy water and the necessity to account for a global understanding of the flow fields, the system was modelled using a CFD code during the design stage. To solve the problem, the activation equations were modelled together with the Navier-Stokes equations coupled with the energy equation.

The reflector vessel has a cylindrical shape with 2.6 m of outer diameter and a quite complex internal structure, so the modelling of the whole system involved a particular effort. Figure 8 shows the computational finite element model and Fig. 9, the general results of the flow pattern (velocity and temperatures) inside the vessel.

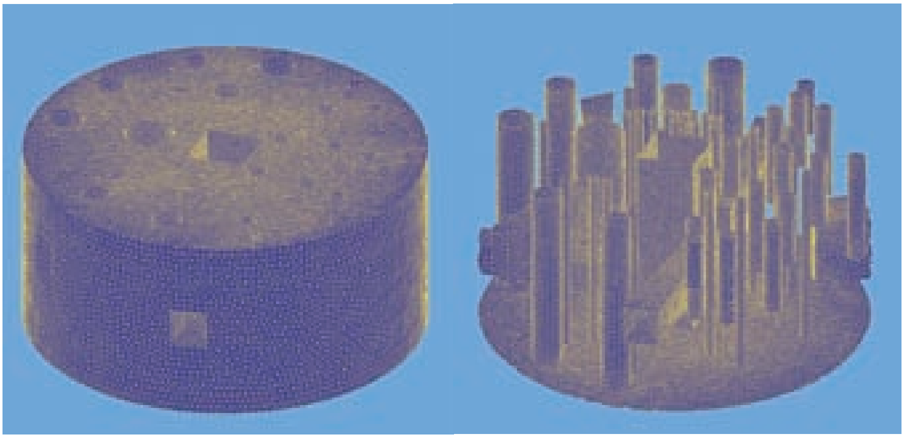


FIG. 8. Computational model and surface mesh of the reflector vessel.

6. SUMMARY AND CONCLUSIONS

A comparison between two particular 'similar' designs and the work performed to achieve different requirements is presented. Both experimental tests and complex analytical tools were used for that purpose.

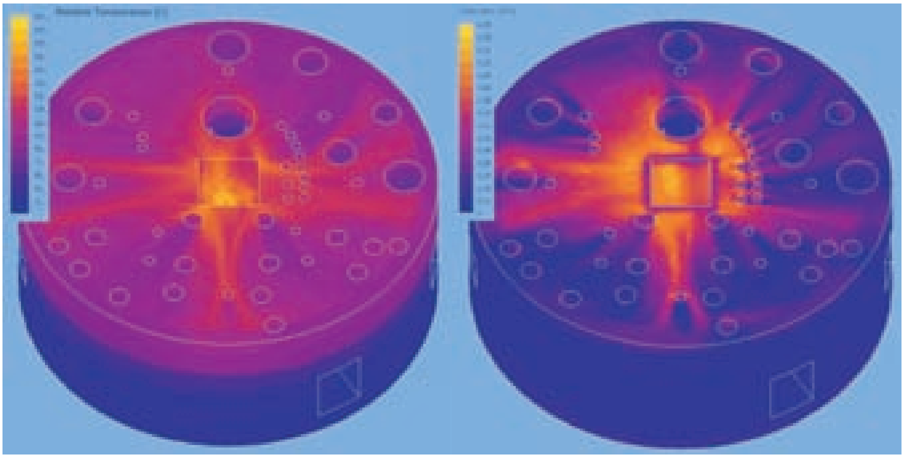


FIG. 9. Results of a computational model of temperature (relative to inlet value) and velocity field.

Phenomena such as hydraulic loads, stratified flows or thermal plumes are very common in this type of reactor and are strongly dependent on the component design. Those phenomena require careful assessment, as they cannot be analysed using simple one-dimensional tools.

Experience shows that even using a proven technology, a significant effort should be made to fulfil new demands, preserving the safety standards in a new design or in a significant upgrade.

REFERENCES

[1] ATOMIC ENERGY AUTHORITY EGYPT, Safety analysis report of ETRR-2, Rep. 0767–5325–3IBLI–001–10, AEAE, Cairo.

[2] AUSTRALIAN NUCLEAR SCIENCE AND TECHNOLOGY ORGANISATION, Summary of the preliminary safety analysis report (PSAR) for the ANSTO Replacement Research Reactor Facility, ANSTO, Sydney (2001).

[3] BUSCAGLIA, G., et al., "Computational fluid dynamics throughout the design process in nuclear applications", in Mecanica Computacional series, vol. XXI.

BLANK

TRANSIENT THERMAL-HYDRAULIC MODELLING OF ETRR-2 FOR OFF-SITE POWER LOSS

A.H. MARIY, S.E. EL-MORSHDY
Reactors Department,
Atomic Energy Authority,
Cairo
Email: slmorshdy@etrrz-aea.org.eg

H.A. KHATER, A.A. ABDEL-RAOUF
Department of Mechanical Engineering,
Cairo University, Giza

Egypt

Abstract

A transient thermal-hydraulic model entitled ETRR-2 Transient Analysis (ETRR2TA) has been developed to simulate the steady state operation and transient behaviour of ETRR-2 core after loss of off-site power. The model consists of five interactively coupled sub-models for (a) coolant; (b) clad; (c) fuel; (d) chimney; and (e) natural circulation flow. The model divides the active core into specified axial regions and the fuel plate into specified radial zones, then a nodal calculation is performed for both average and hot channels with a chopped cosine shaped heat generation flux. The model also predicts the onset of nucleate boiling, onset of flow instability and critical heat flux phenomena. The model is verified by PARET code for the ETRR-2 core with upward flow direction. It is also validated by a benchmark problem for a 10 MW generic reactor with downward flow direction. The comparisons show good agreement. The model results for ETRR-2 is analysed and discussed.

1. INTRODUCTION

Egypt's second research reactor (ETRR-2) is a pool-type reactor with an open water surface and variable core arrangement. The core power is 22 MW, cooled by light water, moderated by water and with beryllium reflectors. It has plate-type fuel elements (MTR type, 19.7% enriched uranium) with aluminium clad. The reactor protection system commands two diverse and independent reactor shutdown systems that extinguish the nuclear reaction. After reactor shutdown, decay heat is removed by a natural circulation mechanism through flap valves. The reactor core is placed in a 4.5 m diameter atmospheric stainless steel tank at a water depth of 10 m. It is built on a supporting grid having 6 by 5 positions available for placing fuel or irradiation boxes. A prismatic structure, known as the lower chimney, is mounted on the core grid, forming with its walls a 438 mm by 489 mm vertical duct around the fuel elements. The upper chimney is a rectangular section duct that prolongs the inner sections of the lower chimney; it has a height of 3557 mm above the core level. It has a lateral derivation of 438 mm wide and 1000 mm high inner opening that allows the passage of water from the core circuit, towards the exit. In case of a total loss of electric power supply, the flow through the core will decrease, depending on the inertia flywheels and primary system characteristics. Once the flow is low enough, the two flap valves placed on the two 12 in. pipes of the cooling system open and a natural circulation mechanism is established through the core where the coolant will be driven by the density difference between the cold leg (primary piping) and the hot leg (core and chimney). The objective of the present paper is to develop a compact computer program that accurately simulates the transient and steady-state thermal behaviour of the ETRR-2 reactor core.

2. MODEL DESCRIPTION

A simplified flow scheme of the model is presented in Fig. 1, where the coolant channel is divided into specified axial regions while the fuel plate is

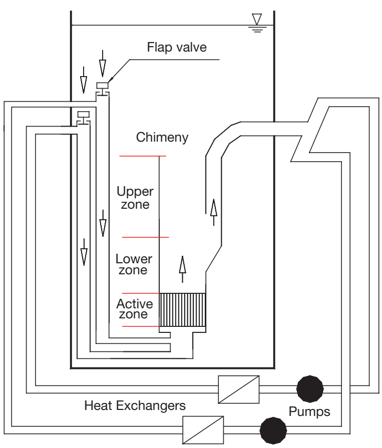


FIG. 1. Model scheme.

divided into specified radial nodes, then a nodal thermal-hydraulic calculation for both average and hot channels is performed with a chopped cosine heat generation flux. The chimney is divided into two zones, the lower zone and the upper zone where the bulk temperatures in both zones are evaluated during the transient state in order to estimate the real buoyancy force for natural circulation. The chimney lower zone is bounded by the core exit and the mean level of the chimney lateral opening, while the upper zone is above the lower zone up to the upper edge of the chimney.

2.1. Coolant model

The core coolant model treats the coolant as one lumped node. It thus assumed that the coolant is well stirred and has a uniform temperature. The general energy balance equation is applied to the core coolant nodes:

$$\rho A \Delta z \frac{I_m^{p+1} - I_m^p}{\Delta \tau} = 2\phi * W_h \Delta z - GA\left(I_i^{p+1} - I_{i-1}^{p+1}\right) \tag{1}$$

where $I_m^p = \dfrac{I_i^p + I_{i-1}^p}{2}$ and $I_m^{p+1} = \dfrac{I_i^{p+1} + I_{i-1}^{p+1}}{2}$

2.2. Fuel plate model

Based on the finite difference technique, the model treats the conduction in the mesh nodes of a half fuel plate in one dimension implicit scheme in the following manner:

2.2.1. *Fuel zone*

$$T_{i,j+1}^{p+1} + T_{i,j-1}^{p+1} - \left(2 + \frac{(\Delta x_f)^2}{\alpha_f \Delta \tau}\right) T_{i,j}^{p+1} + \frac{(\Delta x_f)^2}{\alpha_f \Delta \tau} T_{i,j}^p + \frac{q_i^{p+1} (\Delta x_f)^2}{K_f} = 0 \tag{2}$$

2.2.2. *Fuel centre*

$$T_{i,j+1}^{p+1} - \left(1 + \frac{(\Delta x_f)^2}{2\alpha_f \Delta \tau}\right) T_{i,j}^{p+1} + \frac{(\Delta x_f)^2}{2\alpha_f \Delta \tau} T_{i,j}^p + \frac{q_i^{p+1} (\Delta x_f)^2}{2K_f} = 0 \tag{3}$$

2.2.3. Fuel-clad interface

$$\frac{K_c}{K_f}\frac{\Delta x_f}{\Delta x_c}T_{i,j+1}^{p+1}+T_{i,j-1}^{p+1}-\left(1+\frac{K_c}{K_f}\frac{\Delta x_f}{\Delta x_c}+\frac{(\Delta x_f)^2}{2\alpha_f\Delta\tau}+\frac{\Delta x_f\Delta x_c}{2\alpha_f\Delta\tau}\right)T_{i,j}^{p+1}$$
$$+\frac{\Delta x_f}{2\Delta\tau}\left(\frac{\Delta x_f}{\alpha_f}+\frac{\Delta x_c}{\alpha_c}\right)T_{i,j}^{p}+\frac{q_i^{p+1}(\Delta x_f)^2}{2K_f}=0 \tag{4}$$

2.2.4. Clad zone

$$T_{i,j+1}^{p+1}+T_{i,j-1}^{p+1}-\left(2+\frac{(\Delta x_c)^2}{\alpha_c\Delta\tau}\right)T_{i,j}^{p+1}+\frac{(\Delta x_c)^2}{\alpha_c\Delta\tau}T_{i,j}^{p}=0 \tag{5}$$

2.2.5. Clad coolant interface

$$T_{i,j-1}^{p+1}-\left(1+\frac{h\Delta x_c}{K_c}+\frac{(\Delta x_c)^2}{\alpha_c\Delta\tau}\right)T_{i,j}^{p+1}+\frac{h\Delta x_c}{K_c}T_{co\,i}^{\;p+1}+\frac{\Delta x_c}{\alpha_c\Delta\tau}T_{i,j}^{p}=0 \tag{6}$$

2.3. Chimney model

The energy equation is applied to both the lower and upper zones of the chimney:

$$\rho V\frac{\partial I}{\partial\tau}=G\,A_{po}\left(I_{in}-I_{out}\right) \tag{7}$$

2.4. Natural circulation model

After flap valve opening, the core flow rate is estimated by the momentum equation for a defined control volume for coolant in core channels and pipes up to flap valves as follows:

$$\sum_{i=1}^{n}\rho_i L_i\frac{A}{A_i}\frac{du}{dt}=\oint\left(\rho_o-\rho_i\right)g\,dz-\sum_{i=1}^{n}\Delta P_{f_i} \tag{8}$$

where ΔP_{f_i} is the frictional pressure drop along section 'i'.

2.5. Critical phenomena

2.5.1. Onset of nucleate boiling

The condition under which boiling will be initiated in the coolant channel is when the clad temperature equals to or exceeds the onset of nucleate boiling temperature, T_{ONB}, where:

$$T_{ONB} = T_{sat} + (\Delta T_{sat})_{ONB} \tag{9}$$

where $(\Delta T_{sat})_{ONB}$ is given by Bergles and Rohsenow correlation [1], which is valid for water only over the pressure range 1–138 bar:

$$\left(\Delta T_{sat}\right)_{ONB} = 0.556\left\{\frac{\phi_{ONB}}{1082P^{1.156}}\right\}^{0.463P^{0.0234}} \tag{10}$$

where P is the local pressure in bar and ϕ_{ONB} is in W/m^2.

2.5.2. Onset of flow instability

The average heat flux at the onset of flow instability is calculated using the correlation developed by Whittle and Forgan [2].

$$\phi_{OFI}(W/cm^2) = 0.05*\left[RCpG\frac{WD}{W_hL}\left(T_{sat} - T_{in}\right)\right] \tag{11}$$

where $R = \dfrac{1}{1+\eta\dfrac{D_e}{L}}$ and η is the bubble detachment parameter = 25.

2.5.3. Critical heat flux

In order to calculate the critical heat flux, the two flow models developed by Sudo [3] for rectangular channels are adapted. The models are based on the separated flow model with macroliquid sub-layer with over 800 data of previous experiments at 0.1–13.8 MPa with inlet water subcooling of 0–328 K. The first model is for critical heat flux in the upward flow under a relatively low mass flux and saturated two phase flow condition at the outlet of the channel is represented by:

$$\phi_{CHF}^{*3}\left\{1-\frac{1}{230}\left(\frac{G^*}{\phi_{CHF}^*}\right)^{0.5}\right\}^{\frac{8}{3}}=0.0033\left(1+\frac{\rho_g}{\rho_l}\right)^{\frac{4}{3}}\left(\frac{\rho_g}{\rho_l}\right)^{0.645}\left(\frac{D\lambda}{2D_e^2}\right)$$
$$*\left\{\left(1+\Delta T_{in}^*\right)G^*-\left(\frac{LW_h}{DW}\right)\phi_{CHF}^*\right\} \tag{12}$$

where $\phi_{CHF}^* = \dfrac{\phi_{CHF}}{I_{fg}\left[\lambda\left(\rho_l-\rho_g\right)\rho_g g\right]^{1/2}}$

$$G^* = \frac{G}{\left[\lambda\left(\rho_l-\rho_g\right)\rho_g g\right]^{1/2}}$$

$$\Delta T^* = \frac{Cp_l\left(T_{sat}-T_{in}\right)}{I_{fg}} \text{ and}$$

$$\lambda = \left\{\frac{\sigma}{\left(\rho_l-\rho_g\right)g}\right\}^{1/2}$$

The second model is for critical heat flux in the upward flow under the high mass flux and high subcooled condition at the outlet of the channel is represented by:

$$\phi_{CHF}^{*0.1887}\left\{1-\frac{1}{230}\left(\frac{G^*}{\phi_{CHF}^*}\right)^{0.5}\right\}\left\{\phi_{CHF}^*-F_1G^{*0.8}\left(\Delta T_{in}^*-\frac{A_H.\phi_{CHF}^*}{A.G^*}\right)\right\}$$
$$=0.4598F_2^{0.1887}.G^{*0.7677} \tag{13}$$

where

$$F_1=0.023\left(\frac{K_l\,\mathrm{Pr}_l^{0.4}}{D_e\rho_lCp_l}\right)\left\{D_e\sqrt{\lambda\left(1-\frac{\rho_g}{\rho_l}\right)\frac{\rho_g}{\rho_l}g\Big/\nu_l}\right\}^{0.8}\Big/\sqrt{\lambda\left(1-\frac{\rho_g}{\rho_l}\right)\frac{\rho_g}{\rho_l}g/\nu_l}$$

$$F_2 = \left\{ A_0 A_1 \left(\frac{\sigma}{\rho_l \nu_l} \right) \left(\frac{\rho_g}{\rho_l} \right)^{1.4} \cdot \left(1 + \frac{\rho_g}{\rho_l} \right) \Big/ \sqrt{\lambda \left(1 - \frac{\rho_g}{\rho_l} \right) \frac{\rho_g}{\rho_l} g} \right\}^{2.65}$$

$$* \left\{ \rho_l \nu_l \sqrt{\lambda \left(1 - \frac{\rho_g}{\rho_l} \right) \frac{\rho_g}{\rho_l} g} \Big/ 4\pi\sigma \right\} \left\{ \frac{D_e}{\nu_l} \sqrt{\lambda \left(1 - \frac{\rho_g}{\rho_l} \right) \frac{\rho_g}{\rho_l} g} \right\}^{-0.581}$$

$$A_0 = 0.00536 \text{ and } A_1 = 0.066 \left(\frac{\rho_g}{\rho_l} \right)^{-0.4}$$

2.6. Coefficient of heat transfer

The model estimates the heat transfer coefficient for both the single phase and the boiling two phase flow, it first defines the flow regime at each node and then performs calculations to obtain the heat transfer coefficient as shown in the following sections.

2.6.1. Single phase forced convection

(a) Turbulent regime Re ≥ 1000; Dittus-Boelter [4] equation is used:

$$Nu = 0.023 \mathrm{Re}^{0.8} \mathrm{Pr}^{0.4} \tag{14}$$

(b) Transition regime 2100 < Re < 10000; Nusselt number is calculated by interpolation between the laminar and turbulent correlations.

(c) Forced laminar regime Re ≥ 2100; Sieder and Tate [5] correlation is used:

$$Nu = 1.86 \left(\frac{\mathrm{Re}\,\mathrm{Pr}}{L / D_e} \right)^{\frac{1}{3}} \left(\frac{\mu_{co}}{\mu_c} \right)^{0.14} \tag{15}$$

2.6.2. Single phase combined forced and free convection

The following empirical method [6] for estimating the Nusselt number for combined, forced and natural convection is used:

$$Nu^n_{combined} = Nu^n_{forced} \pm Nu^n_{natural} \tag{16}$$

where n = 3 for vertical plates and the + sign applies when the flows are in the same direction and the – sign when they are in opposite direction.

2.6.3. Free convection

The correlations provided by Churchill and Chu [7] for vertical planes are used as follows:

$$Nu = 0.68 + \frac{0.67 Ra^{1/4}}{\left[1+(0.492/\mathrm{Pr})^{9/16}\right]^{4/9}} \quad \text{for} \quad Ra < 10^9 \tag{17}$$

$$Nu^{1/2} = 0.825 + \frac{0.387 Ra^{1/6}}{\left[1+(0.492/\mathrm{Pr})^{9/16}\right]^{8/27}} \quad \text{for} \quad 10^9 < Ra < 10^{12} \tag{18}$$

2.6.4. Subcooled boiling

Boiling is initiated when the clad temperature is equal to the onset of nucleate boiling temperature given by equation (9). The correlation developed by Chen [8] for saturated boiling, as described subsequently, is extended for use in the subcooled boiling. It is assumed that the total heat flux is made up of a nucleate boiling contribution and a single phase forced convective contribution:

$$\phi(z) = h_{NCB}\left(T_c(z) - T_{sat}(z)\right) + h_{SP}\left(T_c(z) - T_{co}(z)\right) \tag{19}$$

2.6.5. Saturated nucleate boiling

The heat transfer coefficient in forced flow saturated boiling is estimated by the correlation proposed by Chen [8]:

$$h_{TP} = h_{SP} + h_{NCB} \tag{20}$$

where $h_{SP} = 0.023\left[\frac{G(1-x)D_e}{\mu_l}\right]^{0.8}\left[\frac{\mu Cp}{k}\right]_l^{0.4}\left(\frac{k_l}{D_e}\right)(F)$, F is the convective boiling enhancement factor

and $h_{NCB} = 0.00122\left[\frac{k_l^{0.79} Cp_l^{0.45} \rho_l^{0.49}}{\sigma^{0.5} \mu_l^{0.29} I_{fg}^{0.24} \rho_g^{0.24}}\right]\Delta T_{sat}^{0.24} \Delta P_{sat}^{0.75}(S)$, S is the nucleate boiling suppression factor.

2.6.6. Film boiling

If the axial heat flux reaches the critical heat flux, the model calculates the local heat transfer coefficient $h(z_{FB})$ at a distance z_{FB} up the surface from the start of film boiling by the following correlation [1]:

$$\frac{h(z_{FB})z_{FB}}{k_g} = \left[\frac{z_{FB}^3 g\rho_g(\rho_l - \rho_g)I'_{fg}}{k_g \mu_g \Delta T}\right]^{0.25} \tag{21}$$

where $I'_{fg} = \left[1 + 0.68\frac{Cp_g \Delta T}{I'_{fg}}\right]$ and $\Delta T = (T_c - T_{co})$

2.7. Pressure drop

2.7.1. Pressure drop in single phase liquid and subcooled boiling

The pressure drop between two successive axial nodes in single phase liquid or subcooled boiling regimes is given by:

$$\Delta P_z(i) = \frac{2F_{fo}G^2 v_l dz}{D_e} + (v_l - v_{li})G^2 \mp \frac{gdz}{v_l} \tag{22}$$

where F_{fo} is the Darcy friction factor for single phase liquid. It is calculated for rectangular channels as:

for laminar flow [9] $F_{fo} = 12\Big/\frac{GD}{\mu_l}$ (23)

for turbulent flow [9] $\frac{1}{F_{fo}^{1/2}} = 2.0\log\left(\mathrm{Re}\,F_{fo}^{1/2}\right) - 1.19$ (24)

2.7.2. Pressure drop in the saturated boiling

Based on the Martinelli-Nelson correlation [1], the separated flow model is applied in order to evaluate the void fraction at each node and then to calculate pressure drop between two successive axial nodes:

$$\Delta P_z(i) = A\left(\frac{2F_{fo}G^2 v_l dz}{D_e}\right) + r_2 G^2 v_l \mp \frac{Gdz}{x}\int_0^x \left(\rho_g \alpha + \rho_l (1-\alpha)\right) dx \tag{25}$$

where

$A = \dfrac{1}{x}\displaystyle\int_0^x \phi_{fo}^2 dx$, ϕ_{fo}^2, is the two phase frictional multiplier and

$$r_2 = \frac{x^2}{\alpha}\left(\frac{v_g}{v_l}\right) + \frac{(1-x)^2}{(1-\alpha)} - 1$$

3. MODEL VALIDATION

The present model is verified by two methods: the first one is by comparing the model results with the PARET code [10] under the same conditions for the ETRR-2. As the PARET code simulates the core only, the present model is reduced to core calculation without considering the effect of the chimney, piping or flap valves. Figure 2 shows a good agreement between PARET and ETRR2TA for core mass flux and maximum coolant, clad surface and fuel centre temperatures in the hot channel of ETRR-2 under loss of power conditions.

The second validation method compares the model results with a safety related benchmark problem [11]. The problem is solved by the present model and its results are compared with the benchmark calculations. The calculations are performed for a 10 MW generic reactor with downward flow direction and the comparison also shows good agreement, as shown in Fig. 3.

4. RESULTS AND DISCUSSION

The present model is used to simulate ETRR-2 behaviour under loss of off-site power for a typical core configuration of 29 fuel elements. The simulation is performed under two cases, namely, (1) two flap valves open; and (2) one flap valve fails to open. The simulation is made with a total power peaking factor of 3 and pool temperature of 40º C without engineering uncertainties. It is assumed that the reactor is operated at its nominal power, 22 MW before loss of electric power and the decay heat is calculated from the ANS table [12] of normalized power versus time. The pump coast-down for the ETRR-2 core cooling system was estimated by taking actual data from a test of

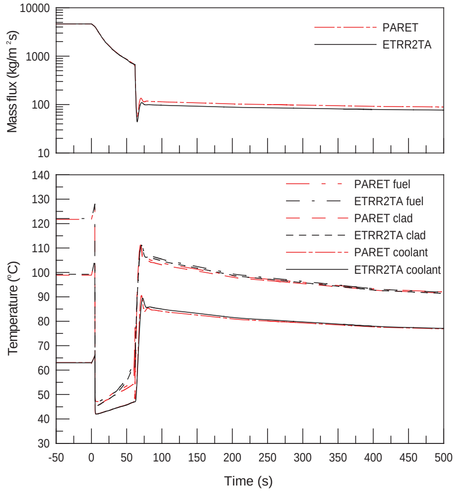


FIG. 2. Comparison with PARET for ETRR-2 core.

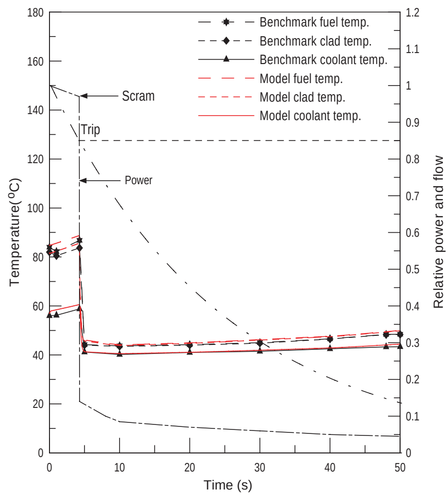


FIG. 3. Comparison with benchmark problem.

11 July 2000 on pumps 1/2 and 2/2 together. A summary of the sequence of events after the loss of power supply is as follows:

(1) Flow starts to coast down due to pump inertia flywheels;
(2) Reactor begins to scram after 3.9 s due to low flow signal at 90% of the nominal flow;

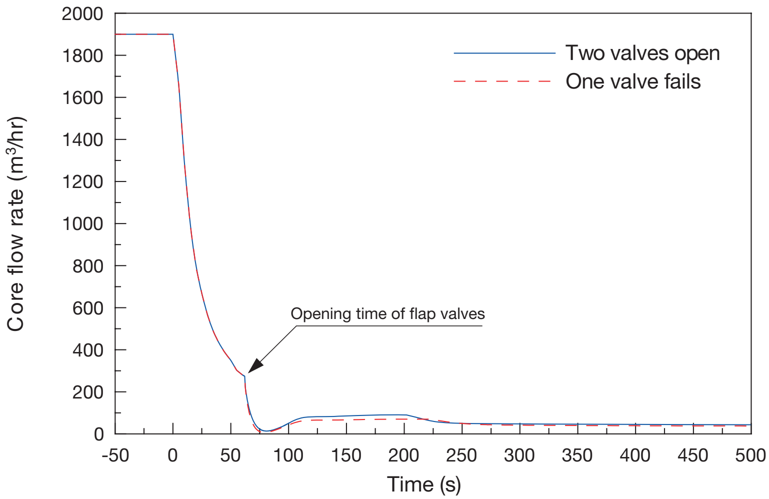


FIG. 4. Coast-down for pumps 1/2 and 2/2.

(3) Scram is affected after 0.2 s delay between reaching the scram point and triggering of control plates and 0.7 s for control plates 85% insertion;

(4) Flap valves open when the pressure force on the flap valve becomes lower than its weight;

(5) Natural circulation takes place.

Figure 4 shows the coast-down curve for the two cases. When flap valves open, part of the flow coast-down due to inertia flywheels is diverted to the pool through flap valves. At this time, the coolant temperature in the cold leg is higher than the pool temperature and though a buoyant force is initiated in the cold leg in the direction opposing the flow. Therefore, the core flow rate decreases rapidly to reach its minimum value after loss of power by about 80 s. This leads the coolant temperature to increase and so the buoyant force in the core increases, leading to an enhancement in the core flow rate. In the natural circulation region, the coolant flow rate in the case of one flap valve failing to open is slightly below the flow in the case of two flap valves opening where, in this case, the flow is passing through only one branch of the returning line to the core and so the friction losses increase.

After scram the reactor core decreased sharply, leading the core temperatures to decrease sharply near the inlet temperature as shown in Fig. 5 for the hot channel. Then the temperature increases gradually as the coolant flow rate decreases. About 30 s after scram, the hot water in the core cooling system

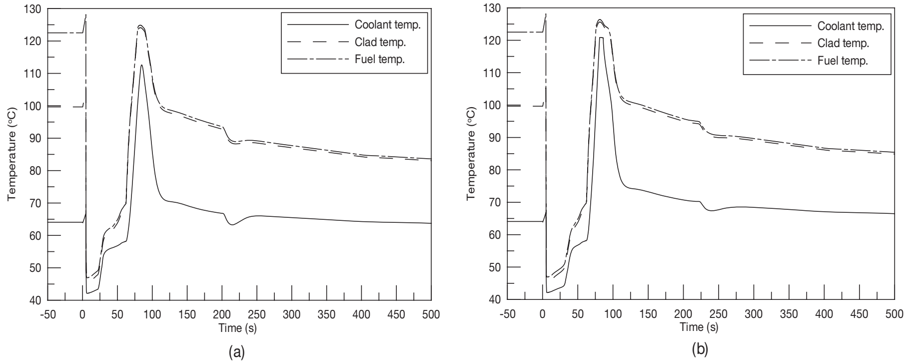


FIG. 5. Hot channel maximum temperatures (a) two flap valves open, and (b) one flap valve fails.

piping between the chimney and the heat exchangers enters the lower plenum. This leads to a temperature rise equal to the core temperature difference before transient (≅10º C). After 62 s, the flap valves open and so part of the pump coast-down flow is diverted to the pool and so the core flow is reduced. This leads the temperatures to increase sharply resulting in maximum temperature values in all curves after the flap valves open by about 20 s. As a result, the buoyancy force increases, which in turn enhances the coolant flow rate and so the coolant temperature decreases rapidly. Then the natural circulation mechanism is established and the coolant temperature decreases gradually as the decay heat decreases. A reduction in the coolant temperatures occurs at about 200 s for case one and 220 s for case two, when the cold water diverted from the pool through flap valves enters the core. For case one, the clad temperature exceeds the onset of nucleate boiling temperature and so sub-cooled boiling is initiated for a period of about 11 s. While for case two, boiling is predicted for about 19 s, 6 s of which is bulk boiling.

Figure 6 shows the critical onset of flow instability and maximum axial heat fluxes in the hot channel, as well as the onset of nucleate boiling heat flux at the position of maximum clad temperature. It shows that the maximum axial heat flux exceeds the onset of nucleate boiling heat flux for periods of 20 s for case one and 26 s for case two. However, the actual boiling periods presented by Fig. 5 are shorter where the maximum axial heat flux during the transient state is not necessarily at the position of maximum clad temperature. It is also found that the phenomenon of flow instability is predicted for periods of 9 s for case one and 14 s for case two.

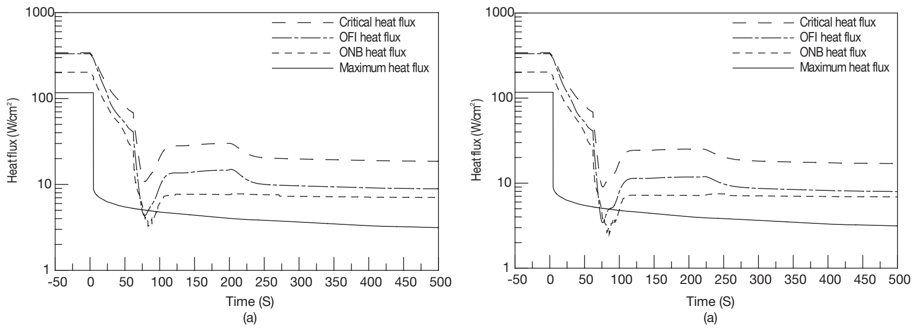


FIG. 6. Hot channel heat flux (a) two flap valves open and (b) one flap valve fails.

5. CONCLUSION

A computer program to simulate the ETRR-2 transient and steady state thermal-hydraulic behaviour has been developed. The program has the ability to simulate the two flap valves opening, or one opening and the other failing. The model is verified by both the PARET code for the ETRR-2 core and a benchmark problem for the 10 MW generic reactor. The model results proved that, even under the condition of no engineering uncertainties being accounted for, the pump coast-down is not enough to remove the decay heat in the core following full power (22 MW) operation without boiling, and bulk boiling is predicted if one flap valve fails to open.

Nomenclature

A	channel flow area, m^2
A_{po}	outlet plenum area, m^2
A_s	surface area, m^2
Cp	specific heat at constant pressure, J/kgK
D	channel gap thickness, m
D_e	equivalent hydraulic diameter, m
F_{fo}	friction factor assuming the total flow liquid
g	acceleration of gravity, m/s^2
G	mass flux, kg/m^{2}s
Gr	Grashof number = $g\beta(T_c - T_{co})L^3/v^2$
h	heat transfer coefficient, W/m^2°C
I	enthalpy, J/kg
I_{fg}	latent heat of evaporation, J/kg
k	thermal conductivity, W/m°C
L	active length, m
Nu	Nusselt number = hD_e/k
P	pressure, Pa
Pr	Pradentl number = $\mu Cp/k$
Re	Reynolds number = GD_e/μ
Ra	Rayleigh number = Gr^*/Pr
q	volumetric heat generation, W/m^3
T	temperature, °C
u	coolant velocity, m/s
v	specific volume, m^3/kg
W	channel width, m
W_h	active width, m
x	distance in radial direction, m
x	steam quality
z	distance in axial direction, m

Greek symbols

α	thermal diffusivity, m^2/s
α	void fraction
ρ	density, kg/m^3
ρ_o	pool density, kg/m^3
μ	dynamic viscosity, kg/ms
σ	surface tension, N/m
Φ	surface heat flux, W/m^2
τ	time, s
τ_w	wall shear stress, N/m^2

Superscripts

p	time step index
*	means dimensionless quantity

Subscripts

c	clad
CHF	critical heat flux
co	coolant
f	fuel
fg	difference of saturated liquid and vapour
g	vapour phase
i	location indicating z direction
in	inlet
j	location indicating x direction
l	liquid phase
NCB	nucleate boiling contribution
ONB	onset of nucleate boiling
OFI	onset of flow instability
out	outlet
sat	saturated
SP	single-phase
TP	two-phase

REFERENCES

[1] COLLIER, J.G., Convective Boiling and Condensation, 2nd edn, McGraw-Hill, New York (1981).

[2] WHITTLE, R.H., FORGAN, R., A correlation for the minima in the pressure drop versus flow rate curves for sub-cooled water flowing in narrow heated channels, Nucl. Eng. Design **6** (1967).

[3] SUDO, Y., Study on critical heat flux in rectangular channels heated from one or both sides at pressures ranging from 0.1 to 14 Mpa, ASME J. Heat Transfer **118** (Aug. 1996) 680–688.

[4] DITTUS, F.W., BOELTER, L.M.K., University of California, Berkeley, Publications on Engineering **2** (1930) 443.

[5] SIEDER, E.N., TATE, G.E., Ind. Eng. Chem. **28** (1936) 1429.

[6] KREITH, F., BOHN, M.S., Principles of Heat Transfer, 5th edn, West Publishing Company (1993).

[7] CHURCHILL, S.W., CHU, H.H.S., Correlating equations for laminar and turbulent free convection from a vertical plate, Int. J. Heat Mass Transfer **18** (1975) 1323.

[8] CHEN, J.C., A correlation for boiling heat transfer to saturated fluids in convective flow, Ind. Eng. Chem. Process Design Dev. **5** 3 (1966) 322–329.

[9] WHITE, F.M., Viscous Fluid Flow, McGraw-Hill, New York (1974, © 1991).

[10] OBENCHIAN, C.F., PARET – A Program for the Analysis of Reactor Transients, Rep. IDO-17282 (1969).

[11] MATOS, J.E., PENNINGTON, E.M., FREESE, K.E., WOODRUFF, W.L., Safety-Related Benchmark Calculations for MTR-Type Reactors with HEU, MEU and LEU Fuels, IAEA-TECDOC-643, Vol. 3, Appendix G, IAEA, Vienna (1992).

[12] AMERICAN NATIONAL STANDARDS INSTITUTE, ANS Standard Decay Energy Release Rates Following Shut-down of Uranium-Fuelled Thermal Reactors, ANSI, New York (1971).

REGULATORY CONTROL OF RESEARCH REACTORS AND CRITICAL ASSEMBLIES IN ARGENTINA

C.D. PERRIN
Nuclear Regulatory Authority (ARN),
Buenos Aires, Argentina
Email: cperrin@sede.arn.gov.ar

Abstract

The Nuclear Regulatory Authority (ARN) was established as an autonomous body reporting to the President of Argentina by Act 24,804 known as the Nuclear Activity National Act, which came into force on 25 April 1997, and is empowered to regulate and control the nuclear activity with regard to radiation and nuclear safety, physical protection and nuclear non-proliferation issues. It must also advise the Executive on issues under its purview. The objective of the ARN is to establish, develop and enforce a regulatory system applicable to all nuclear activities carried out in Argentina. In the field of radiological and nuclear safety control applied to research reactors and critical assemblies, ARN's regulatory activities are directed at controlling three research reactors and three critical assemblies; analysing design and operation related documents; permanently assessing safety during operation; and verifying by means of regulatory inspections and audits the compliance with the provisions of the licence concerned. The paper describes the radiological and nuclear safety regulatory control applied to research reactors and critical assemblies in Argentina: the organization, criteria, thematic, regulatory personnel qualifications, the annual plan and the standards.

1. INTRODUCTION

The Nuclear Regulatory Authority (ARN) of Argentina was established as an autonomous body reporting to the President of Argentina by Act 24,804 (known as the Nuclear Activity National Act), which came into force on 25 April 1997, and is empowered to regulate and control nuclear activity with regard to radiation and nuclear safety, physical protection and nuclear non-proliferation issues. It must also advise the Executive on issues under its purview. The objective of the ARN is to establish, develop and enforce a regulatory system applicable to all nuclear activities carried out in Argentina. The goals of this regulatory system are:

- To provide an appropriate standard of protection for individuals against the harmful effects of ionizing radiation.
- To maintain a reasonable degree of radiological and nuclear safety in the nuclear activities performed in Argentina.
- To ensure that nuclear activities are not developed with purposes unauthorized by the law and regulations resulting therefrom, as well as by the international agreements and the non-proliferation policies adopted by Argentina.
- To prevent the commission of intentional actions which may either have severe radiological consequences or lead to the unauthorized removal of nuclear materials or other materials or equipment subject to control.

With the purpose of fulfilling the objectives mentioned, the ARN has developed and has been provided with three basic capacities:

(1) Legal capacity: established by means of a law, with missions and functions clearly established and legally recognized.
(2) Technical capacity: staff suitably trained, with a high percentage of professionals, a majority of whom have postgraduate qualifications.
(3) Operational capacity: adequate infrastructure, equipment and budget for the fulfilment of its functions.

In Argentina, there are two nuclear power plants in operation, one under construction, three critical assemblies, three research and isotope production reactors, one decommissioned critical assembly, 25 major radioactive facilities and more than 1600 facilities for medical, industrial, research and training purposes which use radioactive materials or sources.

The type of regulatory tasks can be different in three main fields: radiological protection and nuclear safety, safeguards and physical protection. In the field of radiological and nuclear safety control, applied to research reactors (RR) and critical assemblies (CA), ARN's regulatory activities are directed at controlling three research reactors and three critical assemblies, analysing design and operation related documents, permanently assessing safety during operation, and verifying, by means of regulatory inspections and audits, compliance with the provisions of the licence concerned. The principal characteristics of the critical assemblies and research reactors in Argentina are presented in Tables 1 and 2.

TABLE 1. CRITICAL ASSEMBLIES IN ARGENTINA

	Critical assemblies			
	RA-0	RA-2	RA-4	RA-8
Power (W)	1	1	1	10
Type	Tank	Tank	Homogeneous	Tank
Utilization	Teaching and training	RA-3 facility	Teaching and training	Carem fuel test
Fuel	UO_2	UAl	UO_2	UO_2
Fuel element	Rods	MTR	Polyethylene plates	Rods
Enrichment (%)	20	90	20	1.8 and 3.4
Rectiv. excess	$0.40	—	$0.40	Not defined
Status	Operational	Decommissioned	Operational	Extended shutdown
Place	University—Córdoba	Constituyentes Atomic Centre	University—Rosario	Pilcaniyeu Atomic Centre
Criticality	1970		1971	1998

TABLE 2. RESEARCH REACTORS IN ARGENTINA

	Research reactors		
	RA-1	RA-3	RA-6
Power (KW)	40	10000	500
Type	Tank	Tank	Tank
Utilization	Research, training, BNCT, material test	Radiois. production, research, AxA	Research, training, AxA, BNCT
Fuel	UO_2	U_3O_8, U_3Si_2	UAl
Fuel element	Rods	MTR	MTR
Enrichment	20	20	90
Rectiv. excess	$1.50	$8	$2
Status	Operational	Operational	Operational
Place	Constituyentes Atomic Centre	Ezeiza Atomic Centre	Bariloche Atomic Centre
Criticality	1958	1967	1982

2. ORGANIZATION

Radiological and nuclear safety control is carried out in the nuclear reactors area, which is divided into three groups, one group each for each power plant and one for research reactors and critical assemblies. Each group is managed by a team leader. The research reactors and critical assemblies group is composed of three professionals with expertise in nuclear engineering and radiological and nuclear safety. This group is responsible for doing and/or coordinating all the tasks related to the research reactors and the research reactors, according to directives of the manager of the nuclear reactors area.

The research reactors and critical assemblies group is helped by administrative and technical personnel pertaining to the nuclear reactors area and other ARN areas in a matrix organization.

3. REGULATORY CONTROL CRITERIA

In order to reach the objectives previously expressed, inspection, audits and evaluations are planned and developed to a set of tasks that are organized to consider the following general criteria:

- *Prevention:* During regulatory control tasks, attention must be focused on processes of work and on the evolution of parameters in order to detect tendencies that could affect safety. As a result of those inspections and evaluations, when it corresponds, the regulatory body must use their enforcement with the purpose of avoiding the progress of negative tendencies or to correct deviations.
- *Prioritization:* The subjects included in inspections and audits must be planned, taking into account their safety function and the potential risks associated with work processes, as well as the tendencies and deviations detected in previous inspections and audits.
- *Processes evaluation:* Each topic to be controlled must be analysed, taking into account the three basic components:
  - Hardware, including equipment, systems, components and associated physical parameters.
  - Software, including documents and procedures.
  - Human factor, including qualification, skills and attitude towards safety.

In all activities, these three components are mixed with more or less weight, conforming to work processes. The regulator must evaluate each individual component, as well as the process in which components join, attending to their impact on safety.

— *Periodic review:* Based on experience (best plant processes knowledge, new standards, etc.), a periodic review of all the safety aspects involved becomes necessary.
— *Planning:* Annual general planning must be done, taking into account the coverage of all safety aspects, scheduling at least each aspect once per year. In addition, the planning must consider the emergency plan exercise. Thereafter, a more detailed plan must be prepared, including detailed topics, date, inspection teams, etc. Such plans must be sent to and coordinated with the reactor personnel responsible.

4. REGULATORY CONTROL THEMATIC

The subjects and items on which the routine control is focused arise from the standards, the mandatory documentation (especially SAR and OLC), and specific requirements made by the regulatory body. The subjects controlled are emergencies, personnel, nuclear safety, radiation protection, operation, maintenance and special issues. Some examples follow of some of the items controlled.

4.1. Emergencies

— Manual and procedures that include actions to follow in case of the occurrence of different kinds of accidents.
— Instruments, equipment and infrastructure established in the emergency plan.
— Whether personnel at the reactor is suitably trained and know his or her roles.
— Whether staff members of the external emergency organization are suitably trained and know their roles.
— Participation in the design and evaluation of annual emergency plan exercises.
— Whether the findings observed in past exercises are analysed and corrected.

4.2. Personnel

- The programmes and resources for qualification and training of personnel.
- Whether the operating personnel has a licence and specific authorization.
- The activities related to personnel under training and retraining.
- The occurrence of abnormalities and incidents due to human causes.
- Participation in examinations of licences, specific authorizations and retraining.

4.3. Nuclear safety

- Verification if reactor safety systems are operable and functioning.
- Whether the reactor safety functions are periodically tested and the results are documented.
- Whether the parameters affecting safety barriers are maintained under established limits.
- The impact on safety of modifications and new experiments.
- Core management.
- Periodic safety review.

4.4. Radiation protection

- Whether a suitable manual and procedures are available that cover all tasks with radiological risk, and if documents are updated.
- Whether all tasks are done according to the manual and the procedures.
- Whether there are enough and suitable radiological instruments and equipment for measurements and protection.
- Whether radiological instruments are checked and calibrated periodically.
- Whether the radiation protection programme includes an adequate system for records and documents, whether this information is updated and arranged. Whether, from analysing the information, some abnormal situation or negative tendency can be inferred.
- If there are facilities, equipment and elements for contamination measurement and for decontaminating objects and people.
- If radioactive sources are suitably documented, shelled and stored. If there is a procedure for management and essay of radioactive sources.
- If radioactive wastes are suitably managed.
- Whether all personnel who has tasks with radiological risk is qualified and trained.

— The fulfilment of limits and conditions related to occupational and public exposure.

4.5. Operation

— Whether a suitable operation manual and procedures exist that cover all tasks, and whether those documents are updated.
— If all tasks are done according to the operation manual and procedures.
— If there are adequate tools, equipment and elements needed for operational tasks.
— If there are is an adequate system for operational records and documents, if this information is updated and arranged.
— If, from analysing the information, some abnormal situation or negative tendency can be inferred.

4.6. Maintenance

— Whether a suitable maintenance manual and procedures exist and if those documents are updated.
— Whether a preventive and predictive maintenance programme is established and if the tasks and frequencies are being fulfilled.
— If there is an adequate system for technical documents and records.
— If there is maintenance of historical records for all equipment with safety importance.
— If there are enough and adequate tools and equipment.
— If the instruments are periodically checked.
— If there is enough qualified personnel in all maintenance areas.
— If 'spares' stock is adequately identified and stored.

4.7. Modifications and new experiments

— Evaluation of changes in systems, components or procedures proposed by the operator.
— Inspections and audits during construction, modification and commissioning.
— Verification of the fulfilment of procedures related to modifications and changes.
— Evaluation of new experiments or uses.
— Revision of new OLC.

4.8. Special issues (can be classified in the following ways)

— Design;
— Construction;
— Commissioning;
— Modifications and new experiments;
— Decommissioning;
— Projects.

In some cases, there are relevant or special situations that require specific treatment and a slightly different organization. Some examples that we had include the RA-3 power increase from 5 MW to 10 MW; RA-0 critical assembly commissioning; RA-8 critical assembly commissioning; licensing a high pressure and high temperature loop at RA-3, etc. In those cases, an ad hoc committee is constituted, led by the RRCA responsible, and composed of three or four people with different specialities. The work plan is designed based on the specific task.

On the other hand, apart from the regulatory control, there are special projects referred to research reactors and critical assemblies that are managed by the RRCA group. The most important examples are development of programmes for licences and specific authorizations, revision of operational limits and conditions for all installations, revision of all standards related to research reactors and critical assemblies, development of guides for mandatory documentation content.

5. QUALIFICATION

The personnel taking part in the tasks of regulatory control of the research reactors and the critical assemblies can be classified into three groups according to its formation and requirements of qualification: administrative support, technical support and regulatory control.

The administrative support group includes the personnel who develops conventional administrative tasks.

The technical support group is composed of the personnel that pertains to measurement laboratories and specialists in diverse technical fields (dosimetry, shields, thermohydraulics, etc.) that work according to the requirements of the regulatory group. The great majority of this group has a university degree in the field of engineering and physics, and experience in their specialities.

The regulatory control group is formed by the personnel developing tasks in contact with the facilities and coordinating the regulatory control tasks. In this case, despite many of those individuals having their own speciality, they require greater formation than in previous cases. In this sense, in the case of the research reactors and the critical assemblies, the technical knowledge required to be able to develop tasks at a senior level is equivalent to that demanded for a position of head of a reactor. A summary of the required qualification appears in Table 3.

The indicated qualifications must allow the development of the following capacities:

(1) Analysis and evaluation of the following documents:
- Safety report (nucleus, thermohydraulics, instrumentation and control, electrical system, radiological safety, conduct of operations, quality assurance, emergencies, analysis of accidents, dismantling, etc.).
- Mandatory documentation (operations manual and procedures, maintenance manual and procedures, radiation protection manual and procedures, emergencies manual and procedures, quality assurance manual).
- Records and information (dosimetry, effluents discharge, relevant events, wastes, core management, etc.).

(2) Evaluation of design or modifications in: instrumentation and control (hardware, software), electrical, water chemistry, materials, mechanical, thermohydraulics, shields, structures, ventilation, etc.

TABLE 3. SUMMARY OF REQUIRED QUALIFICATIONS

	Qualification
Required knowledge	Radiation protection, nuclear safety, reactor engineering, regulatory standards, quality assurance
Education	Professional with a scientific or technical career
Specialized qualification	Postgraduate degree in radiological and nuclear safety Postgraduate degree in nuclear engineering
Complementary qualification	Quality assurance, regulatory standards
Training	Knowledge of plant systems and operation, planning and organizing inspections and audits, evaluating technical documents and reports, analysing incidents, participating in the preparedness and assessment of emergency plan exercises, etc.

(3) To make audits of radiation protection, emergencies and accident management, quality assurance, maintenance, core management, irradiations and experiments, operation.
(4) To make examinations of licences, specific authorizations, retraining.
(5) Other capacities: revision and writing of standards, writing of licences and authorizations, preparation of annual plans of activities.

6. PLANNING

A general plan is prepared annually. This plan contains the inspections and audits schedule, objectives and scope, inspection and audits teams, specialists required and budget. The plan is prepared for each reactor and is sent to reactor managers in advance, in order to coordinate activities. As an example, a description of the regulatory control activities performed annually in one critical assembly and one research reactor follows.

6.1. RA4 Critical Assembly

Total staffing is 80 staff d/a:

- Two complete inspections including all subjects indicated in the RA-4 Regulatory Control Guide:
  - Inspection team: two persons, senior and junior;
  - Preparation: 4 days;
  - Performance of activities: 2.5 days;
  - Report: 2 days;
- One partial inspection and emergency plan exercise:
  - Inspection team: two persons, senior and junior;
  - Preparation: 6 days;
  - Performance of activities: 3 days;
  - Report: 3 days;
- One partial inspection and examinations (licences, specific authorizations and retraining):
  - Inspection team: two persons, senior and junior;
  - Preparation: 5 days;
  - Performance of activities: 3 days;
  - Report: 3 days.

6.2. RA6 Research Reactor

Total staffing is 146 staff days/a:

- Four thematic inspections including radiation protection, maintenance, operation and nuclear safety:
  - Inspection team: two persons, senior and junior;
  - Preparation: 5 days;
  - Performance of activities: 3 days;
  - Report: 4 days;
- One partial inspection and emergency plan exercise:
  - Inspection team: two persons, senior and junior;
  - Preparation: 6 days;
  - Performance of activities: 3 days;
  - Report: 3 days;
- One partial inspection and examinations (licences, specific authorizations and retraining):
  - Inspection team: two persons, senior and junior;
  - Preparation: 6 days, performance of activities: 4 days;
  - Report: 3 days.

7. REGULATORY STANDARDS

Regulatory standards in Argentina have a performance basis: they are not prescriptive in nature, but they define the accomplishment of safety objectives. How such objectives are achieved depends on the adequate decisions taken by the organization in charge of the design, construction, commissioning, operation and decommissioning.

The regulatory standards in force applicable to the research reactors and the critical assemblies are listed as follows:

General (research reactors and critical assemblies):

AR 0.0.1.	Licensing of Type I installations
AR 0.11.1.	Licensing of personnel of Type I installations
AR 0.11.2.	Psychophysical aptitude requirements for specific authorizations
AR 0.11.3.	Retraining of personnel of Type I installations
A.R. 4.1.1.	Occupational exposure
AR 10.1.1.	Basic Radiation Safety Standard
AR 10.12.1.	Radioactive waste management

AR 10.13.1.	Basic standard on the physical protection of nuclear materials and installations
AR 10.14.1.	Assurances of non-diversion of nuclear materials and of material, installations and equipment of nuclear interest
AR 10.16.1.	Transport of radioactive materials

Applicable to research reactors:

AR 4.1.2.	Limitation of radioactive effluents from nuclear research reactors
AR 4.1.3.	Accident related radiological criteria in nuclear research reactors
AR 4.2.2.	Design of research reactors
AR 4.2.3.	Fire protection in research reactors
AR 4.5.1.	Electric power supply system design for research reactors
AR 4.7.1.	Documentation to be submitted to the regulatory authority prior to the commissioning of a research reactor
AR 4.8.2.	Pre-nuclear commissioning of research reactors
AR 4.9.2.	Operation of research reactors

Applicable to critical assemblies

AR 4.2.1.	Design of critical assemblies
AR 4.7.2.	Documentation to be submitted to the regulatory authority prior to the commissioning of a critical assembly
AR 4.8.1.	Pre-nuclear commissioning of critical assemblies
AR 4.9.1.	Operation of critical assemblies

AR regulatory guides

Guide AR 4	Design of nuclear research reactors
Guide AR 7	Design of critical assemblies

FOCUS ON DEFENCE IN DEPTH IN REGULATORY ASSESSMENT

B. LAWRENCE, V. DIAMOND, D. MACNAB
Regulatory Branch,
Australian Radiation Protection and Nuclear Safety Agency (ARPANSA),
Sydney, Australia
Email: don.macnab@health.gov.au

Abstract

When performing regulatory assessments of licence applications for nuclear facilities, the Australian Radiation Protection and Nuclear Safety Agency (ARPANSA), Australia's Federal Government regulator, places significant emphasis on the implementation of defence in depth. ARPANSA makes the emphasis clear, primarily to its own assessors but also to the operating organisations and the public, by specific mention of defence in depth in selected principles and criteria that are recorded in two key documents used in ARPANSA's regulatory assessment process. The importance is further underlined by structuring those documents by defence in depth level. Some examples are given for the specific case of the replacement research reactor, a 20 MW(th) pool-type reactor currently under construction in Australia. The examples include some regulatory assessment criteria used, and how both the resulting regulatory assessment report and the detailed design approval process for items important to safety reflect that emphasis on defence in depth.

1. INTRODUCTION

When performing regulatory assessments of licence applications and other submissions for nuclear facilities, the Australian Radiation Protection and Nuclear Safety Agency (ARPANSA), Australia's Federal Government regulator, places significant emphasis on the implementation of defence in depth, drawn from the work of the International Nuclear Safety Advisory Group (INSAG) of the IAEA.

The first significant step in that direction occurred in 1994 when ARPANSA's predecessor organization, the Nuclear Safety Bureau, began to formulate a document, now called the Regulatory Assessment Principles for Controlled Facilities [1], also referred to as the RAPs. Under Australian legislation, 'controlled facilities' include research reactors, nuclear reactor fuel stores, nuclear waste storage or disposal facilities, and facilities for the production of radioisotopes.

The aim in producing the RAPs was to compile those regulatory assessment criteria on which ARPANSA would place most importance when performing regulatory assessments of the safety of nuclear facilities. The RAPs document was structured by defence in depth level to underline the importance of the principle. The document was written so it could be applied to the regulatory assessment of any of the above facilities. It contains 123 principles (criteria in [1] are referred to as principles) that span the siting, design and construction, operation and decommissioning stages.

Another significant step occurred in 1997 when ARPANSA began drafting the document entitled Regulatory Assessment Criteria for the Design of New Controlled Facilities and Modifications to Existing Facilities [2], also referred to as the Design Guideline. Again, the Design Guideline was structured by defence in depth level and contains 290 criteria that are particularly relevant to design and construction. These criteria expand on the application of the Regulatory Assessment Principles.

The audience for both documents is primarily the assessors within ARPANSA. Since the documents are public, however, they are available to assist operating organizations in the preparation of the safety analysis report that might accompany an application for a licence or other submission. The documents also serve to inform the public about ARPANSA's regulatory assessment process. During their preparation, the documents were subjected to review comments from an advisory committee external to ARPANSA that has industry and public special interest group representation.

2. EMPHASIS ON DEFENCE IN DEPTH

Both documents [1] and [2] play a key role in making it clear to those audiences that ARPANSA is serious about the implementation of defence in depth. Both state upfront that ARPANSA strongly supports the application of defence in depth and particularly looks for its proper implementation. What constitutes 'proper' implementation is determined by the actual assessment criteria in those documents.

The levels of defence in depth referred to in this paper are those of [3] and are summarized with emphasis on design aspects in Table 1.

TABLE 1. LEVELS OF DEFENCE IN DEPTH

Level of defence in depth	Objective	Essential means
1	Prevent failures, and ensure that anticipated operational occurrences and disturbances are infrequent	Conservative, high quality, proven design
2	Maintain the intended operational states and detect failures	Process control and limiting systems, other surveillance features
3	Protect against design-basis accidents	Safety systems
4	Limit the progression and mitigate the consequences of beyond-design-basis accidents	Accident management and mitigation (including design aspects)
5	Mitigate the radiological consequences of beyond-design-basis accidents	Siting and off-site emergency response

3. REGULATORY ASSESSMENT PROCESS

Regulatory Branch's assessors consider how a licence application or other submission from the operating organization addresses each principle in [1] and each criterion in [2]. ARPANSA recognizes that the relevance of each of the principles and criteria will be different for different types of controlled facilities and takes that into account at the time of the assessment.

In preparing the principles and criteria that are documented in [1] and [2], ARPANSA drew from international publications and experience, especially from INSAG and the IAEA. However, some particular views and convictions of Regulatory Branch staff, formed from experience with Australia's existing research reactor HIFAR (a 45 year old 10 MW(th) heavy water, tank-type reactor) and observation of the nuclear industry worldwide, also significantly coloured the Regulatory Branch's approach to the assessment of defence in depth's implementation.

Some examples of ARPANSA's regulatory assessment criteria are given in the following sections. The examples are by no means exhaustive, and are intended only to give a glimpse of the approach. For more details, reference should be made to documents [1] and [2] on the ARPANSA web site.

Currently under construction in Australia is the replacement research reactor (RRR), a 20 MW(th) pool-type reactor. In the case of the RRR application for licence authorizing construction and subsequent submissions that sought construction approval for items important to safety following their detailed design, all of the assessment criteria examples in this paper were considered highly relevant. The report of ARPANSA's regulatory assessment of the RRR construction licence application using the principles in [1] and the criteria in [2] was itself structured by defence in depth level so as to retain the emphasis on defence in depth throughout the entire assessment process.

4. INDEPENDENCE OF THE LEVELS OF DEFENCE IN DEPTH

There is strong emphasis on ensuring, as far as is reasonably practicable, that the levels of defence in depth are independent of each other. The fundamental principle of defence in depth is that should one level fail, the subsequent level will be invoked and will be effective [3].

ARPANSA Regulatory Branch believes that if a failure of one or more defence in depth levels were to occur, it would likely be because of one or more of the following:

- Inadequate physical and functional independence of redundant systems within a defence in depth level (see the section on extent of redundancy and diversity below);
- Inadequate physical and functional independence of defence in depth levels;
- A safety system performance failure (see the section on performance testing below);
- Excessive reliance on one level of defence in depth to achieve safety.

ARPANSA Regulatory Branch particularly places strong emphasis on the independence of defence in depth level 2 (control systems) and level 3 (safety systems) [3].

When performing its regulatory assessment of the RRR, the Regulatory Branch placed substantial importance on the independence of defence in depth levels, particularly the physical and functional independence of:

- Reactor Control and Monitoring System (RCMS computer based, level 2) from systems on other defence in depth levels;

— First Reactor Protection System (FRPS, computer based, level 3) and the Second Reactor Protection System (SRPS, level 3) from systems on other defence in depth levels;
— First Shutdown System (FSS, level 3) and the Second Shutdown System (SSS, level 3) from systems on other defence in depth levels;
— Post-Accident Monitoring System (PAMS, level 4) from systems on other defence in depth levels.

The following are examples of relevant regulatory assessment criteria. In these and other examples in this paper, the number in square brackets is the document from which the criterion is taken and the number in regular brackets is the number of the criterion in that document. Where the document [1] is also referenced, it means that the criterion is based on a principle (regular brackets) from [1].

[2](1),[1](2)	The design of the facility implements the principle of defence in depth by providing diverse layers of protection as successive levels, as shown in Table 1.
[2](2)	As part of the implementation of defence in depth, the design addresses the use of physical barriers to confine nuclear materials, and addresses the failure of each barrier, including its failure as a consequence of failure of another barrier. Examples of barriers in the case of nuclear reactors are fuel matrix, fuel cladding, primary heat transport system pressure boundary, pool and reactor building.
[2](3),[1](88)	Independence and diversity are provided between levels of defence in depth.
[2](201),[1](97)	Process control systems of nuclear reactors are kept functionally and physically separate from safety systems, from measurement of the process variable (avoiding common sensors) through to shutdown actuation, so that a system is not relied upon to perform both control and safety functions.

5. SPREADING THE BURDEN ACROSS DEFENCE IN DEPTH LEVELS

There is emphasis on appropriately spreading the safety burden across the defence in depth levels, taking into account a wide range of conditions that may be encountered. Examples of criteria are:

[2](4) The facility is designed so that the first two levels of defence in depth carry the primary burden for nuclear safety, with the greatest emphasis on the first level.

[2](5) The design implements defence in depth so that the expected response of the facility following a PIE (initiating event that is postulated in the safety analysis) is as near to the top of the following list as can reasonably be achieved.

The PIE produces no significant safety related effect or a change towards safe conditions as a result of its inherent safety characteristics.

The facility is rendered safe by passive features or by the action of items at defence in depth level 2 that are continuously operating in the state required to control the PIE.

The facility is rendered safe by the action of items at defence in depth level 3 that are brought into service in response to the PIE.

The facility is rendered safe by the specified accident mitigation and emergency response procedures at defence in depth levels 4 and 5.

[1](114) Accident management at defence in depth level 4 is not relied upon at the expense of good design at defence in depth levels 1–3.

[2](6),[1](20) The design takes into account changes in defence in depth effectiveness during facility shutdowns, or during operations with controlled materials including radioactive waste.

In the Preliminary Safety Analysis Report (PSAR) that was part of the RRR construction licence application, there was emphasis on defence in depth. In Chapter 16, for example, the approach to defence in depth was illustrated by a tabular approach for all the identified accident sequences. Examples are given in Tables 2 and 3.

The classification of accidents into design-basis accidents and beyond-design-basis accidents will result from decisions taken by the designer. Those decisions will take into account, among other things, the characteristics of the facility and the particular relationship between the consequences of accidents and their frequency. Accordingly, in [1] and [2] there are no regulatory assessment principles or criteria that attempt to influence those decisions. In the case of the RRR, the PSAR does not identify any design-basis accidents that cause fuel melting if defence in depth level 3 protection works as intended.

TABLE 2. INADVERTENT CONTROL ROD WITHDRAWAL

Level	Main characteristics	Safety feature
1	Conservative design and inherent safety features	Only the central plate moves during the entire operation cycle. The remaining four plates are required for a short period after startup and are withdrawn for most of the operation cycle, ready to shut down the reactor.
		Reactivity compensation through burnable poison in the fuel assembly.
		Proven design of CRD system and devices.
		Limit in safety plate withdrawal velocity.
		CR position is not used for power measurement.
		Adequate QA inspection and maintenance programme.
2	Operation control and response to irregular operation	Alarm on (a) high neutron flux, and (b) high neutron rate.
		Automatic reactivity control system.
		Control Rod Movement Protection Interlock.
3	Control of accidents within the design basis	Safety functions independent of control function.
		FRPS trips the reactor on (a) high neutron flux, and high neutron rate. SRPS trips the reactor on (a) high neutron flux, and (b) no end-of-stroke signal from two or more CRDs.

6. EXTENT OF REDUNDANCY AND DIVERSITY

There is emphasis on ensuring that, within each defence in depth level, the highest level of physical and functional independence is used in redundant systems where the safety significance is highest. Examples from the RRR of relevant redundant systems are the First Reactor Protection System (FRPS) and Second Reactor Protection System (SRPS), both of which are on defence in depth level 3, the First Shutdown System (FSS) and Second Shutdown System (SSS), both on level 3, and the first and second trains of the Post Accident Monitoring System (PAMS) (level 4).

TABLE 3. PRIMARY COOLING SYSTEM PIPE BREAKS

Level	Main characteristics	Safety feature
1	Conservative design and inherent safety features	Large safety margins in piping and valves specification. Piping built of stainless steel following strict application of design codes. Piping properly supported. Piping designed to withstand SL-2 earthquake. High water quality to avoid corrosion. Appropriate maintenance programme. Appropriate system inspection and tests during installation and on service inspection. Leak detectors. Siphon Effect Breakers. PCS pumps adequately fixed to support basis that permit to prevent stress on the connected piping.
2	Operation control and response to abnormal operation	RCMS pump trip signal on low pool water level. High vibration and high temperature indication on motors, flywheels and pumps. RCMS pump trip signal on very high vibration in pump bearings and flywheel bearings. Alarms on: (a) High water level in leak detectors (b) High water level in LOCA sumps (c) Low water level in the reactor pool (d) Low core pressure drop and low flow (depending on the location of the break) (e) High radiation level
3	Control of accidents within the design basis	Two independent and diverse shutdown systems. FRPS reactor trip signal on: (a) Low water level in the reactor pool (b) Low core pressure drop (c) Low PCS flow (depending on the location of the break) (d) High radiation level SRPS reactor trip signal on: (a) Low water level in the reactor pool (b) Low core pressure drop (c) Failure of the FSS Passive redundant siphon effect breakers.

TABLE 3. PRIMARY COOLING SYSTEM PIPE BREAKS (cont.)

Level	Main characteristics	Safety feature
4	Limit progression of accidents beyond-design-basis	Post Accident Monitoring System. Emergency Make-up Water System.

Examples of criteria are:

[2](17),[1](45) A categorization by safety significance is carried out for systems, structures and components (including software), ranking them in terms of safety significance.

[2](43),[1](91) The highest degree of physical and functional independence is used in redundant systems where the safety significance is highest.

[2](44),[1](104) Whenever the categorization by safety significance designates items of process plant as safety related:

(a) The principles of redundancy, independence and diversity, including consideration of common cause failures, are applied to those items to the extent that is appropriate to their categorization by safety significance.
(b) Safety analyses confirm that those items do not cause a design imbalance, that is, failures of the items do not contribute to risk disproportionately in comparison with other accidents.

[2](197),[1](92) Redundant subsystems in safety systems are kept functionally and physically separate, from measurement of the process variable (avoiding common sensors) through to shutdown actuation.

[2](205),[1](94) Diversity is used in redundant safety systems, from measurement of the process variable through to shutdown actuation.

7. PERFORMANCE TESTING

There is emphasis on testing for performance failures as a part of implementing defence in depth. In this context, a 'performance failure' means a failure that occurs when a system, which is important to safety, functions when required (that is, no components fail), but it functions in such a way that it does not provide the intended protection. A performance failure can arise from incorrect (plant modelling and safety) analyses, and can manifest under accident conditions as the wrong type of operation, or as an operation having insufficient or inadequate response or response time. This is relevant to all RRR items that are important to safety, including the provisions for natural circulation cooling under accident conditions, and the FRPS-FSS and SRPS-SSS systems. Examples of criteria are:

[2](243), [1](108)	To demonstrate satisfactory safety system performance, testing of safety systems emphasizes performance failure aspects by simulating as faithfully as reasonably practicable the actual initiating event and conditions. The design facilitates testing for verification of performance, each redundant part of a safety system, and the safety system in its entirety.
[2](48a)	Where computer based systems are used in items important to safety, appropriate standards and practices are applied in the development, validation and testing for verification of performance of the hardware and software at each phase of their development.

8. CONCLUDING COMMENTS

It is recognized (see Ref. [3]) that the implementation of defence in depth may differ from country to country, being design specific and influenced also by individual safety policies and regulations of Member States. The regulations in the Australian legislation, for example, specifically mention some radiation dose figures.

ARPANSA's Regulatory Branch has not regretted formulating its own (limited number) of regulatory assessment guidelines to make it clear to its assessors, the operating organizations and the Australian public, where we are coming from when performing regulatory assessments, including what areas we feel particularly strongly about. This is in line with the openness identified in Ref. [4] as a future challenge for nuclear regulatory bodies. An alternative approach, where we simply indicate to our assessors the many overseas documents, would miss an opportunity for special focus and emphasis on defence in depth aspects that we consider particularly important.

Based on our regulatory assessment experience in and around defence in depth and its proper implementation, Regulatory Branch endorses the exhortation from the CEO of ARPANSA to "communicate, communicate, communicate—there cannot be too much of it." We state, in the executive summaries of the two key regulatory assessment guidelines, our strong support of the application of defence in depth and that we particularly look for its proper implementation when performing regulatory assessments. We include in those documents specific assessment criteria that our assessors are to use, and we ensure that they are used. We present our regulatory assessment results to operating organizations in a manner that shows we have used those criteria in the assessment.

But in spite of that, we realize we have more work to do before we can say that we are focusing on defence in depth and its proper implementation to the extent that we think it deserves. That remains a major Regulatory Branch challenge, one that we are currently addressing, in part, by the use of licence conditions and follow-up of their action status.

REFERENCES

[1] AUSTRALIAN RADIATION PROTECTION AND NUCLEAR SAFETY AGENCY, Regulatory Assessment Principles for Controlled Facilities, RB-STD-42-00 Rev. 1, ARPANSA, Sydney (2001).

[2] AUSTRALIAN RADIATION PROTECTION AND NUCLEAR SAFETY AGENCY, Regulatory Assessment Criteria for the Design of New Controlled Facilities and Modifications to Existing Facilities, RB-STD-43-00 Rev. 1, ARPANSA, Sydney (2001).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Implementation of Defence in Depth for Next Generation Light Water Reactors, IAEA-TECDOC-986, IAEA, Vienna (1998).

[4] OECD NUCLEAR ENERGY AGENCY, Future Nuclear Regulatory Challenges, Report by the NEA Committee on Nuclear Regulatory Activities, OECD, Paris (1998).

BLANK

SAFETY REQUIREMENTS APPLIED TO RESEARCH REACTORS IN FRANCE

H. ABOU YEHIA, G. BARS
Institut de Radioprotection et de Sûreté Nucléaire (IRSN),
Département d'Évaluation de Sûreté (DES),
Fontenay-aux-Roses, France
Email: hassan.abouyehia@irsn.fr

Abstract

After a general overview of the safety requirements applied to research reactors in France, the paper details the specific safety requirements adopted for the various barriers of the pool-type research reactors and related systems. Emphasis has been put particularly on the conditions in which a BORAX-type explosive reactivity accident has been considered for the design of these reactors. The safety of experimental devices as well as the process of periodic safety reassessments are also addressed.

1. INTRODUCTION

In France, there are currently some 20 research reactors in operation with a thermal power up to 70 MW (see Annex). The oldest reactor in operation (ULYSSE) went critical in 1961, and the most recent (ORPHEE) in 1980. Various facilities have been gradually subject to modifications so as to adapt them to changes in the safety requirements which occurred after their construction, especially regarding the seismic risk and fire hazards. Other modifications were implemented to meet new experimental needs; these modifications required safety reassessments of the concerned facilities.

The safety approach adopted for the design of all research reactors currently in operation was based on the method of barriers and the concept of 'defence in depth'. According to this method, public and environment protection consist in interposing in series a minimum of three independent barriers. Safety analysis consists in checking the good behaviour of each barrier in normal and accident conditions, considering the measures related to prevention, control and protection, and safety actions. The safety analysis of these barriers, requiring a study of the development of certain typical accidents, keeps in some cases a conventional character consisting in considering the occurrence of these accidents without identifying precisely the causes which might initiate them.

After an overview of the general safety requirements adopted for the reactors mentioned, the following paragraphs explain the requirements specific to the different barriers and related systems by considering the case of the pool-type reactors representing most of the operating reactors.

2. SAFETY REQUIREMENTS APPLIED TO THE REACTOR DESIGN AND THEIR EVOLUTION

2.1. General safety requirements

Safety requirements for research reactors that are operated in France have changed over time. Some of these requirements, including some requirements concerning the neutronic and thermal-hydraulic core design and the consideration of an explosive-type reactivity accident for the reactor design, were applied for all pool-type research reactors.

The purpose of the thermal-hydraulic safety criteria, adopted from the beginning for the pool-type research reactors operating at high power and using plate-type fuel, is to avoid the flow redistribution phenomenon which might lead to fuel melting. These criteria require a verification for each reactor (after cumulating all uncertainties at the hot spot) of the absence of such phenomenon for operating conditions corresponding to the limits of safety thresholds and for transients, including the transition to cooling regime by natural convection.

The major safety requirement, applied in France only, consisted in taking into account a BORAX-type explosive reactivity accident (see Section 2.4) for the design of the pool-type research reactors using metallic fuels in the form of uranium and aluminium alloy. For such an accident, supposed to lead to a total core meltdown under water, the main safety requirement is to ensure the reactor pool and containment building tightness.

General safety requirements relating to internal and external hazards, and those relating to the redundancy and the separation of protection system channels, to the continuous monitoring of confinement barriers and containment building leak-tightness with respect to underlying soils and underground water, have been gradually established and applied.

It should be indicated that the safety authority (Directorate General for Nuclear Safety and Radiation Protection—DGSNR) has established basic safety rules specifying the general principles for the design of nuclear facilities, the safety qualification and classification of their equipment, as well as the consideration of the risks related to internal and external hazards. As for research reactors, there are today three basic safety rules relating to the filtration devices equipping the ventilation systems, the means of meteorological measurements and the protection against fire hazards. The other rules, mainly established for nuclear power plants, have been applied to research reactors with some adaptation.

For instance, regarding the seismic risk and the risks relating to the industrial environment and transportation of hazardous materials, the rules applied to research reactor design are the same as those applying to nuclear power plants, albeit with some adaptations due to the specific features of certain reactors (short operating time or low radioactive product inventory). For example, in the case of the CABRI reactor (25 MW in steady power level) and PHEBUS reactor (38 MW), operating only a few weeks during the year for performing safety tests, the design of certain structures at the Maximum Historically Probable Earthquake (MHPE) was considered as acceptable instead of a design at the Safe Shutdown Earthquake (SSE). Besides, some research reactors are equipped with a seismic detection initiating an automatic shutdown in case of an earthquake.

It should be noted that for the various research reactors operated in France, the confinement barriers of radioactive products and related systems have been classified as elements which are important for safety according to the decree of 10 August 1984 regarding the quality of the design, construction and operation of the basic nuclear installations. A synthesis of the basis of these requirements is presented below.

2.2. Safety requirements relating to the barriers

2.2.1. First barrier (fuel-cladding)

The main safety requirements relating to the first barrier are the following:

- Nucleate boiling must be avoided for fuel plate cladding under normal operating conditions. The verification of this condition for the hot spot should take into account, in a worst-case scenario, the various uncertainties associated with neutronic and thermal-hydraulic calculations;
- There must be no fuel-cladding dry-out under the various operating conditions; this requirement implies checking the absence of flow redistribution and the absence of Departure from Nucleate Boiling (DNB) in the hottest cooling channel (all uncertainties are cumulated in a projected worst-case scenario at the hot spot);

— Reactors must not be operated with a fuel element affected by a cladding failure; in such a situation, some reactors are automatically shut down and the fuel element in question removed and stored in a leak-tight container.

A number of tests were performed to validate the safety demonstrations relating to these issues. These tests were aimed particularly at determining the fuel behaviour under irradiation and the operating limits of the fuel used, including allowable burn-up.

In the past, the fuel dry-out conditions were directly determined by carrying out tests in the CABRI reactor to simulate flow or power excursion transients. In some tests, the flow redistribution incidents produced by reactivity insertions led to a partial fuel melting. Today, fuel dry-out conditions are determined by using computer codes based on results of experiments carried out in electrically heated channels representative of the core fuel cooling channels. The validation of these codes for neutronic and thermal-hydraulic studies of the reactor cores is an important safety issue. In this respect, the validity of these studies was checked, during reactor startup tests, by using fuel elements equipped with thermocouples to measure directly the cladding temperature.

Finally, it should be noted that prior to the use of a new type of fuel element, the behaviour of the fuel with its cladding has to be verified by means of qualification tests, including irradiation tests covering the operating conditions.

2.2.2. Second barrier

Integrity of the reactor core coolant system and the pool should be ensured in the normal operating conditions and in accident conditions including earthquake. The design and quality of the construction of the second barrier were subject to particular attention from the safety authority.

The main safety requirements relating to the second barrier are the following:

— The core must not be uncovered in the event of a pipe break in the reactor coolant system or a window failure in neutron beam channels; this requirement is generally met through the integrated design of the reactor primary coolant system, which is installed in a ‘water block’, and through the implementation of siphon breakers on the reactor coolant system pipes and automatic isolation valves on the neutron beam channels;

— The mechanical resistance and tightness of the pool should be ensured in the normal operating conditions and in the accident conditions taken into account for the design. The verification of this requirement should take into account accidents that may occur on experimental devices, especially on pressurized irradiation loops.

2.2.3. Third barrier

The main safety requirements relating to the third barrier are the following:

— The mechanical resistance and tightness of the reactor containment building shall be ensured in normal operating conditions and accident conditions. Accident conditions include mainly earthquakes, fires, internal or external floods, internal or external explosions, missiles of internal origin and aircraft crashes;
— The mechanical resistance and tightness of the reactor containment building should be ensured in the case of a BORAX-type accident. The verification of this resistance shall take into account the effect of the water column ejected from the pool during the accident, the pressure increase due to the released thermal energy and the possible ejection of internal missiles;
— The emergency ventilation system associated with the reactor containment building shall be equipped with a filtration system for the air released by the chimney, comprising HEPA filters and iodine traps. The design of this system shall allow the periodic checking of its efficiency, which shall be higher than 1000 (this value applies to molecular iodine in the case of iodine traps). The emergency ventilation system shall be equipped with an air heater upstream the iodine traps, to reduce the air hygrometry in order to keep their specified efficiency in all accident conditions that may lead to an excessive increase of the air hygrometry in the containment building (in particular in the case of a BORAX-type accident);
— The design of the reactor containment building and associated ventilation systems shall be such that the radiological consequences of the accidents are minor for the public and the environment, even in the case of a BORAX-type accident or the melting of a fuel element in the air. It should be noted that in the case of the high flux reactor (RHF, P = 58.3 MW), equipped with a double containment, the envelope accident taken into account for the study of the radiological consequences is the core meltdown in the air.

2.3. Surveillance and protection of the barriers

The safety requirements relating to surveillance and protection systems associated with the barriers aim at ensuring a satisfactory reliability to these systems and a low vulnerability to hazards.

The means of surveillance associated with the barriers, in particular, include:

— For the first barrier: the neutronic, thermal-hydraulic and radiological measurements, as well as the cladding failure detection system;
— For the second barrier: the measurements of water level in the pool, the detection of water leaks of the reactor coolant system and neutron beam channels, the measurements relating to the chemical quality of water (prevention of corrosion of the fuel-cladding and structures located in the pool);
— For the third barrier: the measurements of depression in the reactor containment building, the detection of water leaks in the underlying soils and the measurements of activity in the air released by the chimney.

The measurements mentioned are connected to the control and protection systems of the reactor. The safety actions relating to the different barriers include the power reduction or emergency shutdown of the reactor, the isolation of the reactor containment building and the normal ventilation shutdown followed by the actuation of the emergency ventilation (if a significant contamination is detected in the reactor containment building).

The main safety requirements relating to the surveillance and protection systems are the following:

— Compliance with the unique failure criterion which is achieved by means of redundancy, diversification and physical separation of the control and safety equipment;
— Qualification of the reactor protection system for earthquake conditions and electromagnetic interference;
— Adaptation of measurement ranges of radiological monitors for dose rates resulting from a BORAX-type accident or the meltdown of a fuel element in the air.

For old reactors, successive modifications and renovations were performed on the control and protection systems to meet these requirements.

2.4. Consideration of the BORAX-type accident

2.4.1. Background

In a pool-type reactor, there is more probability of a reactivity accident than in other reactor types, due to the easy access to the core and frequent manipulations of fuel elements and experimental devices.

Tests on BORAX (in 1954) and SPERT reactors (in 1962) in the United States of America, and the SL-1 reactor accident in 1961, showed that light water cooled and moderated reactors with U-Al alloy fuel could, when subjected to sudden large additions of reactivity, sustain violent power excursions leading to a partial or total reactor core melting.

Reactivity insertions of 2% $\Delta k/k$ or more, producing periods of less than 4 milliseconds (ms), can in fact destroy a reactor core through a rapid steam bubble formation and the resulting short but significant overpressure. Several series of reactivity insertion tests were performed in the CABRI reactor in 1966 with various plate fuel lattices. The minimum power excursion periods reached during these tests were about 10 ms. In the experiments carried out, meltdown was limited to a maximum of three fuel plates (out of more than 300 plates constituting the studied cores).

French nuclear safety organizations and authorities were quick to take interest in potential reactivity accidents and to recommend:

— Provisions for preventing important reactivity insertions in the core, particularly during handling operations;
— Allowance for a BORAX-type explosive accident in designing pool-type reactors with uranium-aluminium alloy fuel and significant potential reactivity of the core.

2.4.2. Assumptions adopted for the reactor design

The main assumptions adopted in France for the design of the pool-type reactors in operation, were:

— Complete meltdown of the core under water;
— Total energy release of 135 MJ, including 9% in the form of mechanical energy participating in the deformation and destruction of the internal structures and the ejection of a water column outside the pool; it should be added to it the energy released by the destruction of the experimental devices (hot and cold sources, pressurized irradiation loops, etc.).

These assumptions were used prescriptively for the design of the pools and containment buildings of the various pool-type research reactors using a plate-type fuel with a core potential reactivity exceeding 2% $\Delta k/k$, without considering the initiating event or scenario that may lead to a fast injection of this reactivity.

In the case of the high flux reactor (RHF), the average temperature of the molten aluminium mass and the vaporized fraction of this mass were estimated respectively at 800°C and 5%. In addition, the mechanical energy due to the liquid deuterium vaporization after the cold source destruction during the accident corresponds to 3.7 MJ, and the thermal energy stored in the hot source graphite is 50 MJ. In the case of contact of the graphite with the reflector heavy water, a fraction of this energy may be converted into mechanical energy whose maximum value is estimated at 4 MJ.

The consideration of a BORAX-type accident for the pool design resulted for some reactors (RHF, ORPHEE) in the implementation of a metallic structure separated from the pool liner in order to absorb by deformation a part of the mechanical energy due to the accident.

2.4.3. Validation tests

Due to the great complexity of the structures of some research reactors and the phenomena to be studied, the demonstration of mechanical resistance of the pool structures and reactor containment building, as well as the conservation of their tightness in the case of a BORAX-type accident, were validated experimentally by means of several tests. These tests, intended for checking the design calculations for the structures mentioned, were performed in reduced scale mock-ups (1/3, 1/5 or 1/10) using:

— Explosive material (TNT) in the case of OSIRIS and SILOETTE reactors;
— Pyrotechnic mixture made of iron oxide powder (Fe_2O_3), and aluminium and magnesium in the case of the high flux reactor (RHF), in order to simulate a dispersion of particles at high temperature in water;
— Compressed air gun in order to simulate the water hammer effect on the neutron beam channels of the ORPHEE reactor.

The tests carried out confirmed the envelope character of the assumptions used in the design calculations of the structures. In addition, tests were performed to validate the calculations of pressure increase in the containment building of the OSIRIS reactor, due to thermal exchanges between the water column ejected from the pool and the air of the containment building.

2.4.4. *Evaluation of radiological consequences*

Assumptions relating to the evaluation of radiological consequences of a BORAX-type accident were modified over time to take into account new knowledge acquired on the transfer of fission products from the molten fuel towards the pool water and reactor containment building. The data currently used, concerning the transfer rate of fission products from the molten fuel towards the pool water, are the following:

— 100% noble gases;
— 80% iodine, bromine, tellurium and caesium;
— 10% strontium, barium, ruthenium and rhodium;
— 1% solid fission products (including actinides).

These data take into account the releases estimated from the incident of partial melting of six fuel plates, occurred in the SILOE reactor in 1967.

Concerning the transfer rate of fission products contained in pool water towards the air of the containment building, the main data used are the following:

— Instantaneous transfer:	
Noble gases	5×10^{-2}
Iodine, bromine, tellurium and caesium	5×10^{-4}
— Differed transfer (per day):	
Noble gases	5×10^{-1}
Iodine, bromine, tellurium and caesium	1.25×10^{-4}

3. AGEING AND SAFETY REASSESSMENT

The ageing of facilities has been addressed by gradually refurbishing their equipment and replacing their instrumentation, their control and protection systems and those structures affected by high radiation doses.

In France, nuclear reactors that have been operating for more than 10 years undergo a systematic safety reassessment at the request of the safety authority. For such a reassessment, the applicant has to provide updated safety documents for the installation (safety analysis report, general operating rules, on-site emergency response plan) which take into account the modifications of the installation and those concerning the site conditions due to human activities or improvements in knowledge (e.g. regarding earthquakes). An examination of the integrity of the second and the third barriers has to be carried out, including if necessary non-destructive examinations, in order to check their ability to fulfil the requirements mentioned and to determine needed corrective actions. The applicant has also to provide a new chapter, added to the safety analysis report, presenting operational experience feedback and lessons learned from incidents which have occurred since the reactor first started up. A safety reassessment should include a review of the operating conditions (normal, incident and accident conditions) and the defence in depth aspects, in order to identify possible discrepancies with new safety requirements.

The safety reassessments carried out for different research reactors have not called into question the safety options adopted for their original design. The main requests of the safety authority, following the safety evaluations performed by the IRSN, have concerned the need to upgrade facilities, in particular, to take into account the evolution of safety requirements relating to internal and external hazards. This has led to reinforcements of facilities in order to increase their resistance to earthquakes (RHF, PHEBUS, etc.), and to improvements concerning fire protection and the management of post-accident situations.

4. SAFETY REQUIREMENTS FOR EXPERIMENTAL DEVICES

Experimental devices used in the operating research reactors vary greatly. Before the use or the modification of any experimental device, the operator has to carry out an internal safety assessment. Experiments of important safety implications have to be submitted to the safety authority for review and authorization. The others (with minor safety implications) can be authorized internally by the operating organization according to established procedures.

The experimental devices and the test loops present mainly the following risks for the reactor:

— Explosion of a pressurized irradiation loop which may produce internal missiles affecting the integrity of the reactor pool or the reactor containment building;
— Interaction between the water and the NaK contained in some irradiation devices or interaction between molten experimental fuel and the water contained in the cooling system of the experimental devices;

— Excessive insertion of reactivity in the core.

Finally, it should be noted that the French safety authority is presently establishing safety requirements for the design and use of experimental devices and technical criteria for the determination of their authorization channels.

5. CONCLUSION

Safety requirements currently applied to research reactors in France are consistent with the rules and recommendations issued by the IAEA.

The specific safety requirement for the design of French research reactors is the consideration of a BORAX-type reactivity accident. This requirement resulted in a robust design of the pool and reactor containment building structures and, therefore, a good quality of confinement of the radioactive products in accident conditions.

The safety requirements mentioned and the recent evolutions relating to the safety analysis methodology and to the seismic risk analysis will be applied to the design of the new Jules Horowitz Reactor (RJH, P = 100 MW) which is planned to be built on the Cadarache site.

ANNEX

TABLE A-1. RESEARCH REACTORS (P < 100 KW): MAIN CHARACTERISTICS

Reactor	First criticality	Location	Type, fuel, enrichment, moderator, coolant	Rated maximum power
Neutrongraphy reactor of PHENIX power station	1974	Marcoule	Aqueous homogeneous, uranyl nitrate, 93%	Pulsed
MINERVE	1977	Cadarache	Pool, UAl, 90%, H_2O, H_2O	100 W
EOLE	1977	Cadarache	Pool, various fuels and enrichment, H_2O, H_2O	10 kW
MASURCA	1966	Cadarache	Fast, $(U+PU)O_2$, 30%, air	5 kW
ULYSSE	1961	Saclay	Argonaut, UAl, 93%, graphite, H_2O	100 kW

TABLE A-2. RESEARCH REACTORS (P < 100 KW): OPERATIONAL CHARACTERISTICS AND PRINCIPAL USE

Reactor	Operating schedule	Principal use
Neutrongraphy reactor of PHENIX power station	As required during working days	Non destructive examination of irradiated fuel
MINERVE	As required during working days	Neutronic physics for light water cooled and moderated reactors
EOLE	As required during working days	Neutronic physics for light water cooled and moderated reactors
MASURCA	As required during working days	Neutronic study of fuel lattice arrangement for fast reactors and of accelerator driven reactors
ULYSSE	As required during working days	Training

TABLE A-3. RESEARCH REACTORS (P > 100 KW):
MAIN CHARACTERISTICS

Reactor	First criticality	Location	Type, fuel, enrichment, moderator, coolant	Rated maximum power
High flux reactor	1971	Grenoble	Pool, UAl, 93%, D_2O, D_2O	58 MW
ORPHEE	1980	Saclay	Pool, UAl, 93%, H_2O, H_2O	14 MW
ISIS	1966	Saclay	Pool, U_3Si_2, 19.75% H_2O, H_2O	700 kW
OSIRIS	1966	Saclay	Pool, U_3Si_2, 19.75%, H_2O, H_2O	70 MW
CABRI	1977	Cadarache	Pool, UO_2, 6%, H_2O, H_2O	25 MW
PHEBUS	1978	Cadarache	Pool, UO_2, 2.78%, H_2O, H_2O	38 MW

TABLE A-4. RESEARCH REACTORS (P > 100 KW):
OPERATIONAL CHARACTERISTICS AND PRINCIPAL USE

Reactor	Operating schedule	Principal use
High flux reactor	Continuous operation for 51 days	Basic research (use of neutron beams)
ORPHEE	Continuous operation for 100 days	Basic research, neutrongraphy (use of neutron beams), silicon doping
ISIS	As required during working days	Mock-up of OSIRIS reactor neutrongraphy
OSIRIS	Continuous operation for 25 days	Irradiation and test of fuel and materials production of radioisotopes
CABRI	Intermittent	Research in nuclear safety (power excursions)
PHEBUS	Intermittent	Research in nuclear safety (severe accidents in PWRs)

BLANK

REGULATORY ASPECTS AND EXPERIENCE WITH RUSSIAN RESEARCH REACTORS

S.I. MOROZOV
Department for Supervision of Research and Ship Reactor Nuclear and Radiation Safety,
Federal Nuclear and Radiation Safety Authority of Russia
(Gosatomnadzor of Russia),
Moscow, Russian Federation
Email: msi@gan.ru

Abstract

Gosatomnadzor of Russia is conducting the safety regulation and inspection related to nuclear and radiation safety at nuclear research facilities, including research reactors, critical assemblies and sub-critical assemblies. It implies implementing three major activities: establishing the laws and safety standards of nuclear and radiation safety in the field of research reactors; licensing of research reactors; and inspections (or licence conditions tracking, inspection and application of sanctions). The database on nuclear research facilities has recently been updated, based on the actual status of all facilities. It turned out that many facilities have been shut down, temporarily or permanently, waiting for the final decision on their decommissioning. Compared with previous years, the situation has been inevitably changing. Now there are 81 nuclear research facilities in total under the supervision of Gosatomnadzor (compared with 113 in 1998), explaining their distribution by types and operating organizations. The safety regulation process in the Russian Federation is based on the legal framework consisting of nuclear law, safety standards and regulations. The safety assessment is conducted commensurate with the actual hazard a facility might represent. This fact is reflected in the safety classification of research reactors. The safety status of major nuclear research centres by some decisive factors, such as incidents, effluents and emissions, radiation exposures, and spent fuel and waste management, is outlined. Finally, major problems at current research facilities in the Russian Federation are discussed.

1. INTRODUCTION

The regulatory activity of Gosatomnadzor of Russia [1] includes implementing three major steps:

(1) Establishing the safety standards in the field of nuclear and radiation safety;

(2) Licensing;
(3) Iinspection and enforcement.

Relatively recently, a full set of safety standards and regulations for research reactors (RR) has been established, thus allowing Gosatomnadzor to effectively implement its designated functions in the field of RR safety. A minimum set of these documents is as follows:

Level I:	Fundamentals: — Law “On nuclear energy use” — Law “On public radiation protection”
Level II:	Safety Standard: “General Provisions for Safety of Research Facilities”
Level III:	Safety Rules: — Nuclear Safety — Radiation Safety — Waste Management — Safe Decommissioning of RR — Safety Analysis Report — QAP
Level IV:	Safety Regulations: — Licensing (including peer review and safety assessment) — Inspection

Gosatomnadzor has created and routinely updates the database on nuclear research reactors based on the actual status of all facilities [2]. According to the database, many facilities have been shut down during recent years, temporarily or permanently, waiting for the final decision on their decommissioning. For example, as of September 2003, Gosatomnadzor has 81 nuclear research reactors under its supervision (compared with 113 in 1998). This can be explained by three main reasons:

(1) The experimental programme has been terminated and no new programmes are in place;
(2) There is a lack of financial and human resources;
(3) There are safety problems, e.g. owing to ageing. Currently, 21 operating organizations own RR facilities. The current status of these facilities is given in Table 1.

TABLE 1. CURRENT STATUS OF RR FACILITIES

Type	Operational	Decommissioning	Construction	Total
Research reactors	25	7	2	34
Critical assemblies	28	5	—	33
Sub-critical assemblies	11	2	1	14
Total	64	14	3	81

One of the main difficulties in regulating research reactor safety is the variety of 21 operating organizations with different financial and human resources. Ministries responsible for supporting their operation can only provide limited help. It becomes obvious that a unified governmental programme for RR utilization is urgently needed to decide on how many RR are required and for what purpose, in order to support the national economy of the Russian Federation.

2. RESEARCH REACTOR SAFETY CLASSIFICATION

In order to facilitate the work of Gosatomnadzor in regulating RR safety, a nuclear RR safety classification based on the level of hazard of a facility had been established in 1994, as follows:

Group 1: Nominal power up to and above 100 MW(th) for which there is a potential for severe accidents in all INES scale.
Group 2: Nominal power up to 20 MW(th), devoted to a nuclear core physics study, training and isotope production with a moderate nuclear and radiation risk.
Group 3: Nominal power up to 1 MW(th) where it may not be necessary to organize a forced cool-down of the reactor core in an emergency situation and with a small radiation risk.

3. SELF-ASSESSMENTS AND INCIDENTS

One of the main features of Gosatomnadzor regulating operational safety of RR is the control of annual self-assessment by the operating organization. A corresponding report on that shall be submitted to Gosatomnadzor for review and decision making as a feedback during licence condition implementation inspections.

In 2002, there were 47 incidents (the same number as in previous year) related to the scram of the emergency control system (emergency control rods).

The majority of events were related to:

— Ageing of instrumentation and control equipment;
— Ageing of electric systems equipment;
— Instability of the external electric power supply;
— Human/operator error.

4. SPENT FUEL AND WASTE MANAGEMENT

The major problem and concern of Gosatomnadzor remains the problem of spent fuel transportation from the RR sites. Table 2 summarizes the current status of spent fuel storage facilities at major RR sites.

TABLE 2. STATUS OF SPENT FUEL STORAGE FACILITIES AT MAJOR RR SITES

Site	Reactor	Occupancy (%)
Kurchatov Institute, Moscow	MR	60
	IR-8	36
PhEI, Obninsk	AM-1	60
	BR-10	22
RDIPE Subsidiary (Sverdlovsk)	WWR-2M	80
NIIAR, Dimitrovgrad	CM-3	94
	MIR.M1	97
	RBT-10/1, RBT-10/2	67
	BOR-60	95
	VK-50	56
SPINPh (Gatchina)	VVR-M	37
NIFHI (Obninsk)	VVR-ts	59

As to waste management, all RR sites have specific contracts with the State owned company RADON to deal with radioactive waste problems. Gosatomnadzor does not see any safety problem concerning the ensurance of radiation safety when dealing with radioactive wastes at the sites.

5. RADIATION SAFETY

5.1. Effluents and emissions

The amount of releases (airborne and waterborne) in 2002 was less than in previous years: summed-up releases never exceeded the established control levels [3].

5.2. Occupational radiation exposure

The radiation dose rate to personnel is determined by the modes of the reactor operation at nominal power, by experiment needs, by preventive maintenance and repair of equipment and devices, whether irradiated or located in radiation zones. The current practice at RR in the Russian Federation shows that in areas where there is a constant personnel presence, the radiation dose rate is within 2 to 0.2 μSv/hr, which is well below the permissible one (8 μSv/h). The average radiation exposure at the facilities was 3 mSv/a, whereas the regulatory limit is 20 mSv/a.

6. EMERGENCY PREPAREDNESS

All sites have developed appropriate instructions for personnel during an emergency. There are two plans: an on-site emergency plan and an off-site emergency plan. Both plans are coordinated with each other in some instances. Personnel training on emergency situations is conducted on a regular basis as required by Gosatomnadzor.

7. PHYSICAL PROTECTION

This is one of the important issues and is being successfully resolved with the help of the IAEA and the United States Department of Energy assistance at some selected sites. For other sites, the problem of modern hardware needed for physical protection still exists. There is an action plan and a schedule approved by the Ministry of Atomic Energy to improve the situation in the forthcoming years.

8. SUMMARY OF PROBLEMS

Currently, since most of the RR are quite old and more than 35 to 40 years in operation, four major problems relating to RR safety can be delineated:

- Ageing of equipment that is important to safety. This means that there is an urgent need for the justification of the reactor equipment (structures, elements, cladding, etc.), operability and reliability for the next period of operation. If it is not physically or financially possible to justify or replace the reactor equipment, the decision on their decommissioning has to be made.
- Transportation of spent fuel from the sites. Currently, high costs of its transportation do not allow the problem to be resolved in many cases. There is a need for the Government to intervene.
- High decommissioning costs for some reactors (e.g. reactor MR of the Kurchatov Institute).
- Ageing staff. This problem will become worse in the coming years, owing to lack of incentive to work in the nuclear industry (lack of prestige and low salary).

REFERENCES

[1] Annual Report on Federal Authority Activities in Nuclear and Radiation Safety, Gosatomnadzor of Russia, Moscow (2002).
[2] Russian Federation Research Reactors Data Base, Internal memorandum, Gosatomnadzor of Russia, Moscow (2003).
[3] Radiation Protection Norms, Rep. NRB-99, Ministry of Health, Moscow (1999).

REGULATORY PRACTICE IN A COUNTRY OPERATING ONE NUCLEAR RESEARCH REACTOR

A.T. MIKULSKI
National Atomic Energy Agency (PAA),
Warsaw, Poland
E-mail: mikulski@paa.gov.pl

Abstract

The paper presents the experience gained by a regulatory body in a country operating only one nuclear research reactor. The majority of research reactors in the world have been designed and constructed by experienced companies and their operation is supervised by a local regulatory authority. Although the responsibility of such an authority should cover several topics, its main objective is, nevertheless, safe reactor operation. The regulations, instructions and experience of reactor operation gained in Poland over many years is described in detail, together with some general observations which may be of interest to other countries.

1. INTRODUCTION

The regulatory body plays a very important role in the operation of any nuclear installation, as can be observed during recent years in practice and many international institutions have stressed its importance. The Basic Safety Standards for nuclear reactor operation are one of main achievements of the IAEA, giving an example of good practice to national organizations. These standards are of equal importance both for nuclear power plants and for research reactors. Safe operation of any nuclear reactor is based on two fundamentals, namely, proper design with reliable construction and a safety culture of operating personnel. The first pillar is rather easy fulfilled because there are only a few companies in the world involved in the design and construction of research reactors; the second pillar is based on local authorities in many countries. It can be observed in the past that operators of nuclear power plants (NPPs) are united in two international organizations, i.e. the IAEA since 1957, and the World Association of Nuclear Operators (WANO), since 1989. The first organization is open to all countries and provides coordination at a governmental level, while the second one is a union of NPPs operators. The goal of each is the same, namely, the safe operation of any nuclear reactor. The present status of research reactor safety is very well presented in a paper prepared for the IAEA's General Conference in 2001 [1].

The responsibility of safe operation is put in the hands of an operator, but it should be supervised and supported by a regulatory body. In countries with many reactors, it can be easily fulfilled but in countries with only one or two reactors, it is rather difficult. In this case, broad exchange of information between countries plays a substantial role. The tasks of the IAEA in this field are very important and consist in organizing conferences, symposiums and working groups, and also special INSARR missions, technical cooperation programmes, coordinated research programmes, etc. But from the other side, any operating company should be interested in the application of all available knowledge. Looking at it from the perspective of countries operating one research reactor, the role of the regulatory body will be presented together with some practical examples of its activity.

2. ATOMIC LAW IN POLAND

Basic legislation relating to nuclear safety and radiation protection was established in 1986 as an Act of Parliament on the peaceful use of atomic energy (Atomic Act). It was updated in 2000 to internationally accepted basic nuclear and radiation safety requirements, together with special attention given to achieve full compliance with European Union directives as a prerequisite to Poland's accession to the European Union. This Act entered into force on 1 January 2002, and defines duties and responsibilities of the National Atomic Energy Agency (NAEA), its President and its relations to other governmental bodies. On this basis, several governmental decrees and regulations have been issued.

The Act and the regulations created an adequate legal framework and regulatory infrastructure, oriented to solve the nuclear and radiation safety problems related to research reactors and the application of radiation sources, and also to nuclear power programmes in future. General procedures of licensing of nuclear installation (research reactors, radioactive waste and spent fuel management facilities), in the phases of construction, commissioning, operation, decommissioning or closure are illustrated in Fig. 1. An applicant for a licence must provide full documentation for the nuclear facility, which includes the technical description and nuclear safety analysis; quality assurance (QA) programmes; detailed procedures of operation; instructions; and manuals. On the basis of the review and assessment of documentation, and also on inspections performed by NAEA regulatory inspectors, the President grants a licence for a fixed period of operations under specified conditions, if necessary.

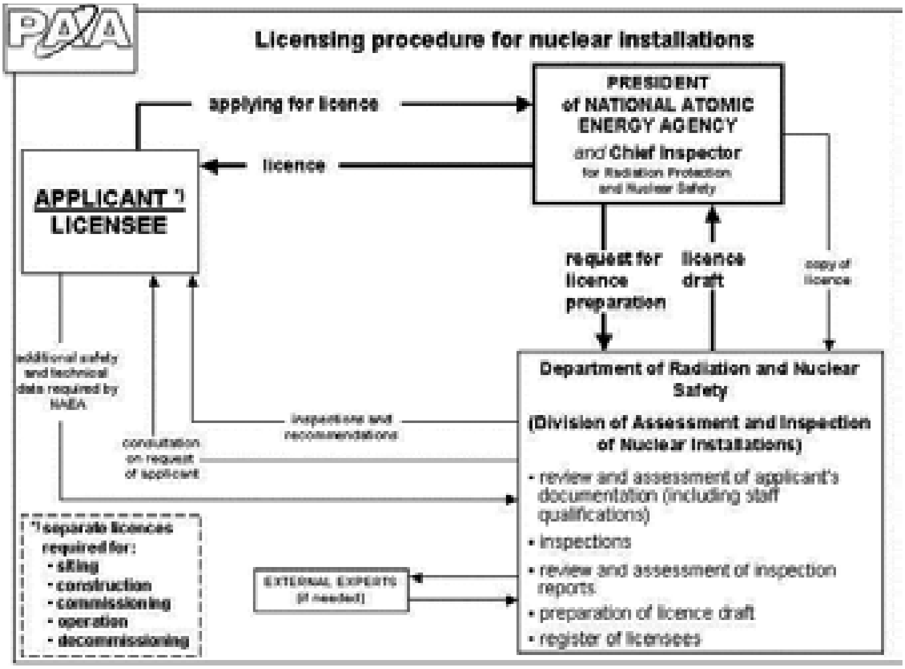


FIG. 1. Licensing procedure for nuclear installations.

Another task of the NAEA is the quarterly presentation of the radiation situation based on radioactive contamination and actual gamma dose rate radiation measurements obtained from 14 stations, which is available on the NAEA web site (www.paa.gov.pl) and shown in the Appendix.

3. REGULATORY PROCEDURES

According to the Atomic Energy Act, the regulatory body's responsibilities include, in particular, the inspections of nuclear facilities. The regulatory inspectors are entitled to:

(1) Visit, at any hour of the day or night, nuclear installations;
(2) Examine documents dealing with nuclear safety and radiation protection;
(3) Verify activities, requiring permits, and their compliance with nuclear safety provisions, etc.;

(4) Undertake, as necessary, independent technical and dosimetric measurements.

The regulatory body has adequate powers to enforce compliance with safety requirements imposed by laws, regulations and licence conditions. Depending on the regulatory assessment of a particular situation, the following enforcement actions can be undertaken:

(1) Issuance of a written warning or directive to the licensee;
(2) Ordering the licensee to curtail activities;
(3) Suspension or revocation of the licence;
(4) Financial penalty collected by means of administrative execution proceedings, punishment by fine or detention;
(5) Recommendation of prosecution through the courts of law.

The regulatory inspectors have an authority to take on the spot decisions in a situation when safety of an installation or personnel is threatened.

4. NUCLEAR INSTALLATIONS IN POLAND

According to the Atomic Energy Act, the following nuclear installations are defined: research reactor, spent fuel and radioactive waste repositories and supervised by regulatory body. All research reactors ever operated in Poland were located in the Institute of Atomic Energy at Swierk near Warsaw (35 km away) and their list is shown in Table 1 together with their present status.

TABLE 1. RESEARCH REACTORS IN POLAND

Research reactor	Thermal power (kW)	Type	First criticality	Present status
EWA	10 000.0	Tank	1958	Shutdown 1995
ANNA	0.1	Channel	1963	Decommissioned 1977
MARYLA	100.0	Pool	1967	Decommissioned 1973
AGATA	0.1	Channel	1973	Shutdown 1995
MARIA	30 000.0	Channel/pool	1974	Operational

The first research reactor of the WWR-type in Poland, named EWA, was put in operation as early as 1958 and shut down 37 year later in 1995. It was designed and delivered from the Soviet Union. A safety analysis report, training of personnel and critical experiment were fully conducted by a contractor. The decommissioning of the reactor is practically finished, fuel was unloaded, all the core internals, the reactor tank and all elements of the primary and secondary cooling systems have been dismantled, removed and stored in a radioactive waste depository, a cooling tower was also disassembled. The general permission for decommissioning was issued by the regulatory body in 1997, but every important step was separately authorized by the regulatory body and performed under special supervision.

The three zero power reactors (critical assemblies) were designed and constructed by Polish specialists in the 1960s and 1970s. They were used for experiments in reactor physics as a training facility, and one was used as a mock-up of the second Polish reactor. Later, they were shut down and two of them are decommissioned.

The second research reactor, MARIA, was of original Polish design but based on a similar Soviet MR-type reactor for loop experiments and with more provisions for physical experiments using neutron beams. It is a high flux reactor operating at a thermal power of about 15 MW, with individually cooled fuel channels located in an open water pool. Its special construction enables simple installation of technological loops, but now it is used mainly for isotope production and research in physics. The reactor was put into operation in 1974, the main reconstruction was done in the period 1988–1993, and is now fully operational. The main technical parameters are given in Table 2 and the cross section of the reactor core and spent fuel storage is presented in Fig. 2.

5. REGULATORY PRACTICE

The regulatory practice is a very important part of regulatory body duties and in our case is based on 45 years' experience of operation of up to five research reactors in the past, but at prezent, it is applied to only one operating reactor. It is limited in this paper to the research reactor, and consists of the following regular activities:

(1) Review of quarterly reports of reactor operation.
(2) Regular inspection.
(3) Granting permission to make any changes in reactor core configuration.
(4) Granting permission to perform any experiments in reactor operation.

TABLE 2. MAIN CHARACTERISTIC PARAMETERS OF THE MARIA REACTOR (2003)

Technical data	Reactor type	Pool
	Thermal power (kW)	Up 30 000
	Max. thermal neutron flux (n/cm^2·s)	4.5×10^{14}
	Max. fast neutron flux (n/cm^2·s)	1.0×10^{14}
	Moderator	Be, H_2O
	Coolant	Light water
	Reflector	Graphite, H_2O
Experimental facilities	Number of horizontal channels	6
	Horizontal channel max flux (n/cm^2·s)	3.0×10^{13}
	Number of vertical channels	13
	Vertical channel max flux (n/cm^2·s)	4.5×10^{14}
	Vertical channel use	Isotope production
Fuel data	Type of element	6 tubes
	Fuel material/enrichment (%)	UO_2/36
	Burn-up average (%)	35
	Fuel fabricator	Russian Federation
Utilization	Hours per day/days per week/ weeks per year	24/5/38 (in 2002)

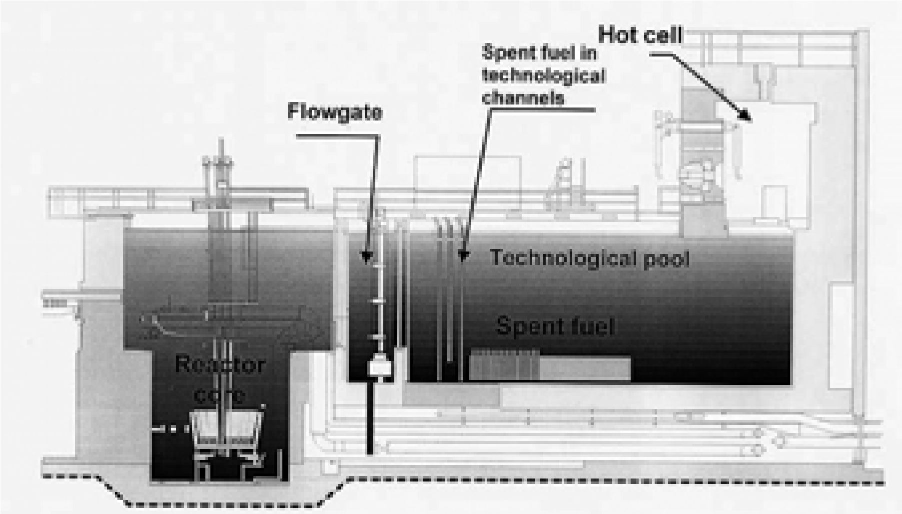


FIG. 2. Vertical cross-section of the MARIA reactor.

(5) Granting permission to perform non-typical irradiation, and other action undertaken in the past few years.
(6) Performing special inspections, e.g. after refurbishment of neutron flux measuring channels, in case any problems arise with irradiation, cooling problems of control rods, etc.
(7) Granting general permission for reactor operation (operating licence) generally under specified conditions.
(8) Granting permission for conversion from 80% to 36% enriched fuel elements.
(9) Granting permission for typical irradiations.
(10) Granting permission for the encapsulation of spent fuel.

All the activities mentioned will be described in detail showing the role of the regulatory body.

5.1. Quarterly reports

Since the restart of the MARIA after modernization in 1995, regular quarterly reports are required by the regulatory body. The aim of these reports is to give a detailed picture of the reactor's operation and to serve as a base for the evaluation by the regulatory body. An example of general performance indicators applied for the MARIA reactor for the last three years is given in Table 3. It can be seen that all indicators are stable, only collective dose increased twice in the last year, caused by the lower quality of new fuel. The appropriate step was already undertaken and the next part of the fuel will have a thicker cladding.

TABLE 3. PERFORMANCE INDICATORS FOR THE MARIA REACTOR

Year	Work time (h)	Availability factors		Unscheduled shutdowns	Employees	Collective dose (man-Sv)
		Total	Per year			
2000	3748	99.0	43.0	5	52	0.085
2001	3580	98.0	40.0	10	56	0.094
2002	3814	99.5	44.5	6	58	0.170

5.2. Remarks

The following definitions of availability factors were used:

— Total ratio of work time to work time plus number of hours of unscheduled shutdowns;
— Ratio of work time per year to number of hours in a year.

Collective dose is based on personal dosimetry (monthly or quarterly readings), excluding natural background (all results below 0.4 mSv, which is the limit of film dosimeters, was set equal to 0.2 mSv).

The analysis of four quarterly reports of 2002 (typical operational chart with 38 fuel cycles can be seen in Fig. 3) done by the regulatory body gives the following observations:

(a) The reactor was operated without any significant problems for all planned 38 cycles;
(b) Reactor power was lower than nominal but adjusted for irradiation needs;
(c) Neutron beam (horizontal channels) were used for physical experiments with utilization up to 48% of reactor operation time;
(d) All revisions and maintenance works were performed according to a schedule;

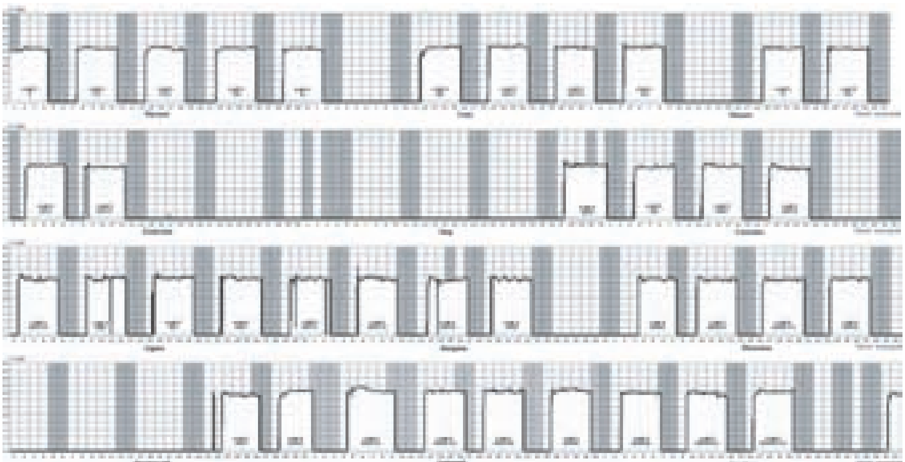


FIG. 3. Operation chart of the MARIA reactor in 2002.

(e) There were six unplanned shutdowns due to:
- — Break in electrical supply systems (five times);
- — Leakage caused by failure of the sealing of one pump of the fuel channel cooling system but only one caused shortening of the fuel cycle due to leakage; in all other cases, the reactor was started within 20 min.

In conclusion, it may be stated that the regulatory body, since the start of operation in 1995 after reconstruction, has no significant remarks concerning reactor operation.

5.3. Regular or special inspections

The list of inspections performed is given in Table 4. They are divided between regular inspections: normally one or two per year and a special one connected with some situation reported in quarterly reports or in special communication to the regulatory body. Regular inspections are generally devoted to some procedures or working conditions of a specified department. In this table they refer to: .

(a) Procedure of fuel conversion;
(b) Fulfilment of requirements in the irradiation process;
(c) Keeping documentation concerning the design of equipment (many drawings are successively transferred to electronic form) and operation (keeping logbooks);

TABLE 4. LIST OF INSPECTIONS OF THE MARIA REACTOR

Date	Number	Type	Scope
18 May 1999	1/99	Special	Definition of thermal power
14 December 1999	2/99	Regular	Procedures for fuel conversion
19–20 June 2000	1/2000	Special	Condition of pressurizer work
31 October 2000	2/2000	Special	Blocking of irradiated sulphur container
18 December 2000	3/2000	Regular	Irradiation—technology and procedures
16–17 October 2001	1/2001	Special	Management of documents
15–16 October 2002	1/2002	Regular	Inspection of the dosimetric department
Planned	1/2003	Regular	Inspection of the operating documentation

(d) The situation in dosimetric systems and the requirement for upgrading;
(e) Documentation of reactor operation (i.e. the working of a computer program recording technological parameters, vibration diagnostic of fuel channel cooling pumps).

Special inspection in this period was devoted to:

(a) Operation of the pressurizer, especially in off-normal conditions;
(b) Improvement of a procedure for sulphur irradiation, explanation of the cause of container leakage, lower amount of loaded sulphur and introducing an additional check of weight by operating personnel.

All inspections are concluded with a special report and suggestions to be fulfilled within a specified time.

5.4. Operating licences

The last operation licence was issued on 29 March 2001 and is valid until 31 March 2004. It was based on a detailed analysis of the reactor's technical and personal conditions and granted specified conditions. In case any of them fail, the regulatory body has the law to suspend reactor operation. The last of these conditions refers to the verification that the safety analysis report was recently fulfilled.

5.5. Fuel conversion

The conversion from 80% to 36% enriched fuel was one of the main tasks realized in more than three years, from 29 April 2001 to 10 June 2002. The strategic plan of conversion was prepared in detail by the calculation of thermohydraulic parameters of new fuel element, its reactivity worth and position in a core to be first loaded and later moved to a central part of a core. It is required that the limit of reactivity increase after loading was set to $2 and maximal power of a fuel channel set to 1.6 MW(th). The conversion was started with loading of the fuel element equipped with special thermocouples to measure the temperature of the coolant outlet to be compared with calculations. The report of operation of the first element was prepared and evaluated by the regulatory body. It was an essential document for the permission for loading a further five elements. After the successful conversion of five elements, permission was granted for loading further elements, until the conversion process was complete. During this process, regular measurements were performed by the fuel element leakage detection system in order to check the tightness of new fuel elements. The conversion was done in 109 fuel cycles, as shown in Fig. 4 and the new core has only 14 fuel elements, in comparison with 17 elements in the past.

5.6. Isotope irradiation

In order to improve the irradiation procedure and based on many years' experience, the list of typical isotope irradiations was prepared. The authorization for typical irradiation specifies the following: name and amount of material, container, chemical characteristic, neutron flux, time of irradiation, heat generation in a sample, analysis of reactivity effects, procedures for QA and the emergency procedure. Typical irradiations include sulphur, potassium chloride, tellurium dioxide, cobalt, iron, potassium bromide and samarium. They are realized during each fuel cycle and listed in quarterly reports giving name of material, time and place of irradiation.

5.7. Vibration monitoring system

The new development in research reactors is the vibration monitoring system. The first model of the system was prepared for the EWA reactor in the late 1980s as a requirement of the regulatory body to monitor the acceleration signal, giving information about the vibration of primary cooling pumps and pipes for this rather old reactor. From 1995, the system was moved to the

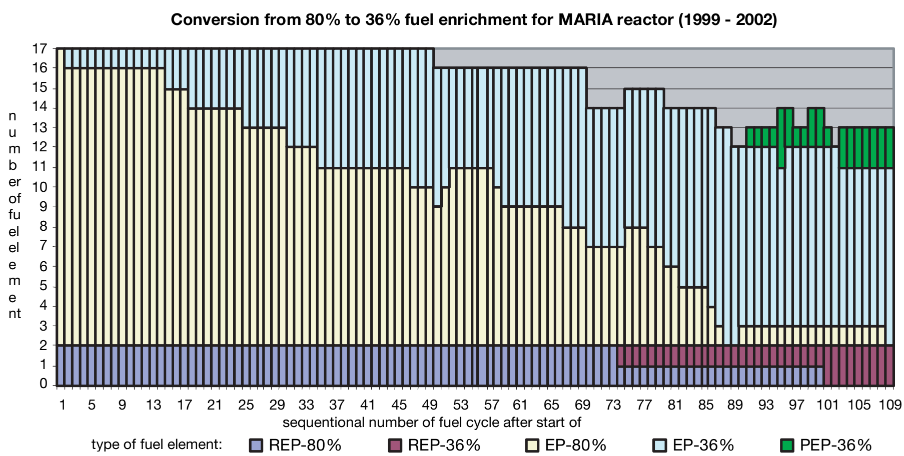


FIG. 4. Conversion of fuel enrichment from 80% to 36% in the period 1999–2002.

MARIA reactor with the enlarging of its capacities. It is used for monitoring vibrations and the temperature of bearing of fuel channel cooling pumps. It operates permanently, measuring these two signals every three minutes and calculating standard deviation (RMS value) of the first one. The RMS and temperature are compared with specified limits as well as in the case where they exceed a warning signal id generated to an operator. The typical plots of both signals are given in following. It can be seen that the RMS value of the acceleration signal (Fig. 5) increased about three times in 27 hours, remained stable but over the next 12 hours (in the morning of the next day), an operator decided to switch off this pump, avoiding probably more severe damage of a bearing in case a pump was operating a full cycle. The bearing temperature (Fig. 6) increased suddenly and very sharply after 56 hours from the beginning of the fuel cycle and the warning signal was generated by a conventional limiting device. But using this diagnostic system, a warning signal based on calculation of a trend of the temperature would be issued at least 12 minutes earlier. A more detailed description of this system is given in [2].

5.8. Encapsulation of spent fuel

The storage of spent fuel elements from both reactors, EWA and MARIA, creates a real problem. They are stored in a water pool in a special storage building (EWA elements) and a technological pool (MARIA elements). The first one already has spent more than 43 years in water and, due to corrosion, a leakage of fission products may happen. Furthermore, the capacity of the technological pool is limited. In this situation, the technology

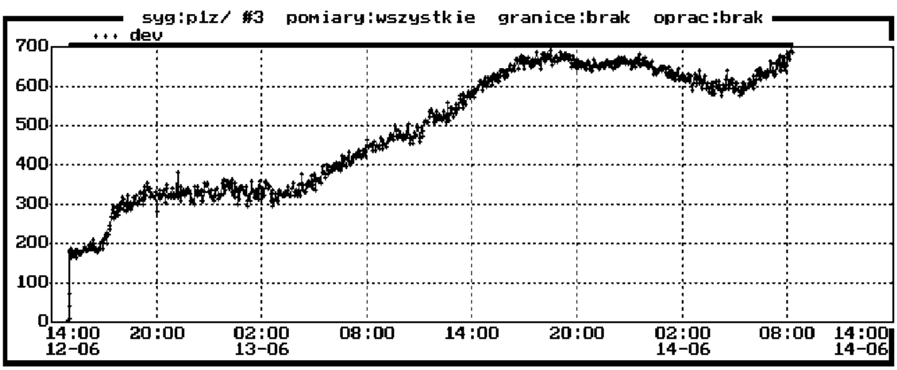


FIG. 5. The RMS of acceleration signal for pump no. 1 during the 17th fuel cycle, in 2000.

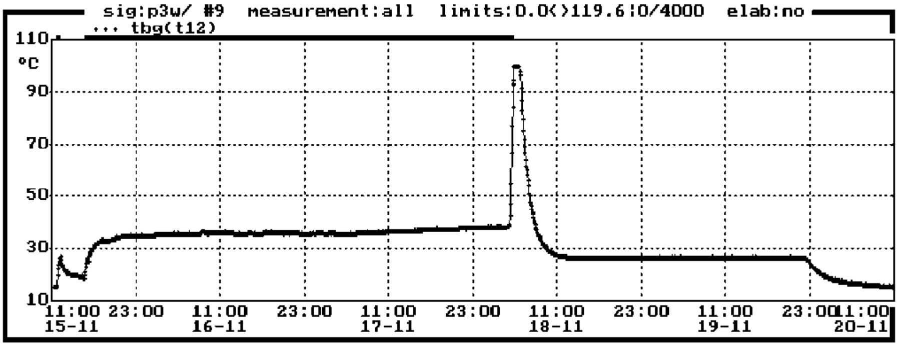


FIG. 6. The bearing temperature for pump no. 3 during the 24th fuel cycle, in 1999.

encapsulation for MARIA spent fuel was prepared. The fuel element is transported to a hot chamber, dried, placed in a container with inert gas and tightly closed. After checking the procedure, it is transported back to the water pool. In parallel, a conception of construction of dry storage in a shaft of the EWA reactor is considered. The process was carefully analysed by the regulatory body and a special licence was granted.

5.8.1. Upgrading of equipment

The process of upgrading of instrumentation has already been started some years ago and consists of:

(1) Fuel channel coolant flow rate measuring system (2001).
(2) New neutron flux measuring system for reactor control and safety systems (2002).
(3) Dosimetric system—upgrading of equipment (in progress since December 2002).

Further new steps in modernizing the reactor operation were required until 2015, in addition to those steps already begun and the purchasing of new fuel elements. The steps may be listed as follows:

(1) Pipelines for a secondary cooling system between reactor building and cooling tower.
(2) Heat exchangers between the primary and secondary cooling system.
(3) Graphite and beryllium blocks.

(4) Power supply system.
(5) Fire protection system.

5.9. Future developments

The activity of the regulatory body in the field of safety cannot be stopped. Knowledge should always be accumulating, and our experience in Poland covers 37 years of operating the EWA reactor and three critical assemblies. For the future, the regulatory body is now working on:

(1) Improving communication between the reactor operator and the regulatory body;
(2) Revising written operational procedures;
(3) Reviewing a new version of the safety analysis report;
(4) Transferring experience from other research reactors to our practice, namely, a PHARE project from the European Union is in progress;
(5) Developing research reactor safety parameter indicators (partially based on similar indicators for nuclear power plants).

The last point seems very promising for the development of rational parameters applicable to assessing reactor operation. Over the last few years, the following three parameters have been evaluated:

(1) Availability factor (total).
(2) Number of unscheduled shutdowns.
(3) Collective dose.

The items mentioned are shown in Table 3, but they are not sufficient in our opinion. They should be appended for a typical research reactor by:

(1) Safety system performance, stating the availability of all standby safety systems (as for nuclear power plants);
(2) Chemistry performance (activity of primary coolant in the case of the MARIA reactor);
(3) Number of failures of irradiation capsules;
(4) Vibration parameters of primary coolant components (pumps, engine, piping, etc).

Work in this field is in progress for the MARIA reactor and may be of interest for other research reactor operators.

6. CONCLUSIONS

The safe operation of research reactors is very important, not only for a particular country but also for the future of nuclear energy in the world. As stated during the seminar entitled Central Europe's Nuclear Challenges, in Warsaw in November 2002, the safe operation of any nuclear installation is of primary importance. Two very important statements on this topic presented during this seminar are cited below:

— "An accident anywhere is an accident everywhere" — T. Taniguchi, Deputy Director General, IAEA;
— "Public opinion will never forget the weak points in activity of a regulatory body" — D. Drabova, Chairwoman, Nuclear Safety Committee, Czech Republic.

The IAEA is promoting the safe operation of research reactors and one of the conclusions of this conference will be to recommend the intensification of work on safety parameter indicators for research reactors performed under the guidance of the IAEA.

REFERENCES

[1] ALCALA-RUIZ, F., et al., "Safety of research reactors", Topical Issues in Nuclear Safety (Proc. Conf. Vienna, 2001), IAEA, Vienna (2002) 217.
[2] MIKULSKI, A.T., Development of vibration monitoring system for MARIA nuclear research reactor, WSEA Trans. Math. **1** 4 (2002).

Appendix

NATIONAL ATOMIC AGENCY PRESIDENT'S BULLETIN OF 16 JULY 2003 ON THE RADIATION SITUATION IN POLAND IN THE SECOND QUARTER OF 2003

In accordance with the provisions in Art. 81 of the Act of Parliament of 29 November 2000 - the Atomic Law (O.J. of 2001, No 3 Item 18, No 100, Item 1085, No 154 Item 1800, O.J. of 2002, No 74, Item 676, No 135, Item 1145 and O.J. of 2003, No 80, Item 717, No 124, Item 1152), hereby given is the following information:

The results obtained from the stations and units performing radioactive contamination measurements were the following:

— Dose rate: 57–133 nSv/h (mean value 89 nSv/h);
— Cs-137 in the air: 0.2–12.1 μBq/m^3(mean value 2.2 μBq/m^3);
— Cs-137 in milk: 0.1–6.5 Bq/dm^3 (mean value 0.7 Bq/dm^3).

Cs-137 content in air and in milk is the basic indicator representing the radioactive contamination of the environmental materials and of the foodstuffs with the artificial (man-made) radioactive isotopes.

Quoted data indicate that the exposure to members of the public with artificial radioactive isotopes existing in the environment and foodstuff remain at a very low level. It poses only several percent of the dose limit for members of the public equal to 1 mSv per year.

This assessment of the radiation situation is in no way affected by an increased level of iodine isotope (J-131) content, registered in the air in the south-eastern region on 7-14 April and caused by an incident in the nuclear power plant in Paks (Hungary).

President of the National Atomic Energy Agency
J. Niewodniczaski

SAFETY RE-EVALUATION OF RESEARCH REACTORS
A new trend in licence requirements

J.P. BOOGAARD, H.T. WIERSEMA
Nuclear Research and Consultancy Group (NRG),
Petten, Netherlands
Email: boogaard@nrg-nl.com

Abstract

The Nuclear Research and Consultancy Group (NRG) has more than 40 years' experience in the nuclear field. This includes experience in the operation of a wide range of nuclear facilities, including operation of the 50 MW(th) high flux reactor (HFR) owned by the European Commission and the 30 kW(th) low flux reactor (LFR). Up until the mid-1990s, the operation of most research reactors was covered by an open-ended licence without a requirement for a periodic safety review. The licensing policy for nuclear installations, however, has been changed over the last 10 to 15 years in most western countries. In the majority of new nuclear act licences, a requirement for a periodic safety review is prescribed. In the framework of the re-licensing process for the LFR in 2000, and in the framework of the running re-licensing process of the HFR, a safety re-evaluation for both reactors has been performed by the NRG at the request of the nuclear regulatory body of the Netherlands. IAEA Safety Series No. 35 has been taken as the basis for a safety re-evaluation of the LFR. Since no reference licensing basis was available for a 50 MW(th) research reactor, a licensing basis for the HFR had to be defined. The safety re-evaluation of the HFR was performed in the re-licensing process, which will be completed by the end of 2003.

1. INTRODUCTION

The Nuclear Research and Consultancy Group (NRG) is the knowledge centre for nuclear technology in the Netherlands. The NRG performs independent research and development, provides studies, consultancy and information services for customers in governmental organizations, industry and the general public. Prerequisites for these activities are the safe, ecological and peaceful application of nuclear technology. A significant part of the NRG's work concerns applications in the field of energy supplies and nuclear installations, but also applications in the non-nuclear market, as well as in the medical sector. The NRG was established in 1998 from the merger of the nuclear activities of the Energy Research Foundation (ECN) of the Netherlands and the Institute for Research and Calibration of Electrical Components (KEMA) of the Netherlands. Although the NRG is relatively young, it has inherited a mature knowledge and experience of more than 40 years from both parent organizations, viz., operation of:

— The 45 MW high flux reactor (HFR) of the European Commission for materials testing, (medical) isotope production, and boron neutron capture therapy (BNCT);
— A 30 kW Argonaut low flux reactor for training and biological applications;
— Hot cells for materials testing and isotope separation;
— A decontamination and waste treatment facility for both internal and external services;
— A 1 MW suspension test reactor and its decommissioning.

In addition, the NRG has overall responsibility of a molybdenum production facility.

European guidelines prescribe that conventional industrial activities with a high potential occupational hazard or potential great environmental impact are only being licensed for 10 years. For nuclear act licences, this policy is often implemented as a requirement for a safety re-evaluation every 10 years. As a consequence, the majority of new nuclear act licences of nuclear power plants have a requirement for a periodic safety review every 10 years. The competent authorities of the Netherlands have also adapted this policy for research reactors. Consequently a safety re-evaluation was performed on request of the nuclear regulatory body of the Netherlands in the framework of the re-licensing process of the HFR.

2. SAFETY RE-EVALUATION

During the lifetime of a reactor, degradation of technical systems is normally mitigated by a dedicated preventive maintenance, periodic testing and inspection programme, including regular replacements of systems and components. Last but not least, by upgrading and refurbishment activities by which the technical status of a reactor is kept up to date. In addition to the continuous development of the technical components, (inter)national legislation and regulation is continuing to develop. Examples of such developments are the IRCP guidelines for dose constraints which have been reduced from 50 mSv to 20 mSv; environmental constraints, such as limitations of the allowable releases of heavy metals (Hg, Ni, Cd), radioactive discharges and noise constraints. IAEA requirements and guidelines also developed a more stringent set of requirements.

Consequently, not only technical degradation should be mitigated but also the operational, personnel and administrative degradation should be mitigated too. The objective of a periodic safety review is to identify and implement a set of improvement measures in order to ensure the safe operation of the reactor for the next decade (see Fig. 1).

A periodic safety review or safety re-evaluation should consist of an evaluation of the technical, organizational, personnel and administrative (TOPA) provisions, with respect to nuclear safety and health physics aspects of the operation and utilization of the facility. A safety re-evaluation should be approved by the regulatory body and consists normally of the following stages:

— *Stage 1*: Compliance check of existing licensing conditions.
— *Stage 2:* Determination of the reference licensing basis.
— *Stage 3*: Comparison of the present situation with the reference licensing basis.
— *Stage 4*: Identification of TOPA weaknesses, shortcomings and establishing the safety improvement measures.

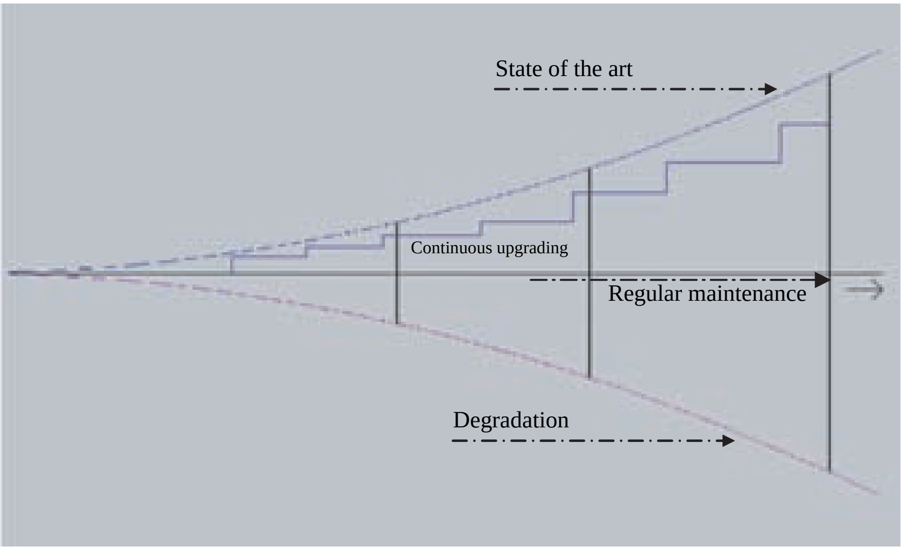


FIG. 1. TOPA development and degradation.

2.1. Compliance check of existing licensing conditions

Although the applicable licensing conditions and related amendments should be fulfilled at all times, a systematic check of those conditions is often not performed on a regular basis. Especially for long standing licences, all relevant licence requirements are often not well known, since they may be obscured in the documentation, such as:

— Safety analyses report and safety analyses performed;
— Operational limits and conditions (technical specifications);
— Additional requirements and agreements with the competent authorities;
— Approval requirements related to modification;
— International legislation;
— Environmental, conventional safety and occupational health requirements.

The following aspects of the operation could be considered in the systematic evaluation of the fulfilment of the present licence:

— Health physics aspects in relation to ALARA;
— Radioactive releases;
— Operational feedback, i.e. lessons learned from incidents, faults and scrams;
— Education and training;
— Organizational changes;
— Reporting of all specific items that are being fulfilled is time consuming and seems to be less beneficial.

The result of this check should provide information on methods of implementation and the fulfilment of the licensing conditions, including an overview of the related operating instructions by which every licensing condition is ensured. More and more, a two-yearly systematic check is incorporated in modern operational licences.

2.2. Determination of the reference licensing basis

The reference licensing basis for the HFR reflects the safety requirements related to the technical status, the operation and the utilization of a modern research reactor with a comparable power level. The following documents serve as the main reference documents for the safety re-evaluation:

— IAEA Safety Series No. 35;
— Nuclear Safety Rules (NVR) of the Netherlands, which originates from IAEA Safety Series No. 50, as far as applicable.

The additional requirements of IAEA Safety Series No. 50 codes are mainly:

— Adaptation of the ‘defence in depth’ concept;
— Design criteria with respect to safety functions and redundancy, severe accidents for emergency preparedness, optimized operator performance, fire protection, emergency core cooling, containment building;
— Review of operation and feedback of experiences;
— Training and qualification;
— Emergency preparedness.

The reference licensing basis has been agreed upon with the regulatory body and the involved competent authorities for formal approval.

In addition to the reference licensing basis, the following documents have also been considered for the safety re-evaluation:

— Existing previous safety evaluations of comparable facilities;
— Accepted good practices elsewhere are to be considered;
— IAEA Safety Series No. 50-SG-O12;
— IAEA-TECDOC-792 (Management of Research Reactor Ageing);
— General insights from relevant PSA and IPE for class I (internal) events performed within the last years on research reactors.

2.3. Comparison of the present situation with the reference licensing basis

2.3.1. Evaluation of deviations from the defined reference licensing basis

During this stage, all safety related TOPA aspects of the installation, operation and utilization were compared with the reference licensing basis, using the specification as established under stage 2 of the safety re-evaluation.

The comparison resulted in an overview of the deviations, including specification of nature/type and a recommendation for improvement from the reference licensing basis. This overview served as an input for the selection process of the modifications and/or improvements to be implemented.

2.3.2. Evaluation of safety related operational experience

The safety related operational experiences of the present installation were evaluated during this stage. Special attention was given to the already implemented improvements in the TOPA field and for routine operations, as well as to incidents and anomalies that occurred during the last decade. Incidents and accidents of similar facilities were also considered in this evaluation.

Typical topics considered in this stage were:

— Safety performance;
— Fuel cycle;
— Health physics;
— Modifications and utilization;
— Management of ageing;
— Fire protection;
— Training and qualification;
— Human factors;
— Experience from other comparable research reactors.

The effectiveness of implementation was also considered in this evaluation. Shortcomings and/or weaknesses serve as additional input for possible improvement measures.

2.3.3. Evaluation of ageing aspects of systems and components important to safety

The reliability of systems and components important to safety and the mitigation of ageing effects of these systems are being evaluated. This evaluation should present the measures to be taken to identify and overcome ageing effects in order to ensure safe operation for the next 10 years. Special attention will be paid to the maintenance and inspection programmes, which should be able to detect ageing effects at an early stage in order to take corrective actions. The systems and components to be evaluated will be limited to systems and components important to safety. The measures identified during this stage will also serve as input for the selection and priority ranking process of the improvement measures. The evaluation of the ageing aspects is not yet completed.

3. RESULTS OF THE SAFETY RE-EVALUATION OF THE HFR

3.1. Compliance check of the HFR licence

The compliance check of the HFR licence did not show any unexpected anomalies. A number of outdated licensing requirements were indicated, such as the following:

— The originally prescribed half-yearly containment test has been replaced by a yearly test with more clearly defined test conditions.
— The originally prescribed copper wire flux measurement prior to every reactor start has been replaced by dedicated core calculations and a yearly verification by copper wire measurements.
— Outdated requirements of the fire brigade have been adapted to modern standards.

These adaptations, which had been agreed with the regulatory body in the past, will be incorporated into the application for a new licence.

3.2. Results of the safety evaluation of TOPA requirements

Although the HFR has been in operation since 1961, the majority of current requirements for a 50 MW research reactor are still met. However, the methodology of the analysis used for the safety analyses report is outdated and not documented in conformance with modern QA standards. The design bases for the HFR was not clearly documented according to modern standards and the postulated initiating events did not follow a consistent structure, as described in the relevant IAEA documentation.

The most important change to the analysis requirements is the inclusion of intermediate and large primary pipe break events including guillotine break. These accidents are qualified as beyond design basis accidents, but for the purposes of the safety analyses, the acceptance criteria of the design basis accidents are used. This approach is justified by the low operating conditions at the HFR (pressure and temperature). The accidents are considered within the design basis accidents as part of the process of minimizing risk. The large pipe break was originally defined as a 20 cm^2 break.

The currently accepted basic philosophy at commercial power stations that no operator intervention shall be needed for the first 30 minutes for accident mitigation is being adopted in the accident analyses for the HFR. During this time, the incident conditions can be reviewed, the diagnosis of the event can be determined and an adequate response plan, including possible manual actions, can be determined.

As a result of the new requirements, a complete set of safety analyses are being performed in the framework of the re-licensing process for the HFR and a new modern safety assessment report is being prepared.

In addition to the deterministic safety analyses, a risk scoping study, consisting of limited level 1 and level 2 analyses and a complete level 3 consequence analysis, have been performed. As a result of the risk scoping study, some additional improvement measures were indicated. The effects of the technical modifications are being analysed with the risk scoping study and will be completed on short notice.

Based on the new insights gained from the severe accident analyses, i.e. (large) pipe break events and the insight from the risk scoping study, the accident management procedures of the HFR have to be adapted. The layout of the panel in the control room will be evaluated as part of the accident management development process.

It was also concluded that the compartment isolation with respect to fire protection could be improved. Measures to separate safety related cabinets, cabling and the uninterruptible power supply systems will be introduced in order to reduce fire prolongation and multiple failure of safety related systems in case of a fire.

Generally, the OPA requirements are satisfactorily achieved in the course of normal operating practices and are addressed in associated documents. However, as with any activity with a long history of cooperation, a thorough review of the conditions of operation as performed for the current licence renewal, reveals various issues requiring further attention. The principal findings are summarized below.

The operation of the HFR involves two organizations that require well defined interfaces regarding tasks, responsibilities and competencies. JRC and NRG interfaces occur at directorate level as well as at the level of the day to day operations. It was concluded that a more detailed description of these interfaces is required. The existing QA systems cover most of the topics relevant for a research reactor. However, due to the requirement that both QA systems of JRC and NRG have to be obeyed, the HFR QA system is unduly complex and does not provide a complete, integrated and coherent set of documents that is easy to be familiar with and to consult. But since NRG will be the sole applicant of the new HFR licence, only NRG's QA system has to be followed in the future and the tasks and responsibilities will be more uniform and transparent.

The link between the licence and the HFR operation is provided by the technical specifications (operational, limits and conditions). These specifications do not follow the consistent application of hierarchical levels of safety parameter settings as described in IAEA Safety Series No. 35-G6; i.e. safety limits, safety system settings and limiting conditions for safe operations. The operating limits and conditions will be systematically introduced based on the recently performed safety analyses. The operating procedures, including the inspection and preventive maintenance procedures, will be adapted to the new set of operational limits and conditions.

Feedback of experience is being carried out based on near misses and malfunctions. A systematic and periodic review of the operating experience (including experiments and maintenance), health physics performance and training/retraining courses) with involvement of the Reactor Safety Committee will be implemented.

Emergency planning generally meets the present requirements. However, the plan is spread over the different QA systems of the JRC and the NRG and is described in a large number of documents. One comprehensive plan for the HFR will be prepared based on the emergency organization of the NRG, with clear interfaces to recently adapted site and regional plans. The emergency procedures will be adapted to mitigate, e.g. large pipe breaks and airplane crashes, and will be combined to a detailed integrated set in order to facilitate operator actions in case of an accident.

Nearly all OPA measures indicated for the HFR will be implemented. The selection process for the technical measures will be completed and agreed with the regulatory body during the coming weeks.

4. CONCLUSION

The nuclear regulatory body of the Netherlands also adapted the policy to perform a periodic safety review for research reactors. Consequently, a safety re-evaluation has been performed for the HFR in the framework of the re-licensing process.

Since no reference was available, a detailed reference licensing basis for the HFR has been defined and agreed with the regulatory body.

Although the HFR has been in operation since 1961, the majority of current requirements for a 50 MW research reactor are still met. However, as with any activity with a long history, a thorough review of the conditions of operation, as performed for the current licence renewal, reveals various issues which have been translated into improvement measures.

Due to the safety review and the implementation of the indicated improvement measures, the safe operation of the HFR will thereby be ensured for the next decade.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Code on the Safety of Nuclear Research Reactors: Design, Safety Series No. 35-S1, IAEA, Vienna (1992).

BIBLIOGRAPHY

NUCLEAIRE VEILIGHEIDSREGELS, Ministry of Social Affairs, The Hague, 1989.

INTERNATIONAL ATOMIC ENERGY AGENCY, Code on the Safety of Nuclear Research Reactors: Operation, Safety Series No. 35-S2, IAEA, Vienna (1992).

INTERNATIONAL ATOMIC ENERGY AGENCY, Periodic Safety Review of Operational Nuclear Power Plants, Safety Series No. 50-SG-O12, IAEA, Vienna (1994).

INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report, Safety Series No. 35-G1, IAEA, Vienna (1994).

INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Guides on Operational Limits and Conditions of Research Reactors, Safety Series No. 35-G6, IAEA, Vienna (1997).

INTERNATIONAL ATOMIC ENERGY AGENCY, Guidelines for the review of research reactor safety, Reference document for IAEA Integrated Safety Assessment of research reactors (INSARR), IAEA Services Series No. 1, IAEA, Vienna (1997).

CHALLENGES IN UPDATING A GENERIC SAFETY ANALYSIS REPORT AFTER 22 YEARS OF TECHNOLOGICAL AND REGULATORY PROGRESS

G.G. KENNEDY
École Polytechnique, Department of Engineering Physics,
Montreal, Quebec
Email: greg.kennedy@polymtl.ca

D.J. WINFIELD
Atomic Energy of Canada Limited,
Chalk River, Ontario

Canada

Abstract

The original Safety Analysis Report (SAR) for the 20 kW (nominal) École Polytechnique Safe Low Power Critical Experiment (SLOWPOKE) research reactor comprised a generic report produced in 1977. This SAR was applicable to all SLOWPOKE reactors constructed in the period 1971–1985. Since that period, IAEA international standards for SARs have changed and been developed into the IAEA's Safety Series No. 35-G1, Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report, intended to be recommendations based on international experience. In addition, national licensing standards for research reactors have become more rigorous since the original generic SAR was produced. The original SLOWPOKE reactor was designed to operate for 20 years and was not intended to be refuelled. After 21 years of operation with the original HEU fuel core, it was planned to convert the École Polytechnique reactor core to LEU fuel in order to extend the life of the reactor for another 30 years. Since this was the first SLOWPOKE reactor to be converted to LEU fuel, preparation of an updated and site-specific SAR was necessary to obtain authorization for refuelling. As part of the SAR update, an independent Operational Safety Review and a review of all safety related LEU refuelling documentation were also performed.

1. INTRODUCTION

The original École Polytechnique Safe Low Power Critical Experiment (SLOWPOKE) [1] reactor generic SAR was used to license the reactor for first criticality with a HEU fuel charge in May 1976, as well as to license seven other SLOWPOKE reactors in Canada and one in Jamaica. When the original SAR was issued, no site-specific information was incorporated into the SAR. Site-specific safety and licensing issues were historically addressed to the regulator in other documentation, outside the generic SAR. By contrast, the updated SAR incorporates, or directly references other documentation where appropriate, a complete range of site-specific and reactor-specific issues. One SLOWPOKE reactor, at Royal Military College, Kingston, Ontario was originally fuelled with LEU in 1985 and continues to be operated. All other SLOWPOKEs have their original HEU fuel charge.

In addition to following the recommendations of the IAEA's Safety Series No. 35-G1 [2], the following safety related activities were performed which provided important information for the SAR production:

- Review of historical experience of other SLOWPOKE reactors (124 reactor years) from annual safety reviews;
- SAR and LEU refuelling activity approval from the Products and Services Safety Review Committee (PSSRC), on behalf of the reactor designer, AECL;
- Development of LEU core transient analysis for comparison with required commissioning transients;
- Operational safety review (OSR) of the existing reactor installation;
- Independent review of HEU de-fuelling procedures and LEU commissioning documentation;
- Verification of analysis predictions with commissioning results and incorporation into the SAR.

The most relevant items from the use of IAEA's Safety Series No. 35-G1 [2] and from these safety related activities are discussed in Sections 3–8.

2. RESOURCES FOR SAR UPDATE

The SAR update work expended about 22 person-weeks effort, not including the transient analysis. An 18-month time period was required for the complete update. This time frame allowed for an internal review process of the SAR, a review by AECL PSSRC, completion of the OSR, acceptance of the

SAR, review of commissioning procedures and acceptance of commissioning data by the regulator, the Canadian Nuclear Safety Commission (CNSC).

Because of the limited complexity and size of the SLOWPOKE reactor, and the detailed generic analysis already available from the original SAR, all the analysis in the SAR was updated using a single analyst, except for the transient analysis. Four reviewers with different areas of expertise were used. The transient analysis results (using the DONJON code) formed part of a longer term programme by staff of the Nuclear Engineering Institute at the École Polytechnique. The transient analysis development work has extended to date, over about seven years, and is a continuing programme.

3. IAEA SAFETY SERIES No. 35-G1

The detailed SAR format provided by Safety Series No. 35-G1 [2] was followed to a large extent in the SAR revision, wherever the format was applicable to the 20 kW reactor design and operation.

A number of changes were made to the scope and format of Safety Series No. 35-G1 [2], and these changes are listed and briefly discussed below. Some additional sections were introduced because some important topics, deemed essential for completeness, were not included in Safety Series No. 35-G1 [2] requirements. Other changes were made because they were not applicable to the 20 kW reactor. Additionally, a number of sections were reformatted into what was judged to be a more logical report structure than defined in Safety Series No. 35-G1 [2].

Changes made to Safety Series No. 35-G1 [2]:

Section 11 Introduction and general description of the facility
Section 107, Experimental Programme.
The 107 title was changed to Current Modifications and Future Research Program in order to incorporate the prime purpose of the SAR update; the LEU refuelling.

Section 2 Safety objectives and engineering design requirements

Section 3 Site characteristics
The order of these main sections 2 and 3 was interchanged, so the respective information is then presented in a more logical order. Section 3 was renamed Safety Philosophy, Design and Operations Criteria, to include general aspects of the design philosophy and also the facility operational and maintenance philosophy.

Section 3	Site characteristics Section 317, Radiological impact; Section 322, Atmospheric dispersion of radioactive materials; Section 324, Dispersion of radioactive materials through surface waters and ground water; Section 326, Mitigation; Sections 317, 322, 324 and 326 are all safety analysis section related subjects, inappropriate in Section 3. These sections were relocated to the main safety analysis section (Section 16 in Safety Series No. 35-G1).
Section 2	Safety objectives and engineering design requirements Section 211, Design for internal fire protection: Section 211 is relocated to the safety analysis section (Section 16 in Safety Series No. 35-G1). Fire is only one of a number of potential internal events, not requiring highlighting under this Section 2 topic. An important subsection, Responsibility, was added to this renamed Section 3 Safety Philosophy, Design and Operations Criteria. The intent is to identify the reactor owner, maintenance and operation responsibilities and any other legal responsibilities of the facility.
Section 4	Buildings and structures This main section was re-titled Building Structure, Services and Supplies. The intention here is to include support systems, such as process water, air supplies, electric power supplies, fire protection, HVAC, gas supplies, liquid drainage systems in a systematic way, not currently documented in Safety Series No. 35-G1. In this regard, Section 9 Electric Power in Safety Series No. 35-G1 is relocated into a subsection of Section 4, as electric power is only one of the numerous service systems in a facility. The incorporation of all the various non-nuclear supply and service systems is judged particularly important to include in the SAR.
Section 5	Reactor This section was re-titled Reactor and Auxiliary Systems and all the reactor systems described therein.
Section 8	Instrumentation and Control Three additional subsections were added here, considered important from an operational aspect: Automatic Operation, Manual Operation, and Switching Between Manual and Automatic Operation.

Section 9 Electric Power
As noted above, electric power is more logically included in Section 4 with all the rest of the building service systems.

Section 10 Auxiliary Systems
This section is removed, as the systems information is more appropriately incorporated into either Section 4 Building Structure, Services and Supplies, or Section 5 Reactor and Auxiliary Systems.

Section 11 Reactor Utilization
This main section was deleted and relocated as an introductory subsection in Section 1 entitled Current Modifications and Research Program.

Section 12 Operational Radiological Safety
Section 1229, 1230, 1231 and 1232 Waste Management Systems: Subsections 1229 to 1232 (Solid Waste, Liquid Waste and Gaseous Waste) are designated under a new major section entitled Waste Management Systems, to highlight their importance. Spent fuel handling and handling of other active components/materials are also now included.

Section 13 Conduct of Operation
Two subsections added are Staff Responsibilities and Authority, and Reactor Staff Liaison with Other Groups.

A main section was included after the Conduct of Operation section: Review of Past Operational Experience. Subsection topics comprised Operational Safety Record (of the École Polytechnique and from generic SLOWPOKE experience) and facility dose and activity release statistics.

Section 16 Safety Analysis
The safety analysis subsections of Safety Series No. 35-G1 were changed to comprise:

- 10.1 Reactor Characteristics (power transient and thermal-hydraulic analysis is contained within this section)
- 10.2 Initiating Event Analysis
- 10.3 Internal Events Analysis
- 10.4 External Events Analysis
- 10.5 Anticipated Operational Occurrences
- 10.6 Summary

Most of the detailed technical content specified in this main section of Safety Series No. 35-G1 safety analysis was still retained, but in detail appropriate for this small size facility.

4. OPERATIONAL SAFETY REVIEW

A list of items identified by the operational safety review is provided. All the items in the following list were addressed before the SAR was submitted to the CNSC and design or operational changes were made where appropriate:

— A reactor room ventilation flow indicator was recommended, and this was installed;
— A natural gas supply line into the reactor room was removed;
— An internal review process, involving facility staff and management, was recommended for the reactor annual safety review, before submission to the CNSC;
— The reactor room electrical service panel was redesigned to satisfy electrical code requirements;
— A review was made of the potential for coincident flooding of the in-core irradiation tubes.

5. SLOWPOKE HISTORICAL OPERATIONAL EXPERIENCE REVIEW

The historical annual safety review reports of all Canadian SLOWPOKE reactors were assessed to identify that any historic generic safety issues that might be applicable to the LEU fuelling and subsequent reactor operation. The annual safety review reports are submitted to the CNSC.

No significant issues were identified from a total 115 HEU and 9 LEU years of operating experience.

6. REVIEW OF LEU COMMISSIONING DOCUMENTS

The following documents were reviewed to provide an independent safety review role, in addition to incorporating relevant information into the SAR where appropriate:

— HEU de-fuelling procedures
— LEU refuelling procedures
— LEU approach to first critical procedures
— LEU Commissioning Report Results
— LEU refuelling QA Plan

7. SAR AND LICENSING APPROVAL

After completion of the first version of the SAR and submission to the CNSC, the reactor was approved for LEU refuelling in July 1997, which then took place in September 1997. Commissioning tests, including full power and power transient tests, based on predictions in the SAR, were completed by the end of September 1997. The power transient model predictions were satisfactorily confirmed by the transient tests.

Following completion of all the commissioning tests, a revised version of the SAR was prepared and issued to the CNSC. The post-commissioning version of the SAR incorporated all safety relevant commissioning report results, particularly those regarding the power transient tests and the validation of the transient model code. The CNSC then provided final acceptance of the revised SAR in June 2000.

8. POST-LEU FUELLING ACTIVITIES

In the weeks and months of full power operation subsequent to the refuelling, a post-commissioning issue that was not anticipated by the SAR became evident. Larger amounts of fission product activity in the reactor water than had been predicted in the SAR were seen in water samples. The source of this increased activity was traced to the presence of trace U-235, leaked from the previous HEU fuel and plated out over the years onto the reactor structural components. The HEU fuel originally used was known to leak increasing amounts of fission products, a factor of 110 increase over 20 years, but it was not known that it had leaked traces of U-235. The reactor water activity level after LEU fuelling decreased by a factor of 440 for noble gas fission products such as Xe-133 and by a factor of 20 for solid fission products. The higher than predicted reactor water activity was identified, analysed and explained in subsequent communications with the CNSC.

The reactor has been successfully operating on the LEU fuel charge since October 1997. The new core is designed for a 30 year operating life. Since refuelling, the reactor has operated a total of 102 000 kW hours on the LEU fuel (October 1997 to end September 2003). The LEU fuel performance has been satisfactory. The current reactor capacity factor is very high for a university research reactor. In 2003, for example, from 1 January to 30 September, the reactor has been operated at or close to nominal power, 75% of the total 40 hour work week. The operating program is dominated by neutron activation analysis, mostly for university research projects, but also for the facility’s commercial neutron activation analysis service.

Further revisions to the SAR are not explicitly mandated by the reactor licence but might be anticipated at periods between 5 and 10 years, or at a time when any major modification, requiring significant licence amendment, was planned.

9. CONCLUSIONS

The LEU fuelling has successfully demonstrated that the original HEU fuelled SLOWPOKE reactors can be refuelled with LEU, with insignificant dose impacts on employees or activity release to the environment. A significant reduction in pool water activity was achieved with the improved fuel manufacture used for the LEU fuel, compared to the original HEU fuelled core. With the LEU fuel, the reactor is achieving a very high capacity factor, satisfying the experimental and commercial activities of the École Polytechnique Engineering Physics Department. Improvements to the reflector configuration are also designed to result in a core lifetime of 30 years.

REFERENCE

[1] KAY, R.E., STEVENS-GUILLE, P.D., HILBORN, J.W., JERVIS, R.E., SLOWPOKE: A new low cost laboratory reactor, Int. J. Appl. Radiat. Isot. **24** (1973) 509.

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Code on the Safety of Nuclear Research Reactors: Design, Safety Series No. 35–51, IAEA, Vienna (1992).

ISO 9001 AND ISO 14001: AN INTEGRATED QUALITY MANAGEMENT SYSTEM FOR AN MTR FACILITY
SAFARI-1 research reactor

J.F. DU BRUYN, C.S.B. PIANI
SAFARI-1 research reactor,
South African Nuclear Energy Corporation (Necsa),
Pelindaba, South Africa
Email: safarjfb@necsa.co.za

Abstract

The SAFARI-1 research reactor, owned and operated by the South African Nuclear Energy Corporation (Necsa), initially obtained ISO 9001 accreditation of its Quality, Health, Safety and Environmental (QHSE) management system via international affiliation from the South African Bureau of Standards (SABS) during 1998 and re-certification according to ISO 9001 (2000) in 2003. With ever-increasing demands on nuclear facilities to demonstrate conformance to environmental policies, SAFARI-1 has now developed an Environmental Management System (EMS) that is compliant with ISO 14001 (1996) and is fully integrated with the SAFARI-1 Quality Management System (QMS). The dynamic involvement of SAFARI-1 in commercial applications demanded that any transition of the original QMS to a fully incorporated QHSE system had to be done in a way that would ensure sustained delivery of a safe and reliable service with continuous quality. At the same time, the primary vision of operating a facility under an efficient financial management programme was essential. The criteria established by the original ISO 9001 compliant QMS were appraised against the additional requirements of ISO 14001 and a suitable superstructure derived for generation and implementation of an inclusive EMS. The transitional integration of this system was planned so as to produce a QMS suitable to quality, environmental and other management related issues for application to the unique function of a nuclear research reactor.

1. INTRODUCTION

The South African Fundamental Atomic Research Installation (SAFARI-1) research reactor is a 20 MW MTR based on the ORR design, i.e. a tank in pool-type reactor. It is owned and operated by the South African Nuclear Energy Corporation (Necsa) on behalf of the Department of Minerals and Energy and is located at Pelindaba, about 40 km south-west of Pretoria.

The major utilization of the reactor, which first went critical on 18 March 1965, is for the production of radioisotopes for medical application (national and export), as well as for the production of Neutron Transmutation Doped (NTD) silicon in the pool-side facility. There are also pneumatic and fast pneumatic systems utilized for Neutron Activation Analysis (NAA). Utilization of beam-ports for institutional (academic and industrial) purposes is encouraged and neutron diffraction and neutron radiography facilities are operational.

The SAFARI-1 research reactor has, since initial operation, applied a management system which was primarily focused on the technical design and safe operation of the plant. The informal system was never developed to comply with a specific code of practice. Since the early 1990s, the challenge was to return to the international arena and to change the employees from a Monday to Friday research culture into a more complex commercial culture, operating the plant 24 hours a day at a fixed operating schedule so as to meet customer and stakeholder requirements. The finding of a Government evaluation in 1997 was clear: “be commercially viable, at least 67% self-reliant or close down”. Under such circumstances, the decision was made, as part of a systematic strategic plan, to implement a formal quality management system in accordance with the ISO 9001.

Due to commercial, regulatory and legislative pressure, as well as anticipated public interest, SAFARI-1 more recently decided to also implement an ISO 14001 Environmental Management System (EMS). The idea from the start was to integrate the ISO 9001 (QMS) and ISO 14001 (EMS) systems with existing corporative licensing requirements, such as radiological and conventional safety, physical security and human resources. SAFARI-1 took the initiative in Necsa—and in South Africa—to implement the ISO 14001 code of practice within a nuclear facility. During the implementation period, guidelines and support at corporative level were virtually non-existent. Assistance elsewhere in the country was also limited, since training courses cater for the more general industry in South Africa.

2. IMPLEMENTATION OF THE INTEGRATED QUALITY, HEALTH, SAFETY AND ENVIRONMENTAL MANAGEMENT SYSTEM

SAFARI-1, the nuclear research reactor located at Pelindaba, has to meet a multitude of requirements relating to corporate policies, strategic planning, quality, conventional and radiological safety, regulatory specifications, security, commercial and financial goals, etc., which need to be incorporated into an overall management system. The control of these various disciplines within the nuclear facility becomes quite complex if the procedures are to be coordinated, maintained and, at the same time, managed so as to achieve suitable levels of staff motivation, which will ultimately ensure appropriate implementation. This becomes particularly relevant where the responsibility of implementation lies within various departments, as is the case with Necsa.

SAFARI-1 is currently only one of a few research reactors to internationally have achieved ISO 9001 (2000) and ISO 14001 (1996) certification. These codes incorporate an interdisciplinary approach, which can integrate all the above organizational requirements, with the emphasis on radiological safety and at the same time encompass a more customer driven focus—this includes customers both internal and external to the SAFARI-1 organization. Such an integration of a QHSE management system providing effective implementation at all levels (system procedures, manuals, legislative and licensing requirements, work instructions, etc.) has now been satisfactorily achieved at SAFARI-1. Control measures have been implemented (audits, cost systems, training, non-conformances, customer complaints, etc.) to verify effectiveness and continuous improvement of the incorporation and implementation of the QHSE management system.

Since such an integrated QHSE management system should provide the individuals or groups with the necessary tools, support and encouragement to carry out the variety of allocated responsibilities properly, SAFARI-1 would like to share this experience gained during the past five or six years. In particular, the methods of integrating QHSE and other systems into an overall management system will be demonstrated to have played a major role in the excellent levels of safety (radiological, environmental and conventional) achieved by the facility in an era of ever increasing commercial demands.

Initially, the ISO 9001 system was developed and established, serving as the basis of the system as it is today. The installation of the QHSE system was planned step by step and was developed, as follows:

— Start of implementation of ISO 9001 (1994) procedures in November 1996;
— Application for certification in February 1998;
— Obtain certification of ISO 9001 (1994) through the South African Bureau of Standards (SABS) in October 1998;
— In 1998, the SAFARI-1 Quality Management System was included into the facility licence by the National Nuclear Regulator;
— Start integration of the system with the Necsa corporative radiological and conventional health and safety management system in 1999;
— Continuous revision of the system to include environmental management requirements (radiological safety requirements only);

- In January 2001, transition to the ISO 9001 (2000) and implementing of the ISO 14001 (1996) was initiated in parallel;
- In September 2002, SAFARI-1 applied for ISO 14001 (1996) certification;
- ISO 14001 (1996) Stage 1 audit was conducted in November 2002;
- ISO 9001 (2000) certification was awarded in May 2003 by the SABS;
- ISO 14001 (1996) Stage 2 and legal audits were conducted in July 2003;
- ISO 14001 (1996) certification to be awarded in November 2003.

3. HOW DID SAFARI-1 IMPLEMENT THE QHSE MANAGEMENT SYSTEM?

The following guidelines were followed through the various stages of implementation:

- A ‘champion’ within senior management must support the process of implementation and the quality representative appointed by management must be competent and committed to drive the project;
- Management and employees must be committed to support the project. Seek continuous support, as management and employees sometimes do not see the value of the system;
- Select a team representing management and employees to help with the preparation of procedures and operational documentation;
- Submit regular feedback to management and employees when a major goal is reached and make an event of it;
- Overcome resistance by removing barriers;
- Allocate clear responsibilities to those who are responsible for the activity. This creates ownership for the individual and groups of employees;
- Schedule regular buy-in sessions and training sessions at all levels to inform and induce personnel when new system requirements are introduced;
- Plan each phase of implementation carefully to fit into other schedules and keep a status list with allocated responsibilities to measure progress of the implementation plan;
- Have regular meetings (once every two weeks, as applicable), in particular, during the first implementation phase, of level 1 and 2 procedures. Discuss the comments in detail, this will also serve as a training session for those involved in the implementation;

— Awareness training in both the ISO 9001 and ISO 14001 is essential for employees at all levels;
— Do not change the natural way of doing things in an organization but let those responsible for the task decide how they want to perform the activity and put this into writing;
— Evaluate conformance to the code of practice (ISO 9001 or ISO 14001) by conducting internal evaluations and implementation audits;
— Determine the scope of the environmental management system and ensure that the correct legislative and regulatory requirements are identified. Obtain professional support in this regard;
— Ensure that emergency and security planning are part of the system requirements;
— Prepare a legal register containing a list of all applicable legislative and licensing documents, referencing the Act(s) and applicable sections;
— Identify all environmental aspects and impacts very carefully (for both conventional and radiological waste types). Evaluate the impacts on the environment by using a quantifiable rating system, which includes probabilities and significance of an event and operational control measures. Impacts with high rating must be controlled through an environmental management programme;
— Identify quality and environmental objectives and targets. They should be realistic, achievable and measurable;
— Overcome barriers or setbacks as soon as possible by maintaining a determination to achieve targets set;
— Remember that the key to success is through the employees. Invest in their training and continuously motivate them;
— The approach of the plan must be selected correctly for the situation. Although there is not generally an incorrect way, unique situations require a unique plan to implement a QHSE system and to blend the culture within a specific organization;
— Select a good and competent registrar. Auditors are not always familiar with the nuclear industry: this could lead to some confusion during an audit.

4. INTEGRATING THE HEALTH, SAFETY AND ENVIRONMENTAL REQUIREMENTS INTO THE QUALITY MANAGEMENT SYSTEM

In order to achieve success of an integrated QHSE system, ensure thorough inclusion of the requirements into related document priority (tier) levels. Do not prepare separate procedures or work instructions for every discipline (see Fig. 1). Due to the commercial, regulatory and legislative control measures involved, a QHSE system should contain at least the following top tier documents:

— Management responsibility and policies for both quality and environmental activities, setting of objectives and targets through well established strategic planning;
— Design and project control for plant modifications and upgrades (for both commercial and safety related activities);
— Document control and maintenance of quality records;
— Product realization, which includes production planning, operational control, conventional and radiological safety measures; product and/or equipment validation and customer feedback;
— Measurement analysis, inspection and testing;
— Calibration of measuring and test equipment, including radiological and maintenance instrumentation;
— Control of non-conformities and continual improvement;
— Internal audits and technical audits, as well as legislative assessments;

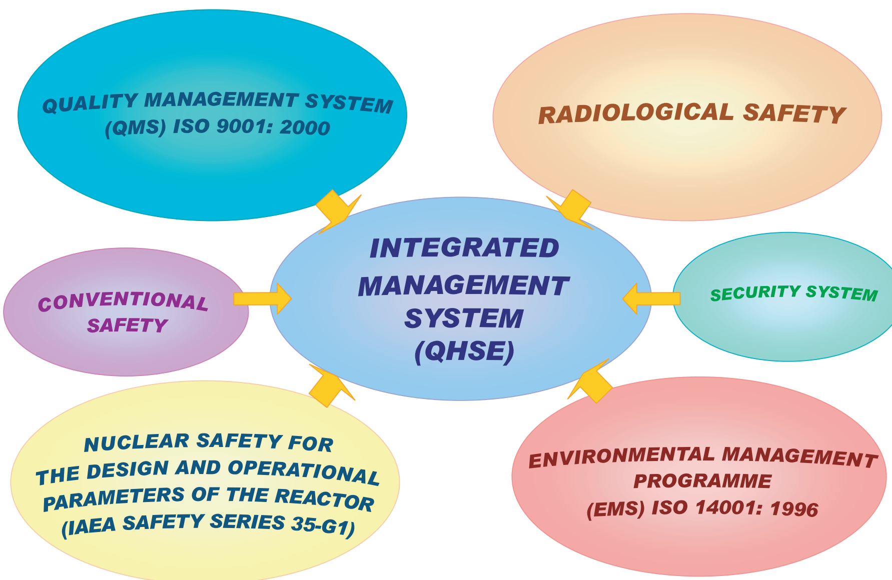


FIG. 1. Various disciplines of a QHSE management system.

— Training and authorization, as well as identification of competency levels in accordance with regulatory and legislative requirements;
— Facility, safety and regulatory control measures must be in place for aspects such as:

- Safety analysis;
- Licensing requirements;
- Safety committees for facility development and upgrades to equipment for both commercial processes and safety purposes;
- Decommissioning and decontamination;
- Radiological and conventional safety aspects;
- Accountability of fissile material (safeguards);
- Maintenance and in-service-inspection;
- Environmental management.

5. DIFFICULTIES AND PITFALLS

Selection of a registrar is very important, especially should your facility be the sole nuclear industry in the country. In particular, an incompetent registrar could be problematic for a satisfactory ISO 14001 certification audit.

Auditing an integrated system is a new challenge and, therefore, coaching of the auditor is often necessary to obtain satisfactory results and to maintain a good relationship with the registrar. Where integrated audit skills are well implemented, such facilities should utilize this for better efficiency and financial outlay. This will also make life easier and could reduce the amount of audits per annum.

The foundation of a good EMS requires an effective mechanism to identify environmental aspects and impacts. These should be identified correctly, according to operational processes or operational areas.

The integration of QHSE systems should be evident throughout all levels of the documentation (i.e. have a section for operational, quality, environmental, radiological and conventional safety, emergency and safeguards).

Legal assessment audits could be problematic where regulating, legislative and corporative requirements and responsibilities are not clearly defined. Services of a legal expert should be acquired who is familiar with the nuclear and waste management policies, to assist in the preparation of the legal register.

A schedule with target dates should be maintained, while regular internal evaluations and management reviews should be performed to verify the effectiveness of the system. Efforts should not only meet, but should surpass stakeholders' expectations.

6. WHY IMPLEMENT A QHSE MANAGEMENT SYSTEM?

Following the previous discussion, the value of implementation of a management system encompassing quality, health, safety and the environment is evident and provides the following benefits:

— Opens new commercial markets and creates stakeholder confidence;
— Is cost efficient and could render better income for quality products;
— Ensures that a satisfactory safety and commercial culture is established;
— Provides for emergency planning;
— Warrants conformance to regulatory and legislative requirements;
— Structures and facilitates training;
— Makes continual improvement evident;
— Meets customer needs/concerns/complaints satisfactorily;
— Establishes quality and environmental goals and objectives, and monitors them to satisfy strategic plan requirements;
— Makes responsibilities clear to all;
— Creates a unique culture (customer and facility/organization needs).

7. CONCLUSION

An integrated management system involving quality, health, safety and environment, as well as various corporative, regulatory and international requirements, must necessarily be developed in conjunction with other management systems and should include all disciplines from human resources to decommissioning and decontamination. Management systems do not originate by themselves and must be carefully planned and implemented to ensure that they function satisfactorily. Such a transition from a more simplistic management system to an integrated QHSE system requires intensive investment in the quality of education and training, quality of behaviour, quality of thinking and quality of decision making to enhance company culture so as to successfully implement such an integrated QHSE system.

In conclusion, the challenge today is: “Aim to be a leader in the field, be better than your competitor.”

REACTOR PROTECTION SYSTEMS
Diverse approaches

C.A. VERRASTRO, D.S. ESTRYK, J.C. CARBALLIDO
Instrumentation and Control Department,
Division of Nuclear Reactor and Power Plants Activities,
National Atomic Energy Commission (CNEA),
Bariloche, Argentina
Email: daestryk@cnea.gov.ar

Abstract

Defence in depth design criteria applied to nuclear instrumentation, in particular, to reactor protection systems (RPS), include redundancy, diversity and fail-safe behaviour. Typically, two out of three ('2oo3'), majority-voting systems meet redundancy criteria. A careful analysis of signal levels and polarity and the use of several techniques, such as lives zeros, bias toward safe state, etc. guarantee the same degree of fail-safe behaviour. Diversity criteria, in general, are met by the whole system using more than one method to protect the integrity of reactor (i.e. rod drop plus boron injection), but not for the single instrumentation chain. Moreover, the increasing information needs of supervision systems encourage the use of digital instrumentation in RPS; if the digital instrumentation has software based implementation, the diversity requirement will be mandatory for the instrumentation of each system. In the paper, three possible configurations of the first protection system (rod drop) are analysed. The first one is the traditional hardware approach, the second one is a software based system, and the last one is a proposed mix system. For all configurations, a redundant system two out of four ('2oo4') is assumed. Availability and reliability points of view are taken into account. The proposed mix system is explained in full detail. A discussion about programmable logic and its considerations are introduced. A CPLD based system in a research reactor (RA1) and its functionality are explained.

1. INTRODUCTION

The starting point of the present paper is the current requests for the first reactor protection system of research/production type. These requests assume a basic configuration that consists of a redundant system of four independent channels. The protective action is generated with the request of at least two of these channels.

In a general defence in depth scheme, the protection logic is a part of the first line of actions to protect the integrity of the installations. The system monitors the operating state of the reactor and provides protective actions and alarms with a high grade of reliability if one or more parameters reach safety limits.

In addition, the system transmits information about input lines, internal status and outputs to other systems without affecting its reliability.

A study of three architectures was accomplished: the first one is the traditional hardware based; the second is a modern one that makes use of software; and finally, the proposed architecture that combines the best of both previous ones.

2. GENERAL OUTLINES

In this paper, reliability and availability are used as a parameter to compare the architectures studied at the level of function blocks. Since both reliability and availability values are close to the unit, the complementary concepts of unreliability and unavailability are used in order to work more comfortably.

For unreliability (UR) and unavailability (UA) estimation, the following simplifications were introduced:

— All unreliability and unavailability values are one or more orders of magnitude lower that 10^{-3}. Therefore, second order infinitesimals can be eliminated.
— For comparison purposes among distinct architectures, global values H for hardware unreliability and unavailability are assumed.
— The unreliability/unavailability values relative to software are expressed as S.

3. DESCRIPTIONS OF THE ARCHITECTURES

3.1. Hardware logic based architecture

The protection logic for research and production reactors has been using ‘1oo2’ or ‘2oo3’ redundancy hardware or hardwired systems. However, the first analysed architecture is a hardware based ‘2oo4’ system, in order to be consistent with the other studied architectures. It is shown in Fig. 1.

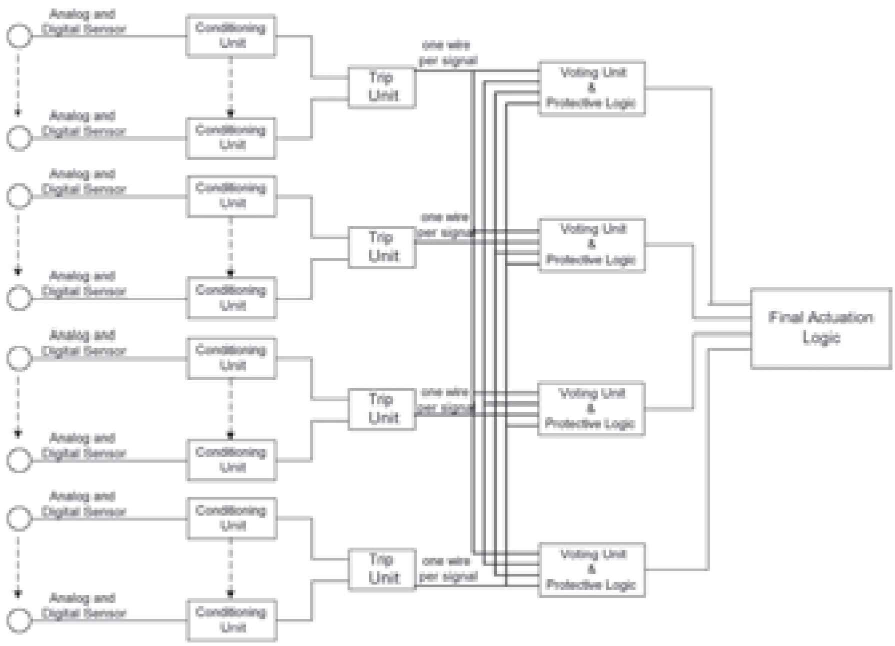


FIG. 1. Hardware logic based architecture.

Each of the four channels acquires several input signals. Each analogue signal is processed by a high/low comparator. The resulting binary signals are galvanically isolated before being sent over each of the four voting units, the supervision and control system (SCS) and the security panel. Analogue signals and setup trip levels have to be galvanically isolated before being sent to the SCS and to the security panel.

The computation of the UR and UA faults trees lead to the following results (see Ref. [1]):

UR = H
UA = H

In this case, UR and UA values are a consequence of a single not redounded final actuation logic.

3.2. Software based architecture

Modern systems (i.e. Star of Framatome, TMR of Triconex, Teleperm XS of Siemens) use microprocessors and software. It is difficult to assign a figure to software reliability; the quality is ensured with extensive (and expensive) verification and validation methods.

The studied software based architecture is shown in Fig. 2. This architecture takes into account the use of software at different levels:

- *Data acquisition:* That is, smart sensors transfer information in a bidirectional way by means of field bus.
- *Set Point Unit:* The Set Point Unit is a peripheral device connected to the Trip Unit (TU). It is used to set the safety limits.
- *Trip Unit:* The TU compares the input signals with its safety limits.
- *Digital Communication Channel (DCC):* The DCC provides the information from the TU to the Voting Unit (VU) and the Protection Logic (PL).

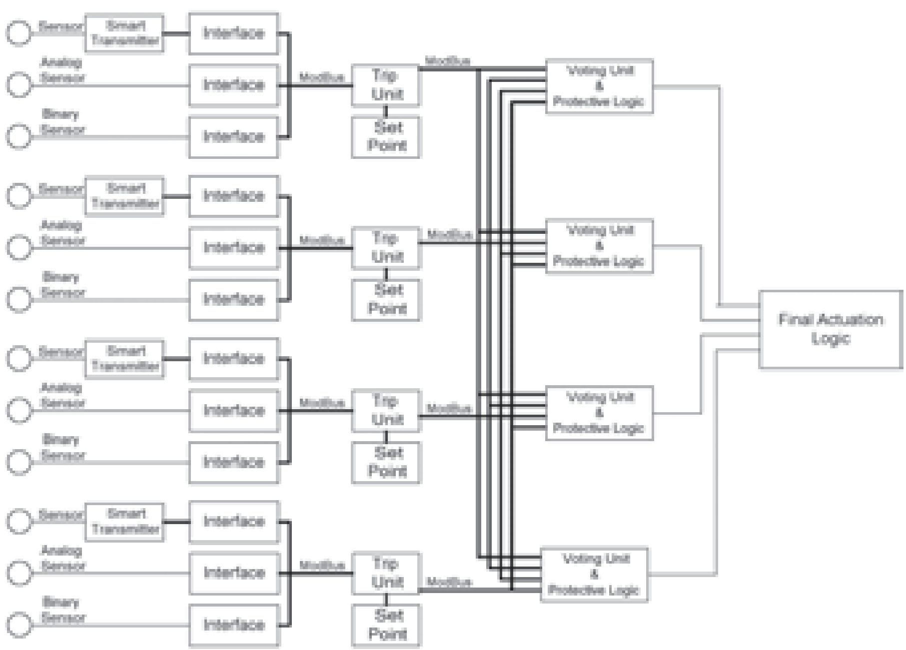


FIG. 2. Software based architecture.

— *Voting Unit and Protection Logic:* The VU and the PL process the logical signals obtained from the four TU through each DCC. The voting unit determines the coincidence of logical signals from the different channels. The coincidence criterion will be two out of four ('2oo4'). The protection logic implements the required logical equations and generates the triggering signals for the protective action.

The final actuation logic receives the PL signals and makes the final voting '2oo4' issuing a single order of protective action.

In this case, the computation of the UR and UA faults trees lead to the following results (see Ref. [1]):

$$\mathrm{UR} = 5\ \mathrm{S} + \mathrm{H}$$
$$\mathrm{UA} = 5\ \mathrm{S} + \mathrm{H}$$

Low unreliability S values are difficult to achieve, and almost impossible to get near H magnitude. For very high reliability requirements, the licensing process is very hard and expensive.

A possible way to handle this problem could be the approach of Framatome in their Star systems, which fulfilled the diversity criterion, using two microprocessors for each function in each channel. Both microprocessors were programmed by different developers' teams using different technologies. Hence, the probability of common mode faults notably decreases.

3.3. The proposed hardware/software architecture

The suggested scheme (Fig. 3) takes the advantages of modern systems, improves the reliability and the functionality compared with the hardware architecture and simplifies the scheme, reducing the system's costs.

The computation of the UR and UA faults trees lead to the following results (see Ref. [1]):

$$\mathrm{UR} = \binom{2}{N^\circ\, of\ Rods} \mathrm{H}^2$$

In the case of having 6 rods:

$$\mathrm{UR} = 15\ \mathrm{H}^2$$
$$\mathrm{UA} = \mathrm{S} + 6\ \mathrm{H}$$

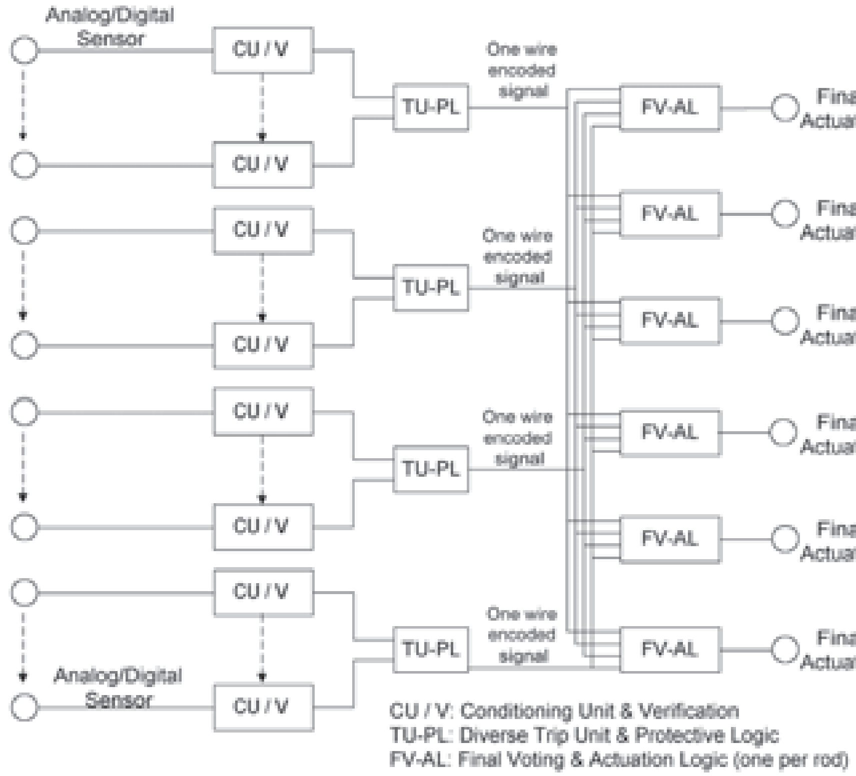


FIG. 3. The proposed hardware/software architecture.

The main issues to achieve these objectives are described in the following points.

3.3.1. Mixed implementation (hardware/software)

A mixed architecture is suggested with four independent identical channels with its functionality supported by software with hardware backup.

3.3.2. One final actuation logic per rod

For hardware based architecture, the UR value depends on the final actuation logic unreliability (Section 3.1). It is proposed to implement an architecture that uses a final actuation logic per rod. These modifications meet the single failure criteria, ensuring that no single failure could avoid the protective actions. In Fig. 3, architecture with 6 rods is shown.

3.3.3. Final voting logic

The previous voting logic is eliminated, resulting in savings in both wiring and galvanic isolators, which are essential in the achievement of electric independence of channels.

The previous cross-correlation of the input signals is a technique of noise suppression aimed to avoid spurious trips. However, it includes a functional dependence among channels, since if a signal in one channel is out of service, a noise in the same signal in another channel produces a system trip. That is why their replacement by auto-correlation techniques on every signal input is proposed.

Moreover, the use of an encoded trip signal and the inclusion of the cross-correlation at the final voting logic level is suggested.

These final voting units will produce a trip when two out of four channels request protective action, with at least one trigger event in common. This unit must work in the two out of three mode ('2oo3'), making the maintenance during operation of one of the four channels possible.

4. DIVERSE TRIP UNIT AND PROTECTION LOGIC

The Diverse Trip Unit and protection logic is the main module of each channel in the proposed architecture (mentioned in Section 3.3).

Figure 4 shows a scheme of this unit. The following points give a more detailed description.

4.1. Verification modules

Verification modules are simple devices meant to accomplish a verification of the wiring and the input signal devices (Mux, ADC, Ports, connectors, etc.) by means of average in the case of analogue signals and checksum in the case of digital ones.

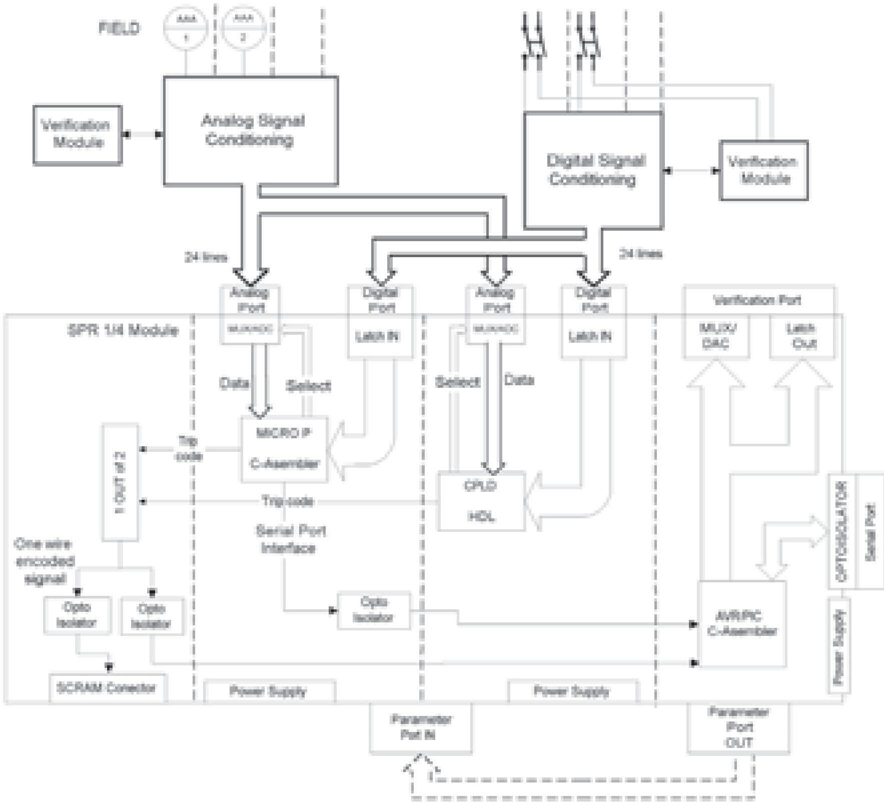


FIG. 4. Diverse Trip Unit and protection logic.

4.2. Microprocessor module

The microprocessor module has the functions of acquisition, margins verification and protection logic. Its output is an encoded trip request. Moreover, it transmits the internal state of the channel to the SCS.

4.3. Complex programmable logic device module

The complex programmable logic device (CPLD) module is a hardware redundancy of the functions of acquisition, margins verification and protection logic. Its output is an encoded trip request.

There are re-configurable CPLDs (EEPROM, flash based), which prove to be practical and economical during the development time, and there are One

Time Programmable (OTP) CPLDs (OTPROM, antifuse technology), which are preferable from the reliability point of view. In RPS, the use of OTP CPLDs is strongly suggested.

Proposed architecture bases its reliability on programmable hardware. The question that arises is whether the programmable hardware has the same order of reliability as hardware. The elements confirming this hypothesis are:

(1) A one-time programmable CPLD can not be modified once programmed.
(2) CPLD's manufacturers give failure rates of the same magnitude of hardware.
(3) CPLD's behaviour resembles hardware behaviour, both using simple and separated blocks with different parallel functions, giving the possibility of performing V&V process of every block in a simple way. This differentiates them from software based implementations, which have an execution cycle.
(4) These devices are largely used in the aeronautical and aerospace industry, where the required reliability levels have the same order as nuclear applications.

4.4. Supervision module

The supervision module is a fast RISC microprocessor that takes the internal state of the channel, the input signals and the set points of the microprocessor module, and sends the information through a serial optically isolated bus toward the SCS. Moreover, this module sets the safety limits to the CPLD and the microprocessor module.

In addition, this module makes an automatic test of the channel periodically by applying known signals to the inputs.

4.5. Dynamic voting module

The dynamic voting module has the function of unifying the output of two parallel modules (CPLD and microprocessor modules), requesting a trip in '1oo2' mode. This module produces an encoded output, requesting protective action and issuing the trigger events to the final actuation logic and to the SCS.

5. RECENT EXPERIENCES

5.1. Upgrade of RA1 protection logic

The upgrade of the research reactor RA1 RPS started with the replacement of one of the two channels with a compatible CPLD based module. After a six month testing period, the second channel was also replaced. In Fig. 5, a scheme of the implemented protection logic is shown.

The internal design of the CPLD logic was accomplished by hardware description language (VHDL).

Exhaustive processes of V&V were fulfilled to corroborate the good functioning of the compilation process and CPLD programming. Complete automatic testing for the combinational logic was done. Moreover, the sequential logics were easily verified because they had few states.

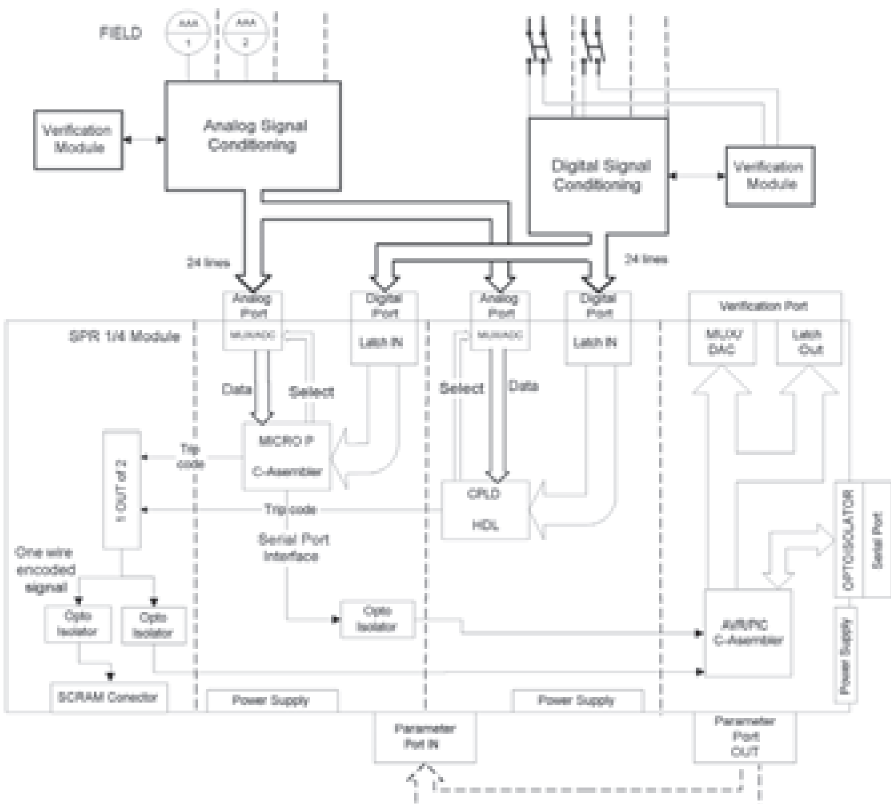


FIG. 5. Implemented protection logic for RA1 based in a CPLD.

The next step in the upgrading process will be the addition of a software based module. The software core of this module has been used for more than 10 years for the interlock function in the research reactor RA3.

5.2. Other essays

Other essays were accomplished to prove the feasibility of certain functions. The most important was the development of a Trip Unit, based in a multiplexed 12 bits ADC. The prefixed margins were kept in a PROM and the comparison was achieved by a CPLD. Moreover, the auto-correlation function of the inputs was accomplished by means of the same CPLD.

6. CONCLUSIONS

In terms of reliability, the proposed architecture presents the lowermost levels of unreliability (15 H^2). This result is achieved firstly, by means of hardware backup and secondly, by the addition of one voting logic for each final actuator.

Regarding availability, the obtained value (S + 6 H) is not the best.

In this way, reliability depends on hardware, while availability depends on software.

The system has all the functionality of a software based system, its installation and licensing can be much less complex and drastically reduces wiring, due to the elimination of the coincidence logic in the TU, transferring this function to the final voting logics.

Although it cannot be taken for granted that the programmable hardware has the same order of reliability as the hardware, several elements can be taken to lead to this conclusion.

Finally, the experience achieved provide evidence of the feasibility of the proposed architecture.

REFERENCES

[1] VERRASTRO, C.A., CARBALLIDO, J.C., Arquitectura del Primer sistema de Protección de Reactor, AATN XXII, Argentina (2000).

[2] CLULEY, J.C., Electronic Equipment Reliability, Chapter 3, 2nd edn, Macmillan Press Ltd, London (1981).

[3] DENNIS LAWRENCE, J., Software Reliability and Safety in Nuclear Reactor Protection Systems, Rep., Lawrence Livermore National Laboratory, USA (1993).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Research Reactor Instrumentation and Control Technology: Report of a technical committee meeting, IAEA-TECDOC-973, IAEA, Vienna (1997).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Specification Requirements for Upgrades Using Digital I&C, IAEA-TECDOC-1066, IAEA, Vienna (1999).

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, Modern Instrumentation and Control for Nuclear Power Plants: A Guidebook, Technical Reports Series No. 387, IAEA, Vienna (1999).

[7] INTERNATIONAL ELECTROTECHNICAL COMMISSION, Separation within the Reactor Protection System, Publication 709, IEC, Geneva (1981).

[8] LEITCH, R.D., Reliability Analysis for Engineers, Oxford Science Publications, Oxford (1995).

[9] MISHIMA, T., SHIRAKAWA, T., The design requirement and development of software for the digital safety protection system in Kashiwazaki-Kariwa units 6 and 7, Tokyo Electric Power Co. Inc., Tokyo.

[10] NUCLEAR REGULATORY COMMISSION, Evaluation of Topical Report BAW-10191P on Star System Components for Reactor Protection System Digital Upgrades, NRC, Washington, DC (1995).

[11] NUCLEAR SAFETY BUREAU, Criteria for the design of nuclear installations, Regulatory Guideline RG-5, draft, NSB, Canberra (1998).

[12] NUCLEAR SAFETY BUREAU, Safety Assessment Principles for Nuclear Facilities, draft, NSB, Canberra (1998).

THERMOHYDRAULIC DESIGN AND SAFETY ANALYSIS OF RESEARCH REACTORS

A. HAINOUN
Reactor Safety Analysis Division, Nuclear Engineering Department,
Atomic Energy Commission of Syria (AECS),
Damascus, Syrian Arab Republic
Email: ahainoun@aec.org.sy

Abstract

The paper presents briefly the trend of thermal-hydraulic design and a safety analysis of medium and high flux research reactors. This field of deterministic safety analysis is being considered by the IAEA in the framework of a coordinated research project (CRP) initiated in 2002 on the Assessment of Analytical Tools for Different Research Reactor Types. The objective of this project is to establish a forum of international experts in order to integrate the activities for improvement and verification of selected computer codes that can be considered as reference tools in the safety analysis of research reactors, similar to that of power reactors. This undertaking supports the international ambition of improving the safety features and standards of research reactors, which can be useful for countries with long experience with research reactors, and very helpful for countries that have research reactors with low neutron flux and which may be looking to extend them or build other reactors with higher neutron flux. In this regard, the methodological approach on modification, verification and application of advanced computer codes for the safety analysis of research reactors is presented. Furthermore, a semi-empirical correlation for the first design limit regarding the onset of flow instability, for medium and high flux reactors, has been suggested.

1. INTRODUCTION

Research reactors primarily serve to generate neutrons for research purposes. In order to achieve the high neutron flux densities necessary for this application, high power densities are required, which in some very high flux reactors reach several MW/l and are thus considerably higher than the power densities of modern power reactors. These high power densities are provided by a fuel element concept in which the active part of the fuel element generally consists of a bundle of thin, vertically arranged aluminium plates, in which the uranium is embedded. The plates are arranged in such a way that cooling gaps result between them with a gap width of a few mm. The heat generated by fission in the plates is released via short heat conduction paths to the coolant water flowing between the plates at high velocity (some m/s up to some ten m/s), which flows out of a common distributor into the cooling gap and at the outlet flows back into a common plenum. With this parallel arrangement of the fuel elements, the driving differential pressure for all cooling gaps is practically identical; it depends on the cooling water requirements of the most greatly heated channel (hot channel). The latter is selected in such a way that the water is greatly subcooled at the outlet, relative to the minimum possible coolant pressure.

1.1. Classification of research reactor

Worldwide, there are more than 100 research reactors of different types with a power higher than 1 MW. Using the criteria of core power density (MW/l) and depending on the relation between neutron flux density and core power, the research reactors can be subdivided into three different categories (see Fig. 1):

- Low flux reactors with neutron flux: $\phi < 10^{14} cm^{-2} \cdot s$.
- Medium flux reactors with neutron flux: $10^{14} cm^{-2} \cdot s \leq \phi < 10^{14} cm^{-2} \cdot s$. The following reactors belong to this category: MAPLE-X10 in CRL-Canada, JRR-3M في TRE-Japan, ORPHEE in LBL-France, BERII and FRJ2 in Germany.

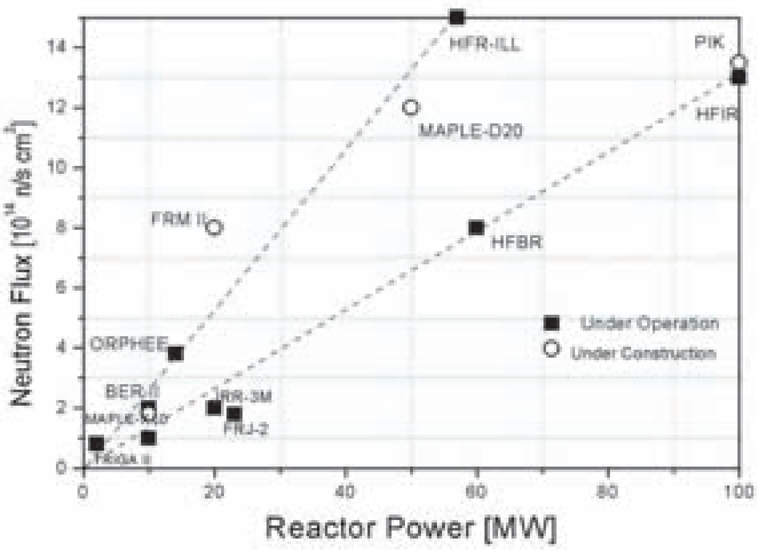


FIG. 1. Dependency of neutron flux on reactor power for various reactor types.

— High flux reactors with neutron flux: $\phi \geq 5 \cdot 10^{14} cm^{-2} \cdot s$. The following reactors belong to this category: HFR in Grenoble, France; SM-ZR in SRINR, the Russian Federation, HFIR in ORNL, the United States of America, FRMII in Munich, Germany [7].

2. EXTENSION, VERIFICATION AND APPLICATION OF THERMOHYDRAULIC DESIGN AND SAFETY CODE FOR RESEARCH REACTORS

The operation limits of nuclear reactors have been set to prevent fuel-cladding damage. By the design basis accidents, damage would occur by PWR at departure from nucleate boiling (DNB) resulting from vapour film forming at the fuel surface; and by BWR when the liquid film on the fuel surface dries out. The heat flux at DNB or liquid film dry out is called critical heat flux (CHF). PWR and BWR are designed in such a way that the operating limits do not permit an exceeding of CHF.

Fuel elements of research reactors have a compact core structure with multi parallel channels and are, therefore, subject to the so-called thermo-hydraulic instability (THI)—or flow excursion (FE)—which differs from CHF that would occur at a fixed channel flow rate. During postulated design accidents in research reactors, thermohydraulic instability (THI) would occur before the CHF limit is reached.

The compact core structure and the special design of the fuel elements of research reactors make particular demands on the thermohydraulic construction and on the technical safety measures. In the case of accidents in research reactors, transients take place in the range of seconds. This is due, not least, to the high heat flux of 150 W/cm^2 and above, and the low system pressures which for many types of reactors are about one bar. With various accidents, such as impairment of forced cooling by failure of the coolant pumps, thermohydraulic flow instabilities may arise in the narrow cooling channels due to steam formation mostly in subcooled boiling regime, which results in the critical heating surface load being exceeded within a few seconds affecting the fuel plates, which, due to their extremely low melting point of 658° C—by Al-U alloys—are destroyed within a few seconds. Many other thermal-hydraulic phenomena are resulting from the specific construction of the research reactor core.

Extensive research work has been performed worldwide for a number of years with the aim of developing and verifying various thermohydraulic program systems for analysing the thermohydraulic in light water reactors during transients and loss of coolant accidents including, for example, the

RELAP and CATHARE codes and the ATHLET code. However, it is not possible to apply these codes, which were developed for power reactors, in safety analyses of research reactors without additional extensions and verifications due to the specific features of the latter reactors.

According to the above mentioned facts and in order to perform a comprehensive safety analysis for research reactors, which have not been so extensively researched as power reactors, it is necessary to improve, modify and verify the advanced program packages, such as RELAP, ATHLET, CATHARE, CATHINA, and such programs developed in the past with limited capability in the design of research reactors, including COBRA and PARET.

In the frame of assessment of analysis tools for research reactors by the AECS, the computer programs ATHLET, PARET and COBRA-3C/PERTR have been tested and partially modified and verified for the application on design and safety analysis of research reactors. The methodological approach used in the testing and verification of the different programs is presented in Fig. 2.

2.1. Verification and application of the program ATHLET

Analysis of thermal-hydraulics by leaks and transients (ATHLET) is an advanced code for the simulation of design basis and beyond design basis accidents (without core degradation) of light water reactors. ATHLET is being developed by the German Society for Reactor Safety (GRS). It has a modular structure, i.e. several routines are combined into independent function units with their own time integration, including the thermo-fluid dynamic module (TFD), heat conduction and transfer module (HECU), neutron kinetic module (NEUKIN) and general control simulation module (GCSM), with the numerical integration method FEBE. The one-dimensional, two-phase fluid dynamic is based on the conservation equations for the steam and water mass, steam and water energy and the mixture impulse (5-equation version). The mechanical non-equilibrium of the two phases is described in the impulse equation by a full range drift-flux setup on the basis of the “extended envelope theory” [5].

For application of ATHLET code in the safety analysis of research reactors, a new model was developed and implemented [3] to describe void formation in subcooled boiling regime in order to simulate thermohydraulic instability by research reactors. The model implemented in ATHLET to describe non-equilibrium effects is based on the competing effects of evaporation and condensation in the subcooled boiling regime. It describes the rate of bubble generation on the superheated heating surfaces and the subsequent condensation of the steam in the subcooled core flow. The extended code was validated for the conditions of medium flux reactors—heat flux up to 2.5 MW/m^2, flow velocity up to 5 m/s and system pressure up to 140 kPa using the experiments of FZJ [3].

To extend the use of ATHLET in the conceptual safety analysis of high flux research reactors, such as HFR/Grenoble, ANSR/USA and FRMII/Germany, two extensions were implemented [HAA-2001]. The first deals with

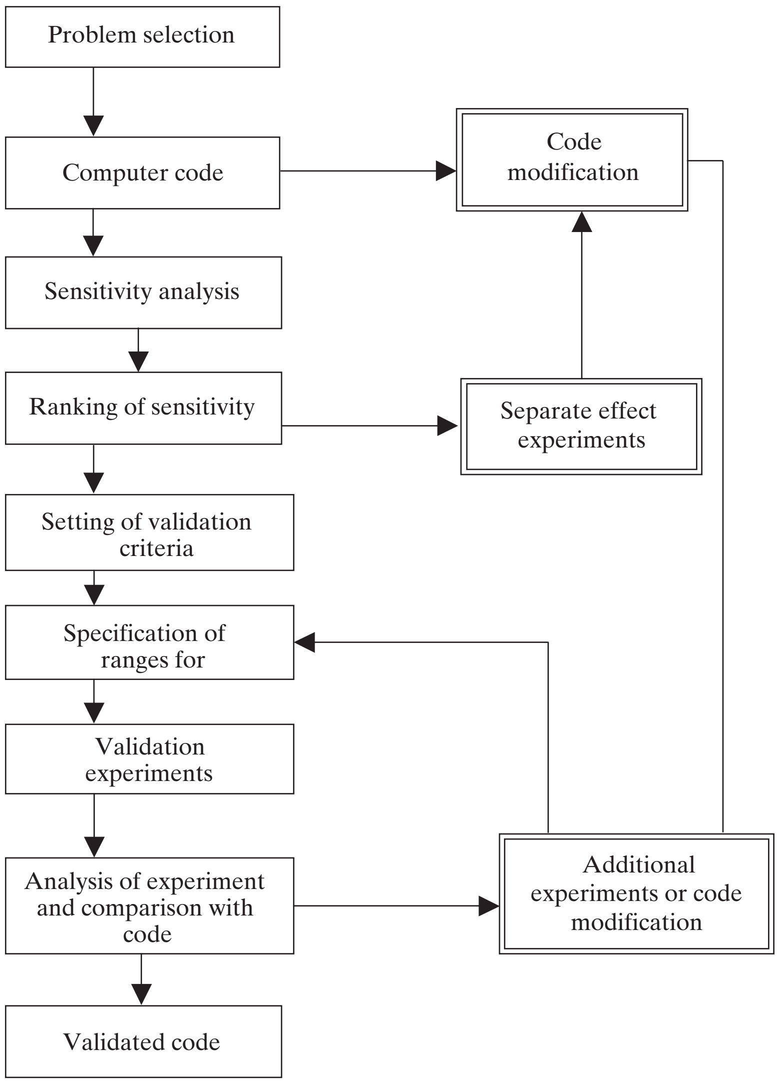


FIG. 2. General scheme of computer code validation for safety assessment of research reactors.

the description of the steam condensation rate for highly turbulent flow and the second with a CHF correlation in subcooled boiling conditions. The ATHLET code is validated by recalculation of experiments on thermohydraulic instability at very high flow velocities (up to 28 m/s) and extreme heat flux (up to 20 MW/m^2) conducted in the THTL test facility at Oak Ridge National Laboratory. The thermal-hydraulic test loop (THTL) is an experimental facility that was constructed to support the development of the American Advanced Neutron Source Reactor (ANS) as well as the German research reactor FRM-II, built in Munich. Both reactors have very high thermal neutron flux (about $8 \times 10^{14} cm^{-2}$ for FRM-II and should be much higher for ANSR) with compact core design to accommodate very high power densities using very high coolant mass flux and subcooling levels [9].

Some examples of verification results for the modified ATHLET code are presented in Fig. 3. More details are given in [4]. From Fig. 3, it can be seen that the calculated mass fluxes at the onset of flow instability (OFI) agree very well with the experimental data. These results demonstrate the capability of ATHLET to simulate flow instability, since the most important factor identifying the limit of stable region, namely, the mass flux rate, can be calculated in good agreement with the experiment.

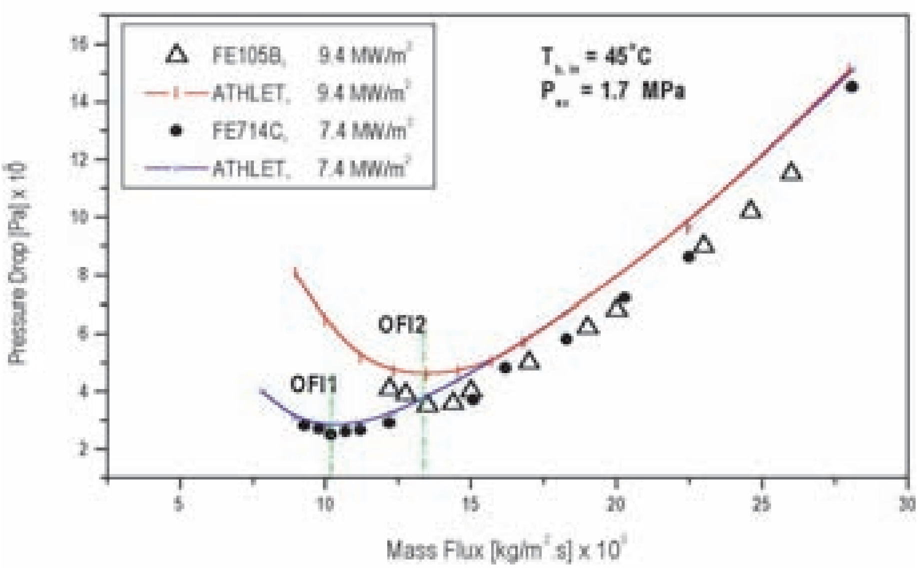


FIG. 3. Comparison of ATHLET results with THTL test data for subcooled flow instability in the stiff mode [4].

2.2. Limiting criteria for the onset of flow instability by medium and high flux research reactor

The first approach to predict the onset point of instability, or so-called flow excursion, was proposed by Whittle and Forgan [12]. They used the critical parameter η to correlate their experimental data. This parameter is identical to Bowring parameter describing the onset of significant void (OSV) in sub-cooled boiling. It depends on heat flux ($\dot{q}$), outlet subcooling ($T_{out}-T_{in}$) and flow velocity (w):

$$\eta = \frac{w \cdot (T_{sat} - T_{out})}{\dot{q}} = const. \tag{1}$$

Where η ranges between 14 and 25 [12].

The experimental observations and theoretical analysis [3] indicate that the onset of flow instability (OFI) occurs after the onset of significant void as consequent of steam formation in the channel. Hence, the OFI is related directly to the increased void formation after OSV, so that the critical inlet flow velocity at OFI can be estimated using a criterion similar to that given in Eq. (1). Obviously, the critical velocity is dependent on the geometry data and the thermal-hydraulic conditions of the channel, such as pressure, heat flux and inlet temperature.

Using the energy balance for the heated channel, one gets the following expression for the temperature rise of coolant in a uniformly heated round channel:

$$(T_{out} - T_{in}) = \frac{4}{\rho \cdot c_p} \frac{L}{d} \cdot \frac{\dot{q}}{w} \tag{2}$$

Where L and d are the length and diameter of the heated channel respectively.

Combining Eq. (1) and Eq. (2) gives the following expression:

$$(T_{sat} - T_{in}) = \frac{\dot{q}}{w} \cdot \left(\eta + \frac{4}{\rho \cdot c_p} \frac{L}{d} \right) \tag{3}$$

The second expression on the right side indicates geometrical dependencies. The evaluations of experimental data result in introducing the parameter L/S for channel heated length and gap width, respectively. The use of channel width enables the considering of different fuel element geometries. Rearranging Eq. (3) provides the following simple correlation for the critical inlet velocity at OFI in subcooled boiling:

$$w_{OFI} = C\frac{L}{d}\cdot\frac{\dot{q}}{(T_S - T_{in})} \tag{4}$$

w [m/s]: inlet flow velocity at the OFI
L [mm]: heated channel length
d [mm]: channel gap width
$\dot{q}\,[MW/m^2]$: maximum heat flux
T_s, Tin [$°C$]: saturation and inlet temperature
$T_s - Tin$ [$°C$]: inlet subcooling

C is a proportional constant dependent on geometry form. Considering the verification results for ATHLET [HAA-92, 96], [HAA-2001], one gets the following values:

— C_1 =0.58 for rectangular channel;
— C_2 =0.32 for annular channel (with only inner tub heating).

The application range of Eq. (4) is limited to the following thermo-hydraulic conditions:

— Coolant: upward flow water;
— Channel gap width: 1.2–4 [mm];
— Heated length: 500–600 [mm];
— Inlet water temperature (T_{in}): 45–60 [$°C$];
— Exit pressure of system (p_{ex}): 0.1–1.7 [MPa];
— Range of average heat flux: 1–14 [MW/m^2];
— Range of velocity: 1–21 [m/s].

Figure 4 shows the accuracy of this correlation by comparing its prediction with the experimental data for annular geometry from Ref. [3] for the condition of medium flux reactors and the experimental data for rectangular geometry from Refs [9, 10] for high flux research reactors.

The proposed correlation predicts by a given maximum heat flux of the hot channel the amount of inlet flow velocity at which an onset of instability is expected. It presents a simple procedure to estimate the minimum allowed flow

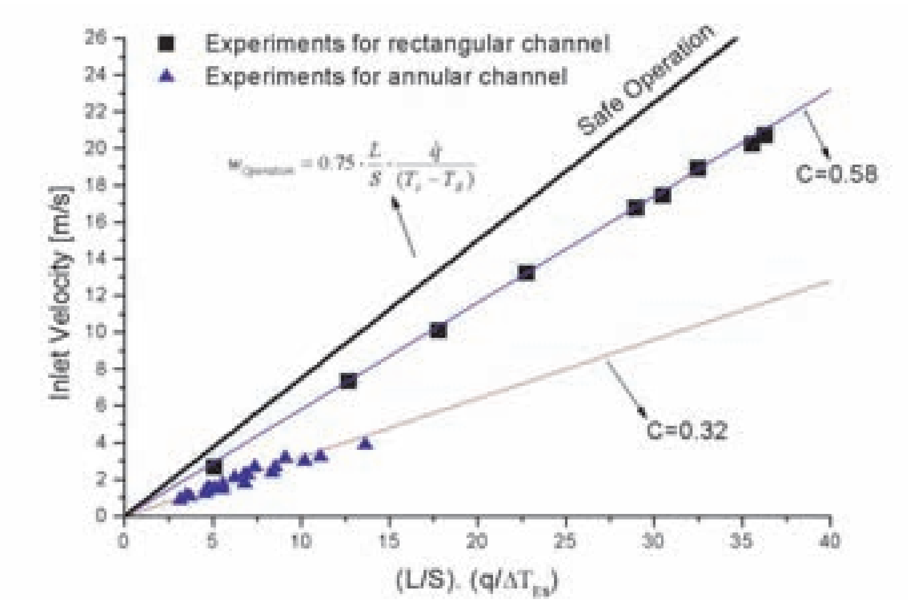


FIG. 4. Dependency of critical inlet velocity at the onset of flow instability on various thermohydraulic parameters for different fuel element types of research reactors.

velocity at which the fuel element can be operated by considering the geometry and the thermodynamic boundary conditions. No additional information about the subcooled void fraction in the channel is necessary. However, ATHLET calculations show that the void fraction at OFI attains values by 20–30% at the channel outlet [4].

According to this first design limit, the safe operation of the fuel elements by the given pressure and heat flux is ensured by inlet velocity higher than the critical value:

$$w_{operation} > w_{OFI} \Rightarrow w_{operation} = S\ w_{OFI} \tag{5}$$

The safety parameter S characterizes the safety margin that considers all physical and technical uncertainty. For conservative estimation S can be set to 1.3. Thus, a new weighted constant C_S can be given by considering the safety parameter:

$$C_S = C_1 \ \ x \ \ S = 0.58 \ \ x \ \ 1.3 = 0.75 \tag{6}$$

Finally, the following correlation can be given that is applicable for conservative estimation of inlet flow velocities for both geometries (Fig. 2):

$$w_{Operation} = S \cdot w_{OFI} = 0.75 \cdot \frac{L}{S} \cdot \frac{\dot{q}}{(T_s - T_E)} \tag{7}$$

3. CONCLUSION

This paper presents an overview on the trend of thermohydraulic design of research reactors. It is obvious that increasing the core power density to achieve higher neutron flux reflects in a more compact core structure. This implicates more difficulty in the thermohydraulic design, including the reducing of channel width and increasing of system pressure. Consequently, detailed thermohydraulic safety analyses are necessary to incorporate integral phenomena in the transient region and not only under steady state conditions.

However, the proposed steady state correlation for the first thermohydraulic design limit shows that medium and high flux research reactors follow the same physical trend. Thus, to generalize the application of the presented correlation, additional experimental data will be used, such as the data of Costa [1], Waters [11] and Whittle [12] and the collected data by Moritz [6]. In this regard, the OFI correlation proposed in the program COBRA-3C/RERTR [13] will be tested and modified.

ACKNOWLEDGEMENT

The author would like to thank I. Othman, Director General of AECS, for his encouragement.

REFERENCES

[1] COSTA, J., Mesure de la perte de pression par accélération, Double-Phase, WinfritII (1967).

[2] HAINOUN, A., MEISTER, G., et al., Theoretische und experimentelle Untersuchungen zur thermohydraulischen Instabilität von MTR-Reaktoren, Jahrestagung Kerntechnik, Tagungsbericht (1992), pp. 469–473.

[3] HAINOUN, A., WOLTERS, J., HICKEN, E., Modelling of void formation in the subcooled boiling regime in the ATHLET code to simulate flow instability for research reactors, Nucl. Eng. Des. **167** (1996) 175–191.

[4] HAINOUN, A., SCHAFFRATH, Simulation of subcooled flow instability for high flux research reactors using the extended code ATHLET, Nucl. Eng. Des. **207** (2001) 163–180.

[5] LERCHEL, G., AUSTREGESILO, H., ATHLET Mode 1.2 Cycle A, User's Manual, Rep. GRS-p-1/Vol.1, Rev. 1, Gesellschaft für Reaktorsicherheit, Munich (1998).

[6] MORITZ, U., Auswertung von Messungen zur Thermohydraulischen Instabilität, Rep. JTK-91, Bonn (1999).

[7] Neutronenquelle München FRM-II, Statusbericht der Projektgruppe "Neuer Forschungsreaktoren" der Fakultät für Physik E21, Technische Universität München (1992) (in German).

[8] ROUHANI, S. Z., AXELSSON, E., "Calculation of void volume fraction in the subcooled and quality regions", in Proc. Mtg on European Two-Phase Flow Group, Oslo, 18–20 June 1968.

[9] SIMAN-TOV, M., et al., FY 1993 Progress Report on the ANS Thermal-Hydraulic Test Loop Operation and Results, Rep. ORNL/M-3789, Oak Ridge National Laboratory, Oak Ridge, TN (1994).

[10] SIMAN-TOV, M., et al., "Static flow instability in subcooled flow boiling in parallel channels", in Proc. 2nd Int. Conf. on Multiphase Flow, Kyoto, April 1995.

[11] WATERS, Heat Transfer Experiments for the Advanced Test Reactor, Rep. BNWL-216, Battelle Northwest Laboratories, Richland, WA (1966).

[12] WHITLE, FORGAN, A Correlation for the Minimum in the Pressure Drop Versus Flow-Rate Curves for Sub-Cooled Water, NED (89-99) (1967).

[13] Supplementary information for COBRA-3C/RERT, 1982.

BLANK

EXPERIENCE WITH PROBABILISTIC SAFETY ASSESSMENT FOR THE BR2 REACTOR

S. DECLERCQ, P. GUBEL, F. JOPPEN, E. KOONEN
SCK•CEN,
Mol, Belgium
Email: fjoppen@sckcen.be

Abstract

The BR2 reactor is a 100 MW(th) research reactor operated by the Belgian Nuclear Research Centre SCK•CEN. It is one of the major MTR-type reactors in the world and it is the major nuclear infrastructure of SCK•CEN. First criticality was obtained in June 1961 and power operation started in January 1963. BR2 was shut down in the middle of the 1990s to replace the beryllium matrix and to inspect the vessel. This is a requirement in the licence which sets a limit on the fast neutron dose of the beryllium in the core. Before that time, it was already clear that a refurbishment was needed in order to guarantee a reliable and safe operation for another period of about 15 years. In 1992, the management of the SCK•CEN ordered a probabilistic safety assessment (PSA) as a means to define the objectives and priorities of the refurbishment programme. In 1997, after restart of the reactor, the authorities asked for an extension of the PSA in order to include the influence of failure of the support systems. This article gives an overview of the methodology and results of this PSA. However, the main purpose of the paper is to give the experience with the development of the PSA, to indicate the difficulties that were encountered and how they were treated. Detailed PSA results are only given to illustrate these topics.

1. INTRODUCTION

The BR2 reactor is a 100 MW(th) research reactor operated by the Belgian Nuclear Research Centre SCK•CEN. It is one of the major MTR-type reactors in the world and it is the major nuclear infrastructure of SCK•CEN. First criticality was obtained in June 1961 and power operation started in January 1963. More information about the operation of BR2 can be found in Ref. [1].

BR2 was shut down in the middle of the 1990s to replace the beryllium matrix and to inspect the vessel. This is a requirement in the licence which sets a limit on the fast neutron dose of the beryllium in the core. Before that time, it was already clear that a refurbishment was needed in order to guarantee a reliable and safe operation for another period of about 15 years. In 1992, the management of the SCK•CEN ordered a probabilistic safety assessment (PSA) as a means to define the objectives and priorities of the refurbishment programme. A discussion about the impact of the PSA on the refurbishment programme can be found in Ref. [2]. In 1997, after restart of the reactor, the authorities asked for an extension of the PSA in order to include the influence of failure of the support systems.

2. DESCRIPTION OF BR2

BR2 is a pressurized reactor with a normal operating pressure of 1.26 MPa, cooled by natural water. The reactor tank is placed in a pool filled with water. The maximal thermal power that can be evacuated by the heat exchangers is 100 MW. The maximum allowable heat flux on the fuel elements is 470 W/cm² for routine operations and up to 600 W/cm² in special conditions.

Figure 1 gives a scheme of the BR2 hydraulic circuits. The reactor vessel and the reactor pools are located inside the containment building.

The other major components (primary pumps, heat exchanger, pressurizer) of the primary circuit are located outside the reactor building. Two

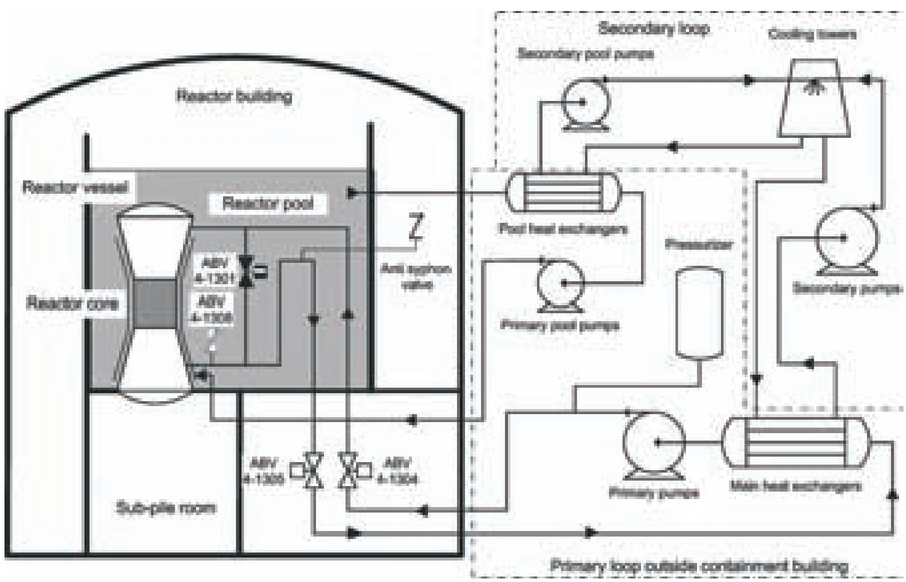


FIG. 1. BR2 hydraulic circuits.

isolation valves in the primary circuit, one for the inlet line (ABV4-1304) and one for the outlet line (ABV4-1305), are foreseen and have to close in case of certain events, such as loss of pressure or loss of flow. In this case, a connection valve between the inlet and outlet pipe (ABV4-1301) will open such that further cooling of the reactor, after scram, is possible by natural circulation in the pool. A connection valve exists (ABV4-1308) between the reactor pool and the primary circuit. This is a non-return valve which opens when the pressure in the primary loop is lower than the static pressure of the pool water. In this way, water can enter the primary loop in case of a loss of primary water.

After scram, the core can be cooled by natural convection if the isolation valves are closed and the bypass valve is opened. In case the bypass should fail to open, it is still possible to cool the fuel elements by internal convection in the reactor vessel, but it is supposed that the additional cooling of the vessel wall by water flowing through the shroud must remain operational. This water comes from the pool cooling circuit.

3. THE PROBABILISTIC SAFETY ASSESSMENT MODEL

The probabilistic safety assessment model of BR2 is based on two generic event trees. One event tree is used for initiating events leading to loss of coolant accidents (LOCAs), while the other describes the consequences for all other initiating events. The failure probabilities of the different nodes in the event trees were estimated by simple methods in the original PSA analysis. Later, a number of these nodes were developed using fault trees.

3.1. The original PSA model

A short description of the original PSA model is given in this paragraph. A more detailed discussion can be found in previous publications (see Ref. [3]).

The first protection after the occurrence of an initiating is the shutdown of the reactor. All event trees start with a node 'reactivity control' which means scram of the reactor. Failures of the scram system are assumed to lead to accident without scram and core damage.

3.1.1. LOCA initiating events

LOCA initiating events are categorized in two ways, namely, size and location of the LOCA. For the size of the LOCA, four different cases are identified:

(1) *Large LOCA*: The flow of water being lost is greater than the maximum flow through the pool communication valve. The valve has a capacity of 500 l/s. A large LOCA is assumed to lead to core damage.
(2) *Medium LOCA*: The flow of water being lost is smaller than the maximum flow through the pool communication valve, but the pool communication valve opens before the isolations valve in the primary circuit are closed. A quantity of water will be lost.
(3) *Small LOCA*: The flow of water being lost is greater than the normal primary make-up system (200 l/h), but the primary circuit isolation valves will be closed before the opening if the pool communication valve. If the protection system works correctly, no water will be lost outside the containment building.
(4) *Minor LOCA*: The flow of water being lost can be compensated by the normal primary make-up system.

The possible locations for a LOCA are:

(1) Leak from the reactor vessel to the reactor pool.
(2) Leak from the reactor vessel bottom head to the sub-pile room.
(3) Leak from the inlet pipe work to the reactor pool.
(4) Leak from the outlet pipe work to the reactor pool.
(5) Leak from the anti-siphon outside the reactor pool.
(6) Leak from the inlet pipe work between the isolation valve and the pool wall.
(7) Leak from the outlet pipe work between the isolation valve and the pool wall.
(8) Leak from pipe work, pumps and heat exchangers outside the reactor building to the atmosphere.
(9) Leak in the heat exchangers from the primary side to the secondary.

In this way, all LOCA initiating events could be identified by a code Axy, where:

— The letter A indicates that the initiating event is a LOCA;
— The number x indicates the size of the LOCA;
— The number y indicates the location of the LOCA.

It is to be noted that not all combinations are possible. For example, a large LOCA (category A1y) is only possible outside the reactor building (y = 8).

3.1.2. The other initiating events

The non-LOCA initiating events are described by another generic event tree. These other categories are:

B. Reactivity increase initiating events.
C. Support systems failure initiating events.
D. Loss of secondary flow.
E. External events.

The generic event tree for the non-LOCA initiating events starts with the reactivity control node and develops further the cooling of the reactor.

Event trees for category E (External events) are not further developed. It must be noted that the two most important external events, namely, fire and seismicity are treated in deterministic studies and have led to important modifications.

3.1.3. The frequency of initial events

In order to obtain the resulting core damage frequency, it is necessary to assign an occurrence frequency to the different initiating events. In a number of cases, operating experience could be used, for example, in case the LOCA's 8 minor leaks outside the reactor building were detected during five years of operation. This gives an occurrence frequency for initiating event A48 of 1.6/a. It was further assumed that the probability of a minor LOCA escalating into a small LOCA is 10%, giving 0.16/a for the occurrence frequency of LOCA category A38. This technique was also used for reactivity induced initiating events. However, the occurrence frequency of most of the initiating events had to be estimated by judgement.

3.1.4. The failure probabilities of the different nodes of the event trees

The failure probabilities of the different nodes (protection systems) are more important. The aim of the PSA is to get information about possible weaknesses in the protection systems and to define modifications in the efficient way.

In the original PSA, the failure probability of only one node was calculated using fault trees, namely, for the scram system. The failure probability for a number of nodes was estimated using historical data, taking into account the number of times the system was operated (take this number n) and the number of failures, which was in most cases zero. The failure probability was than taken to be equal to 1/n, assuming that the next operation of the system could be a failure. In some cases, generic data were used.

3.1.5. Conclusions for the original PSA

The PSA led to a number of conclusions and to some modifications. A core damage frequency of 7.5 x 10^{-4}/a was calculated. This figure is questionable, due to rather arbitrary choice of the frequency of the initiating events and the uncertainties about the failure probabilities of the different protection systems. The most important conclusions and actions were:

— It is clear that the isolation valves and the bypass are very important. Ideally, the isolation valves should be doubled, but this is not possible. The volume of the room where the valves are located is not large enough to accommodate a second pair of valves. However, the activation circuits were renewed and made redundant. Further, the closure time of the valves was made shorter in order to limit the loss of water in case of a LOCA outside the containment building.
— During the development of the fault trees of the scram system, a potential common cause was discovered in the scram system. A single integrated circuit intervened in two independent scram signals. Correction was simple and straightforward by adding a second independent integrated circuit. This improved the reliability of the scram system by a factor of 15.
— The pool communication valve plays an important role in early compensation of LOCAs.

3.2. Extension of the probabilistic safety analysis (PSA) model

3.2.1. Definition of the extended PSA

The original PSA model contained no fault trees, except for the scram system. It was thought to be possible that by failure of important support systems, such as electricity, different nodes could fail due to common cause. The authorities asked to investigate further this possibility. This was done by developing fault trees for the different nodes of the original fault tree which could depend on support systems.

Fault trees were developed for the following systems:

— The isolation valves ABV4-1303 and ABV4-1304;
— The bypass valve ABV4-1301;
— The primary circulation pumps;

— The secondary circulation pumps;
— The pumps for pool cooling.

According to this more detailed analysis, small LOCA initiating events (category A3y) are responsible for three quarters of the core damage frequency. The most contributing component is the pool communication valve ABV4-1308. This is a confirmation of the result of the original PSA. More frequent testing of this non-return valve would lower the failure rate for the calculation. However, due to its location in the lower part of the reactor pool, the valve can only be tested during the replacement of the beryllium matrix when the reactor vessel is completely empty. This action is foreseen after a period of 15 years of operation.

Another important contribution is the 110 V DC electrical power supply for actuation of the bypass valve ABV4-1301. This is a backup for the normal 220 V AC power supply. The power supply is important since the valve does not move to a safe position on failure of the power supply, but stays in its position. The reason for this design is that inadvertent opening of the valve during reactor operation would directly lead to core damage.

The fault trees of the isolation valves of the primary circuit are a typical example of the consequences of using data from operating experience. The valves are of unique design so that no generic data are available. However, during the lifetime the valves are regularly moved, at least two times each operation cycle. They never failed. This allowed the conclusion that the failure rate is lower than 10^{-3} per demand. The failure rate of the activation system was calculated using detailed fault trees. The activation system was completely renewed during refurbishment. It consists of standard components and its failure rate can be calculated with sufficient confidence. As shown in Fig. 2, the failure rate of the mechanical valve determines the failure probability of the whole system. This can mask the result of the additional study, which was aimed to study the influence of support systems. The extension of the PSA indicated no further weaknesses.

4. CONCLUSIONS ABOUT THE USE OF PSA

Firstly, it is acknowledged that a PSA is a major project and the necessary means must not be underestimated. The PSA model must be detailed enough to be useful. A rather simplified model will just generate information that is already known. The use of detailed fault trees seems to be essential.

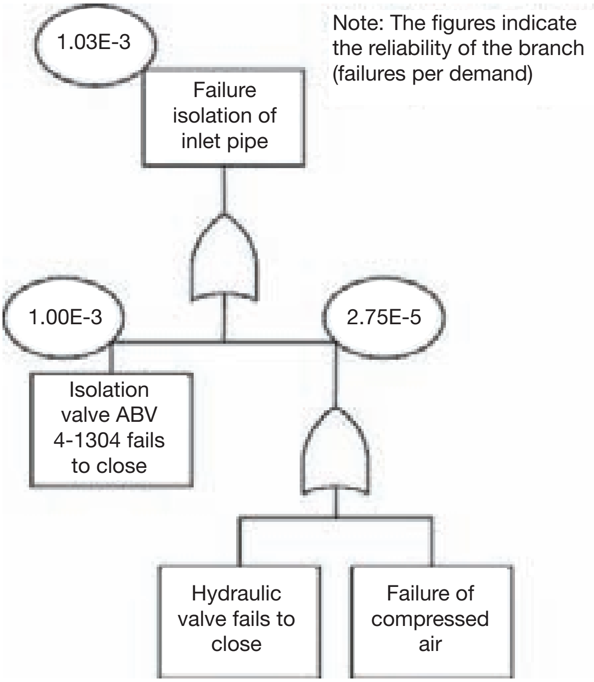


FIG. 2. Part of the fault tree for the inlet isolation valve.

The BR2 reactor was designed and constructed in the late 1950s. During its lifetime, a number of modifications to the installation were implemented. For this reason, the interconnection between different systems such as electricity, instrumentation and compressed air is complicated and is not always according to the modern safety standards as published by the IAEA (see Ref. [4]). This also makes fault trees of this system rather complicated. On the other hand, the fault trees are useful to gain insights into these complicated systems and to make it possible to check that sufficient redundancy for the safety systems is available.

Obtaining good reliability data for the different components is a major problem, although efforts have been made to collect information [5, 6]. A great variety of research reactors exists which differs in design, thermal power and utilization. This makes it difficult to use generic data from other installations. In some cases, even home equipment is in use. For older installations, it is possible to use historical data, but even for the oldest installations, this still gives very conservative estimates of reliability in the case where there are components with very few failures. The same remarks can be made on data about the frequencies of the initiating events.

The uncertainties about the reliability data and about the frequency of the initiating events do not make the results of the PSA useless. Care must be taken to use a consistent set of data. In this way, the PSA will still indicate which are the most important components and give potential dependent failures. This information is useful to determine the priority of efforts to improve safety.

ACKNOWLEDGEMENT

The detailed fault trees of the support systems and the final conclusions of the PSA were developed with the help of the Belgium company Tractebel Energy Engineering, Nuclear Engineering, Safety and Systems Section, with special thanks to A. D'Eer.

REFERENCES

[1] KOONEN, E., JOPPEN, F., GUBEL, P., "Safety challenges encountered during the operating life of the almost 40 year old research reactor BR2", Topical Issues in Nuclear Safety (Proc. Int. Conf. Vienna, 2001) IAEA, Vienna (2001).

[2] POULEUR, Y., "The probability safety assessment impact on the BR2 refurbishment", Safety and Reliability, Nuclear Engineering (Proc. 3rd Int. Conf. Kyoto, 1995), JSME/ASME, Kyoto (1995).

[3] VERBOOMEN, B., "The BR2 Probabilistic Safety Assessment Project", ibid.

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Code on the Safety of Nuclear Research Reactors: Design, Safety Series No. 35-S1, IAEA, Vienna (1992).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Manual on Reliability Data Collection for Research Reactor PSAs, IAEA-TECDOC-636, IAEA, Vienna (1992).

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, Generic Component Reliability Data for Research Reactor PSAs, IAEA-TECDOC-930, IAEA, Vienna (1997).

BLANK

ENGINEERING CONSIDERATIONS AND EXPERIENCE RELATED TO REFURBISHMENT AND SAFETY UPGRADING OF THE 40 MW(th) CIRUS RESEARCH REACTOR

A.V. KHARPATE
Bhabha Atomic Research Centre,
Trombay, Mumbai, India
Email: avk@apsara.barc.ernet.in

Abstract

CIRUS, a 40 MW(th) research reactor located at the Bhabha Atomic Research Centre (BARC), Trombay, India, went into operation in 1960. After over three decades of satisfactory operation, signs of ageing related degradation started appearing in some of the reactor systems. In view of this, detailed ageing studies of the systems, structures and components of the reactor were carried out from 1992 to 1995, to assess the condition of their health and to identify refurbishing requirements towards life extension and safety upgrading for the continued safe operation and better utilization of the reactor. A long outage of the reactor was taken and refurbishment work was completed. This included in-core components, piping and equipments, civil structures, electrical systems, instrumentation and control systems, several safety upgrades, as well as the development of a few special remote repair techniques.

1. BRIEF DESCRIPTION OF CIRUS

CIRUS is a vertical tank-type research reactor of 40 MW(th) capacity. The reactor is natural uranium fuelled, heavy water moderated and light water cooled. The reactor core is housed in a calandria, a cylindrical aluminium vessel with aluminium lattice tubes located between the top and bottom tube sheets. The reactor vessel is surrounded by two annular rings of graphite reflector, cast iron thermal shield and a heavy concrete biological shield. On the top and bottom of the reactor vessel, there are water cooled aluminium and steel thermal shields. Concrete biological shields are also placed at the top (see Fig. 1).

Fuel assemblies are located inside the lattice tubes and are cooled by demineralized light water in a closed loop with the coolant flowing from top to

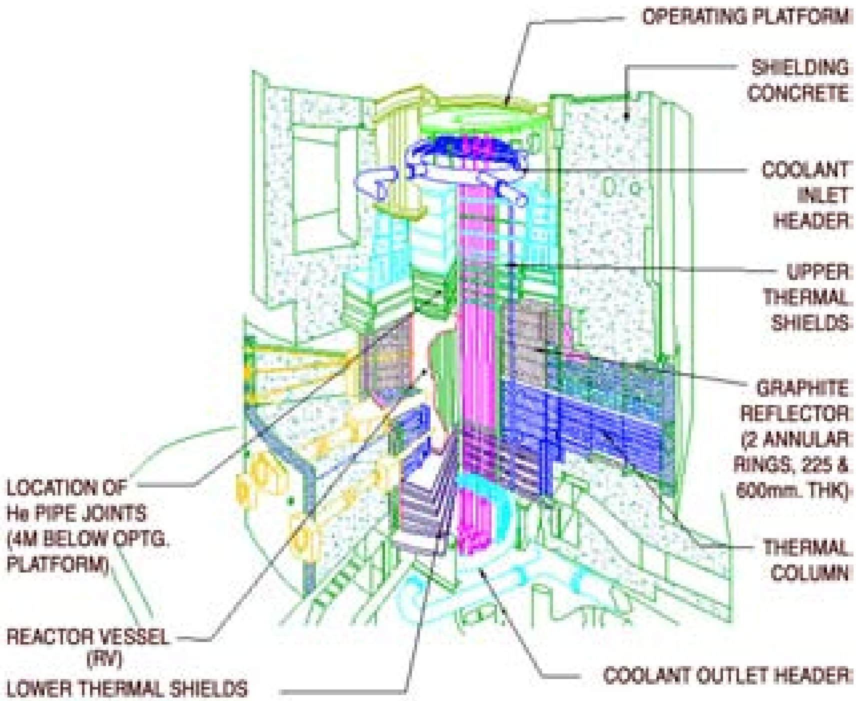


FIG. 1. CIRUS reactor structure cross-section.

bottom. On stoppage of the coolant re-circulation pumps, the reactor is automatically shut down and shutdown cooling is established by one-pass gravity assisted flow of water from an emergency water storage tank (called a 'ball tank', based on its shape), located at a higher elevation than the reactor. The coolant outlet from the core is led to a low level concrete tank ('dump tank') from where water is pumped back to the ball tank using pumps provided with emergency power supply. The heat from re-circulating the primary coolant water, the heavy water moderator and the thermal shield cooling water is transferred to seawater, the secondary coolant in tubular heat exchangers with seawater flowing in a once-through mode.

A once-through ventilation system is provided for comfort, and cooling of some of the reactor structure components. Air flow is maintained from lower radioactive areas to higher radioactive areas and is led to a 125 m high stack through a set of HEPA filters. In the unlikely event of an accident, the containment is boxed up and air is vented through an iodine removal system in a controlled manner to keep the containment pressure below atmosphere during the post-accident management phase.

2. CONSIDERATIONS FOR REFURBISHMENT

CIRUS has been in operation since 1960 and after initial teething troubles, had given a good performance with an average availability of around 70%. However, in the early 1990s, systems, structures and components (SSCs) started showing signs of ageing (see Fig. 2). To assess the residual life of these, detailed ageing studies were undertaken.

The reactor was designed and built to the standards prevailing in the 1960s. Although some safety upgrades were incorporated during its service period, it was also considered necessary to review the safety aspects of the SSCs and upgrade them to current standards to the extent possible.

3. AGEING STUDIES

Several items were identified for conducting detailed studies and were broadly grouped as:

— In-core components;
— Safety systems;
— Important out-of-core components;
— Civil structures.

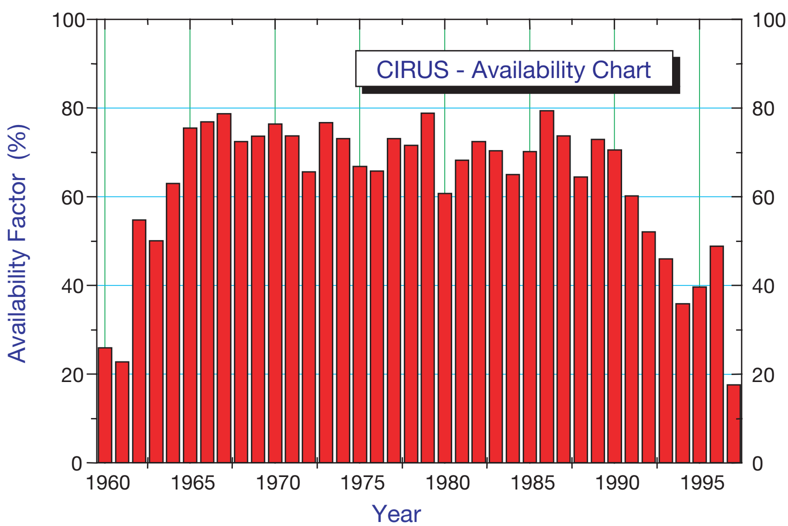


FIG. 2. CIRUS availability chart.

3.1. In-core components

3.1.1. Reactor vessel (RV)

The reactor vessel (RV) is a vertical cylindrical shaped vessel made of an aluminium alloy. Between the shell and the top tube sheet, an expansion bellow is provided to take care of thermal expansion and contraction. The RV is provided with 199 vertical lattice tubes.

The effect of irradiation on RV material was studied on equivalent irradiated samples of similar material. Results showed the tendency of saturation in tensile strength and reduction in elongation to 3.7%, but local ductility of 42% in area reduction was retained. It was concluded that RV can be considered healthy for many more years of operation.

Remote visual inspection of the RV tubes and their eddy current testing for wall thickness monitoring and volumetric examination for flaw detection was done. The condition of the tubes has been found to be good and no unacceptable flaws have been detected.

The expansion bellow of RV joins the vessel shell to the top tube sheet by helium tight lap weld. The bellow has been exposed to neutron fluence and subjected to cyclic thermal stresses. A finite element analysis was performed to assess fatigue life of the bellow for the fluctuating thermal load. The fluctuating stresses in the bellow were assessed to be well below the endurance limit of the material. From these studies, it was concluded that there was no necessity for replacing the reactor vessel.

3.1.2. Thermal shields

Assessment of the steel and aluminium thermal shields comprised remote visual examination and pressure testing. The outer surfaces of the shields and the inner surfaces of the lattice tubes fitted in the shields were found to be healthy. A small leak was observed from the top aluminium thermal shield at the joint between one of the two coolant inlet pipes and the shield tube sheet. This was repaired by a special technique, as explained in Section 4.4.2.

3.1.3. Graphite reflector

The air cooled graphite reflector around the RV is in two annular segments and undergoes concurrent annealing with the reactor in operation dissipating heat to coolant air flowing between the inner and outer segments. Detailed studies were carried out for estimating the stored energy in the graphite reflector, especially since the reactor had seen prolonged operation at

20 MW(th), and it was apprehended that concurrent annealing at this power level may not be adequate. Sample blocks were removed from the reflector and stored energy was estimated using differential scanning calorimetery. The reactor was subsequently operated at 40 MW(th) for a short period and fresh samples were taken and analysed. These studies showed that the stored energy levels are within acceptable limits and there is no requirement for carrying out the annealing operation using an external heat source.

3.2. Safety systems

3.2.1. Reactor regulation and protection system

Reactor power is monitored by ion chambers and power regulation is done by moderator level control. The reactor protection system comprises signal generation and transmission units and shutdown devices in the form of six nos. fast-acting boron carbide shut-off rods backed up by partial moderator dump.

Saturation characteristics of the ion chambers were observed to be deteriorating during service. The periodicity of monitoring saturation characteristics was accordingly enhanced. The original signal processing units of the reactor regulating system were based on vacuum tube technology. Component obsolescence and an increasing trend in failure rates of system components was observed during the late 1970s. The units were replaced by solid state devices during 1984. Performance of the new units has been satisfactory.

3.2.2. Shutdown cooling

As part of ageing studies, a test to determine the flow coast-down following a main coolant pump trip showed that the test results conformed with the design intent. Assessment of the integrity of valves in the shutdown cooling flow path was done by pressure testing and using acoustic emission technique.

3.2.3. Power supply systems

A reliability analysis of the class I power supply sources comprising two DC motor generator sets, backup battery banks and class II power supply sources comprising two motor alternator sets was carried out using performance data and found to be satisfactory. Reliability analysis of the emergency diesel generator (DG) sets in the class III power supply system indicated slightly lower reliability. On analysis, this was shown to be primarily on account of problems with the circuit breakers of the DG sets which have been replaced with new ones of higher reliable design.

3.3. Important out-of-core components

3.3.1. Primary coolant system

The major components of the system comprise five nos. centrifugal pumps, six nos. shell and tube-type heat exchangers and carbon steel pipe lines with associated valves etc.

Piping: Towards assessing the condition of the inside surface of the coolant loop piping, a pipe piece from the hot leg of the coolant loop was subjected to metallurgical examination. The thinning of the pipe wall corresponded to a corrosion rate of about 0.035 mpy compared with the design value of 0.1 mpy. Microstructure of the surface also did not show any evidence of preferential attack.

The stainless steel ring header at the coolant inlet to the reactor is subjected to transient pressures during the starting of the main coolant pumps. The header and metallic expansion joints joining the header to the inlet piping were ultrasonically tested and found healthy. Cross-headers are provided across the ring header and trunion valves mounted on the cross headers are connected to individual fuel assemblies for providing coolant inlet. A similar arrangement exists at the bottom end of fuel assemblies to lead the coolant out. An in-situ metallographic examination on these cross-headers revealed the presence of the sensitized microstructure near a weld joint which had developed a crack on one of the bottom cross-headers. Further investigations revealed that two pipe sections of dissimilar metal and of different grain sizes had been welded together and this had led to the development of a crack on the cross-header body. Repairs were carried out with considerable effort due to difficult site conditions.

Pumps: The main coolant pumps are horizontal centrifugal pumps coupled to motors through gear elements used as speed increasers. All pump assemblies are provided with fly wheels for improving the flow coast-down characteristics following pump trip. The pump sets are mounted on the floating foundation using cork pads.

In the last few years, the maintenance efforts on these pumps had increased, primarily due to frequent failure of the gear elements. Vibration levels on the pumps were also found high. Isolation efficiency of the cork pads was measured and found to have deteriorated significantly. The cork pads of all the pumps were replaced to restore the isolation efficiency.

Heat exchangers: Severe erosion of the cupro-nickel tubes of the primary coolant heat exchangers was observed, mainly in the section of the tube bundle facing the primary coolant inlet nozzle on the shell. Metallographic examination of the tubes indicated denickelification of the tubes. Studies indicated that the failure was caused by impingement due to high fluid velocity. All tube bundles were replaced. The heat exchanger shells, however, were found healthy by ultrasonic testing.

3.3.2. Heavy water and helium system

Heavy water is used as a moderator in the reactor with helium serving as cover gas. The system consists of the reactor vessel, heavy water tanks, circulating pumps, shell and tube-type heat exchangers, a helium gas holder, vapour recovery units, recombination units and heavy water and helium purification units.

Piping: Heavy water and helium system piping and components are made of SS 304, except for the reactor vessel and tubes of freezer dryers used for removing heavy water vapour from the cover gas, which are of aluminium. The pH of system heavy water is maintained around 5.8 with a view to minimize corrosion of the aluminium components. On a few occasions, the tubes of freezer dryers had developed leaks leading to the ingress of trichloro-ethylene into the system which is used as the cooling medium in the freezer dryers. Under irradiation, trichloro-ethylene decomposes and the resulting chloride contamination can cause stress corrosion cracking of SS components.

A few pipe pieces were removed from the helium system for metallographic studies. Microstructural examination near welds indicated carbide precipitation and a few fine cracks up to a depth of 25 m. Replacement of the affected pipe sections has been done as a part of refurbishing. Welds on piping carrying heavy water were also examined but no significant flaws could be detected.

Heat exchangers: There are three heavy water heat exchangers where the system D2O flows on the shell side and is cooled by seawater flows through the tubes. The tubes are of duplex type, the inner tube carrying seawater is of cuprous-nickel and the outer tube is made of SS material. In 1980, pin holes were observed on the SS shell of one of the heat exchangers. Chloride induced stress corrosion cracking, initiating from outside the surface was attributed as the cause. The source of chloride was identified as seawater splashing on the surface during venting of the seawater side of the heat exchangers. The defective shell was replaced by a new shell fabricated out of SS 316 L material. Eddy current testing of the tubes of the heat exchangers indicated no significant degradation.

Heavy water storage tanks: There are three SS 304 tanks in the system of storage capacities ranging from 8 to 20 t of heavy water. The condition of these tanks was checked by ultrasonic testing of all the weld heat affected zones and no significant defects were revealed.

Flanged joints between aluminium extension pipes of RV and system SS piping. There are eight flange joints with elastomer gaskets between the aluminium tubes extending from the top of the reactor vessel and SS helium system pipe lines, located above the top of the upper steel thermal shield about 5m below the operating platform in an inaccessible area. The system helium leak rate, which was of the order of 2 m^3/d had gradually increased to 8 m^3/d. Most of the leakage was found to be from these flange joints.

3.4. Civil structures

All major structures were inspected and observations have been documented for future comparisons. The general condition of the reactor building, annulus building, reactor structure block, storage block, in pile block and dump tanks of main coolant system have been found to be satisfactory.

Reactor containment is a carbon steel dome shaped structure made of 20 mm thick plates cylindrical shell and 12 mm thick hemispherical top. This is covered by an aluminium reflector with a 3 m annulus gap. The containment structure was inspected visually and, at select locations, ultrasonically. The containment structure was found to be sound.

A detailed assessment of the reactor ventilation exhaust stack made of reinforced concrete and the ball tank made of pre-stress concrete, including extraction of concrete core samples for the assessment of compressive strength, carbonation depth, chloride and sulphate depths, etc., ultrasonic pulse velocity test by surface probing at selected levels to assess the homogeneity of surface concrete, corrosion potential measurement at selected levels and rebound hammer test to assess the integrity of cover concrete were carried out. The results of these tests indicate that the structures are in a sound condition. The bulk concrete is in reasonably alkaline condition and sulphate and chloride contents are below the threshold levels indicating low potential for corrosion in near future.

Since the major structures, such as the reactor containment building, the ball tank, dump tanks, etc. were built to standards prevailing at the time of their construction in the 1950s, a detailed seismic analysis of these structures was carried out. It was found that the structures meet the current standards, except the ball tank which needs additional reinforcements at the central shaft region.

4. REFURBISHMENT

Immediately after the start of the extended reactor outage for refurbishing, irradiated fuel from the core was unloaded. Several additional inspections were then performed, which were not possible earlier with fuel residing in the core, and further refurbishing requirements were identified. Radiation fields at various locations in and around the reactor vessel were measured after core unloading and at different periods of radioactive decay after the shutdown. This information is considered valuable towards planning for any future decommissioning of the reactor.

4.1. Piping

A large part of the carbon steel piping of the main coolant loop is laid subsoil at a depth of 4 to 5 m below grade, with pipe sections joined by couplings using elastomer sealing rings. Pressure testing of pipelines revealed a small leakage in one pipe and was suspected to be from one of the coupling seals. Techniques such as Pearson survey and acoustic emission were tried for locating the leak but were not successful due to the interconnection of several pipes leading to the loss of electrical and acoustic signals. Due to the large soil cover over the piping, the use of the isotope tracer technique was also found to be not feasible for the purpose. A fluorescent chemical dye was injected into pipes to locate the leaks. This method was successful. As expected, leakage was found from a coupling on one of the main coolant re-circulation pipes.

To enable a detailed inspection of the pipes and their protective coatings, large-scale excavation was carried out to expose all pipes. The condition of large diameter piping was found to be good but small diameter pipes were seen to have surface pittings, especially near the ends of the pipe sections. The protective coating was intact but its condition had deteriorated due to ageing. All small diameter pipes were replaced. Elastomer sealing rings on all couplings were replaced and pipes were pressure tested. The new sealing rings were formed by in situ vulcanization to avoid the need for any cutting operations for the large diameter pipes. Special gadgets were developed for in situ vulcanization. Old protective coatings made of tar-felt and bitumen were removed, the pipe surfaces cleaned by high pressure water jets and a new protective coating made of self-adhesive modified rubber bitumen material was cold applied in two layers. Couplings were separately coated with water-repellent mastic compounds.

Some sections of the cover gas system SS piping that were found to have undergone intergrannular stress corrosion cracking were replaced. Several other pipes of the auxiliary systems, such as service water, chilled water and machinery cooling water, which had shown ageing related degradation, were also replaced.

4.2. Equipment and structures

Major equipment that was refurbished during the outage are main coolant pumps and main coolant heat exchangers. The gear elements of main coolant pumps speed increasers have been replaced with that of a modified design to reduce vibration levels. The cork pads of the pump foundations have been replaced to restore the isolation efficiency. Primary coolant heat exchanger tube bundles were replaced. The corroded support structure of the heat exchangers was also replaced. In addition, this opportunity was also utilized for carrying out the major overhauling of mechanical handling equipment, fuel handling machine and emergency power supply diesel generators. A large number of valves in various reactor systems, especially those that are not easily removable during reactor operation, were also serviced and examined by magnetic particle tests for any flaws.

A complete overhaul of the air conditioning equipment that supply conditioned air to the reactor building and control room was carried out. Several ventilation air ducts and dampers, where ageing related degradations were seen, were repaired or replaced. For carrying out refurbishing of reactor building ventilation and air conditioning equipment, their outage for a significant period was required. In order to permit the continuation of various other refurbishing jobs in the reactor building during this outage, a temporary filtered air supply system was installed.

Visual inspections, integrity checks and checking of the metallic linings of various drain collection sumps by non-destructive techniques was done and necessary repairs were carried out. The oil-filled radiation shielding window of the isotope handling facility was replaced by a new dry-type window assembled in-house.

Seawater is used as a secondary coolant in CIRUS and is pumped from a pumphouse located at the end of a 1 km long jetty in the bay. Return flow of seawater from various heat exchangers is led back to the sea through a pipe header into a well-shaped concrete structure called stilling pool. The piping and other steel structures inside the stilling-pool had corroded significantly due to long years of service in a corrosive marine environment. All these components were replaced.

The ball tank has a bottom cupola and is supported on a conical shaped concrete support tower. Through the bottom cupola, a vertical concrete central shaft extending over the maximum water height in the tank is provided for inspection purposes. A few years ago, a small leak had developed near the bottom of this shaft. Repair attempts by pressure grouting from the dry side of the shaft with the tank filled with water did not succeed. During the refurbishing works, the tank was emptied and repairs were carried out from inside. In order to enable work inside the confined space of the tank using epoxy compounds for a final coating after repairs, special ventilation arrangements were established.

4.3. Desalination unit

A low temperature vacuum evaporation method based desalination unit had been developed by the Desalination Division of BARC. A desalination unit of 30 Te/d, which would use low temperature waste heat from the reactor to heat seawater to a temperature of about 55C has been coupled. The heated seawater fed to this unit will be desalinated and further purified to generate demineralized water for reactor usage. This is done to serve as a demonstration of using low grade heat from a research reactor for the purpose of desalination.

4.4. Special repair works

4.4.1. Remote repair of helium flange joints

As explained earlier, some of the flange joints (tongue and groove) between 50 NB aluminium pipes connected to RV and SS helium system pipes were found to be leaking. These flange joints are situated deep inside the reactor structure at a distance of about 5 m below the operating platform in a 200 mm vertical gap between the steel thermal shield and concrete biological shield above the reactor vessel.

The operating experience had shown that the helium leak rate during reactor operation was much less than the leak rate during reactor shutdown due to the thermal expansion of pipes which bring the flanges closer, thereby reducing the leakage. This also confirmed that the gaskets still retained sealing capability. The same was verified by testing the flange joint with irradiated gaskets in a full-scale mock-up station. Additionally, tests were also conducted using very old and deteriorated gaskets. These tests confirmed that existing system gaskets at the site are still capable of providing leak-tight joints provided the flanges. Accordingly, the work on development of a suitable technique to tighten the flanges remotely was taken up. Among the various alternatives, design of an encircling split clamp was preferred.

Challenges: Severe limitations of access and availability of only small-size openings in the reactor structure posed a serious challenge for the development of a remote repair technique:

(1) Repair work had to be carried out remotely from the top of the reactor, which is about 5 m above the leaky flange joints.
(2) Split sealing clamps of 175 mm diameter were required to be fixed around the leaky flanges. The maximum size of the opening available for lowering the clamps was a diameter of 150 mm.
(3) As the leaky flange joints were located away from the central thimble, the clamps had to be dragged sideways from about 1.2 m to reach the flange location. The sealing clamps had to be turned around the flange joints to properly orient them for fixing around the leaky flanges.
(4) The clamps had to be tightened remotely around the flange joints.

Design and development: The following design and development work was undertaken to meet the challenges mentioned above (see Fig. 3):

(1) Sealing clamps: Special split sealing clamps with tapered inner faces were developed. The clamps were split into two halves, connected by a double hinged joint so that they could be lowered to the required elevation through the 150 mm diameter central hole of the reactor. Ball rollers were fixed at the bottom of the clamps to facilitate dragging.
(2) Tightening and gauging tool: A tightening tool with a universal joint, spring-loaded telescopic tube and bevel gear arrangement was developed for tightening the clamps around the flanges. As permissible tolerances for flange tightening were small, dimensions of individual flanges were measured with specially developed gauging tools.

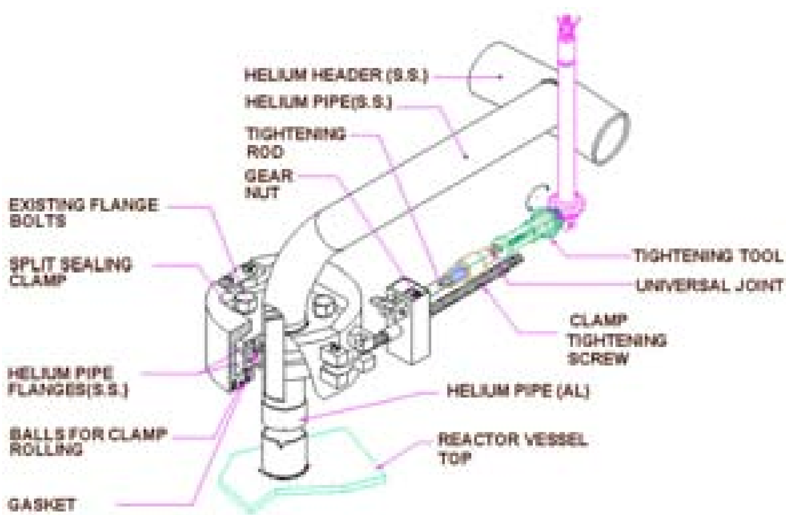


FIG. 3. Design and development undertaken.

(3) Rope mechanism: All activities of lowering, shifting, manoeuvering and enveloping the flanges with the clamps were done with nylon ropes. The entire operation was somewhat akin to manipulation of marionettes in a puppet show.

Implementation: A full-scale mock-up station was set up (see Fig. 4), for qualifying and calibrating the tools and procedures for remote positioning of the sealing clamps around each flange joint and their final tightening. Sealing claims with specific dimensions were made to suit each flange. The mock-up was used extensively for calibrating the sealing clamps and training of personnel for handling during removal. Miniature video cameras and light assemblies installed in the 8-in. gap at flange locations were used for viewing the work area on monitors placed at the top of reactor.

In order to ensure that the weld joints between the aluminium pipes and the reactor vessel top tube sheet were not overstressed, a detailed analysis was carried out to finalize the tightening sequence and the required compression. After completing the tightening of all flange joints to the permitted extent and

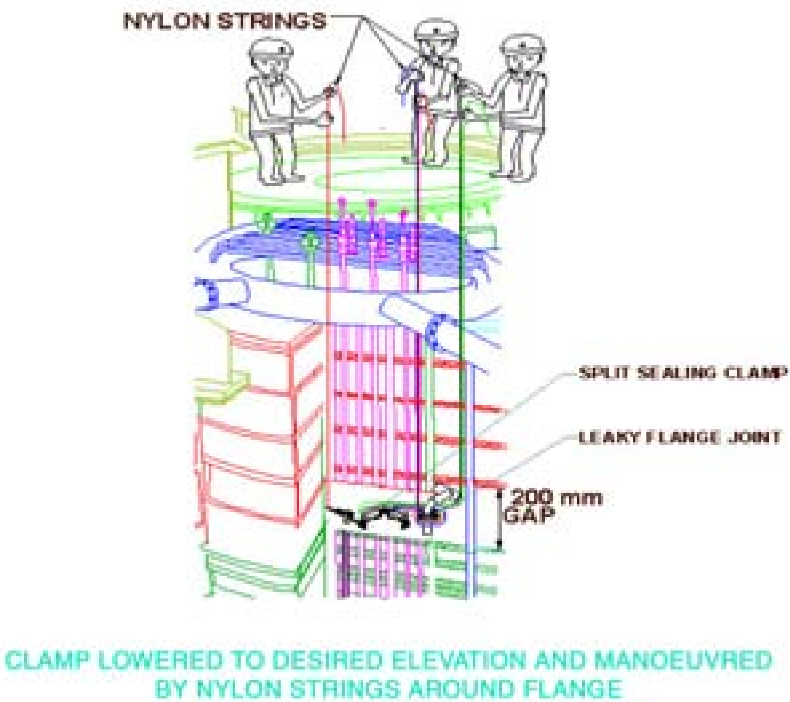


FIG. 4. The full scale mock-up station.

in a specified sequence, helium leakage from the flange joints came down to less than 20%.

4.4.2. Rectification of leak from the upper aluminium thermal shield

As indicated in Section 3.1.2, a small leak was observed from the upper aluminium thermal shield. Detailed investigations revealed that the leakage was from the weld joint of one of the two 32 NB Sch. 80 aluminium coolant inlet pipes connected to the upper aluminium thermal shield.

Location: The exact location of the leak was identified by monitoring the stabilized water level, in the vertical inlet pipe under stagnant conditions. It was confirmed by installing an inflatable seal arrangement inside the leaky pipe and inflating the seal at different elevations to isolate the leaky location and ensuring that the leak was completely stopped. This location was about 5.5 m below the operating floor of the reactor (see Fig. 5).

Examination: The pipe was visually inspected with the help of a fibroscope and microvideo camera to assess the condition of the inner surface. Eddy current testing was also carried out for volumetric examination. A length of about 65 mm of the pipe around the leaky region was found to be eroded.

Design constraints: The design should meet following:

(1) The leak location being inaccessible, the repair was to be carried out remotely.

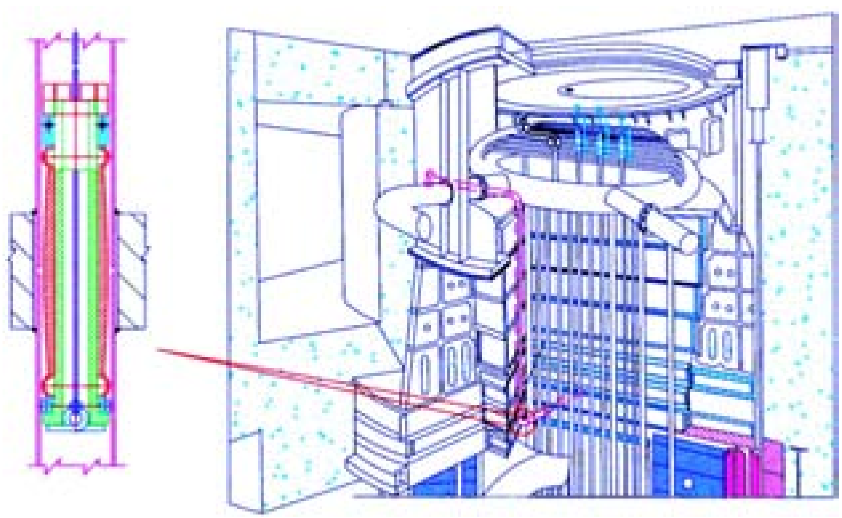


FIG. 5. Rectification of the leak from the aluminium thermal shield.

(2) The design flow through the pipes was to be maintained.
(3) The vertical gap for insertion of any component into the pipe was only 150 mm.
(4) The defective length of 65 mm was to be covered.

Design, development and installation: A hollow aluminium plug, with expandable rings, having a C-shaped cross-section at both ends and a straight portion in the middle to cover the defective region, was developed and qualified in a mock-up station after extensive trials. In view of the site constraints, a flexible hollow link assembly was engineered, for installing the plug remotely. The inner surface of the pipe was cleaned, using an emery brush and a deburring tool. The plug was then installed covering the leak area and the rings were expanded by remote tightening. The shield was hydro-tested satisfactorily.

5. SAFETY UPGRADES

CIRUS, being a reactor of vintage design, did not have some of the safety provisions incorporated in modern research reactors. The long refurbishing outage was, therefore, utilized for carrying out certain safety upgrades. Some of these are listed below.

- A new fire detection system and other fire safety measures, such as fire barriers and fire-retardant coating on cables were incorporated.
- The ball tank central shaft region was strengthened by providing additional metallic reinforcements to meet the seismic requirements.
- Strengthening of the support to many pieces of safety related equipment and valves was carried out to qualify them for safe shutdown earthquake criteria.
- Physical separation of some of the safety related components was done to guard against common cause failures due to fire, flooding, etc.
- An emergency ventilation exhaust system of the reactor building was earlier provided with an alkali scrubber and silver coated copper mesh filters for trapping radioiodines under accident conditions. These have been replaced by the more efficient and easy to maintain combination filters made of activated charcoal and high efficiency particulate air filters.
- The failed fuel detection (FFD) system installed in CIRUS comprised compressed air strippers for stripping gaseous fission products from water samples drawn from the coolant outlet of each fuel channel and its on-line

monitoring for beta activity. Performance of this system was not up to design intent and maintenance of system components required excessive effort and manrem consumption. In view of these problems, a new FFD system has been installed. In the modified system, water samples are led into metallic chambers after providing sufficient delay to permit the decay of short lived activity and are monitored for gamma radiation.

6. CONCLUSION

Extending the life of research reactors involves several engineering considerations, including a well planned ageing study of the SSCs. While specific reactors of different designs will need different approaches, some of these can be generalized. Special techniques may also need to be devised for specific problems. A refurbishment plan based on these considerations will also have to take into account the techno-economic considerations. The experience of the refurbishment of CIRUS had shown that the useful life of the reactor has been extended at a fraction of the cost of a new reactor with enhanced safety.

ACKNOWLEDGEMENTS

The author gratefully acknowledges the support rendered by the colleagues from the Reactor Group and other divisions of the Bhabha Atomic Research Centre in the preparation of this paper. The author wishes to express deep gratitude to S. Sankar, Director, Reactor Group and R. Chowdhury, Associate Director, Reactor Group for their constant guidance and encouragement throughout the preparation of this paper.

COMPREHENSIVE IN-SERVICE INSPECTION OF KARTINI REACTOR TANK LINER AND FITNESS FOR CONTINUED OPERATION

S. SYARIP
Reactor Division of R&D Center for Advanced Technology,
Yogyakarta, Indonesia
Email: syarip@batan.go.id

P. STATHERS
Australian Nuclear Science and Technology Organisation (ANSTO),
Lucas Heights, Australia

Abstract

The preliminary comprehensive in-service inspection of the Kartini reactor core and tank liner by using non-destructive testing (NDT) methods to maintain its safe operation, have been done. The type of NDT used were visual examination using under-water camera and magnifier, replication survey using dental putty, hardness test using an Equotip-D indentor, thickness test using ultrasonic probe, and dye penetrant test. The visual examination showed that the surface of the reactor tank liner was in good condition. The hardness readings were considered to be consistent with the original condition of the tank. The slight increase in hardness at the reactor core area, and this is consistent with the neutron fluence experienced, i.e. ~10^{15} n/cm^2, where high fluence will cause higher increased hardness of the material. Results of an ultrasonic thickness survey showed that, on average, the tank thickness is between 5.0 mm and 6.5 mm, a low 2.1 mm thickness exists at the top of the tank liner in the belt area. But, because the area is a double layer aluminium plat, its decrement in thickness does not influence the reactor safety. The replica and dye penetrant test at the low thickness area and several suspected areas showed that there could be some defect from the original manufacture. Therefore, it can be concluded that the reactor tank liner is still feasible for continued safe operation.

1. INTRODUCTION

For the safe operation of nuclear reactors, it is mandatory to examine the reactor components periodically, and this examination, carried out during the service life of the component, is called an in-service inspection. The Kartini reactor has been in operation since 1979, and its operation licence will end in

November 2005, and it is expected that the operation will continue for another 10–15 years. Therefore, it is essential to ensure the safety aspect of the reactor, and a comprehensive inspection/assessment is thus required.

This preliminary inspection activity covers tank wall, tank base and reactor internal components, and is comprehensively conducted for the first time after 22 years of reactor operation. The main purpose of this activity is to identify the possibility of corrosion effects, defects, deposit accumulation, and other indications which will affect the safety of reactor operation. The method used is non-destructive testing (NDT) methods.

2. DESCRIPTION OF THE KARTINI REACTOR TANK LINER

The Kartini reactor tank liner is made of aluminium material from an abandoned IRT-2000 reactor tank, where the IRT-2000 research reactor project was abandoned in 1964. This aluminium tank wall was chemically analysed and found to be consistent with aluminium association AA 1050 (99.5% aluminium). The function of this reactor tank is as a *reactor tank liner,* i.e. as a liner between reactor coolant water and concrete. The original thickness of reactor tank liner is 6 mm, with 6250 mm height, and 2000 mm diameter. The tank liner and its weld joints is shown in Fig. 1.

As shown in Fig. 1, vertically, the tank liner has three joints; the joints were welded in such a way that they formed a belt. Besides for a double layer joint, the belt also has a function for strengthening the joint. The inner side of this reactor liner always contacts with reactor coolant water. The condition of reactor coolant water always kept in the following technical specification: pH between 5.5 and 6.5, conductivity 6 M/cm, impurity (especially for Na, Ca, Mg, and Si content) less than 1 ppm.

3. METHOD

The reactor was shut down for more than two months, the reactor core was emptied of its fuels and control rods, then the reactor pool tank was drained of water to a level of 0.5 m above core grid. A layer of lead shield was used to cover the top of the reactor core region in order to reduce the exposure rate. The measurement is done manually by using NDT methods, i.e. visual examination, hardness test, thickness test, replication test and dye penetrant test.

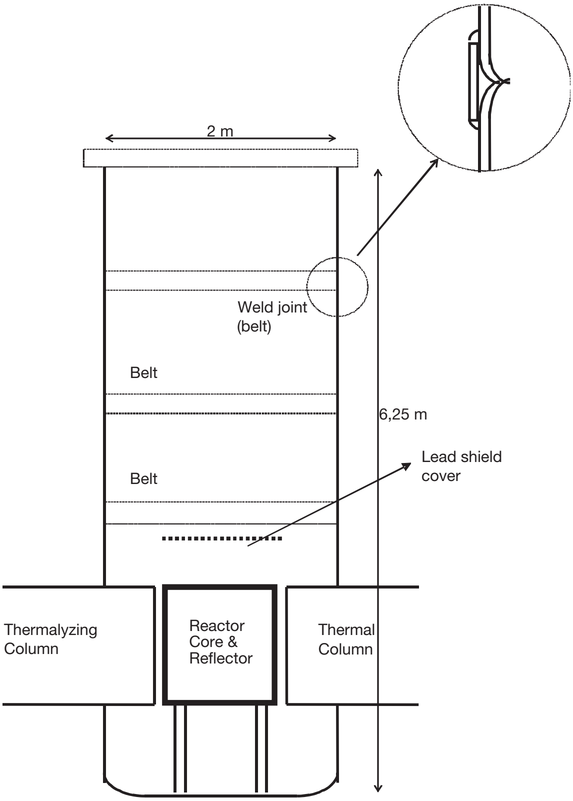


FIG. 1. Kartini reactor tank liners and its well joints.

4. RESULTS AND DISCUSSION

A detailed visual inspection consisted of both direct visual inspection with the aid of a magnifier, and video examination of the core/reflector region of the tank using an underwater camera. The visual examination showed that the surface of the reactor core and tank liner was in good condition, and consistent with other reactors manufactured from the same material and length of service. The surface was coated with a light hydrated oxide consistent with the service environment.

A hardness survey using an Equotip D indentor was conducted; 67 readings were taken and the results indicated that the tank was in the annealed condition. The region of the tank wall at the mid-core region showed a slightly increased hardness, as expected. The hardness readings were considered to be consistent with the original condition of the tank. There was a slight increase in hardness at the reactor core area and this is consistent with the neutron fluence experienced, i.e. ~10^{15} n/cm^2, where the higher the fluence, the higher the hardness of the material, shown in Fig. 2.

An extensive ultrasonic thickness survey of the tank was done at a grid spacing of 5 cm by 5 cm for the wall and 2.5 cm by 2.5 cm for the belt region. A 9.5 mm diameter transducer with 5 MHz of frequency was used for the survey. In excess of 4200 readings were taken of the tank belts and the tank wall. Over 800 readings were taken of the submerged section of the tank. Results of the ultrasonic thickness survey showed that, on average, the reactor tank thickness is between 5.0 mm and 6.5 mm, a low 2.1 mm thickness exists at the top of the tank in the belt area. But, because of the area is a double layer aluminium plat, its decrement in thickness does not influence the reactor safety.

Dye penetrant testing was conducted on the accessible welds on both the belts and vertical welds. The survey revealed small welding defects approximately 10 mm long. Examination under a geological magnifier (times 10) indicated that the defect has been in the tank since the time of original manufacture. The location of the defect is consistent with the condition expected to cause a hot tear. The defect is at the transition of a thick weld metal section where excessive welding heat has caused contraction stresses to tear the weak hot metal of the weld cap during the cooling of the weld metal. Replicas were taken of the defect for future reference. The replica and dye penetrant

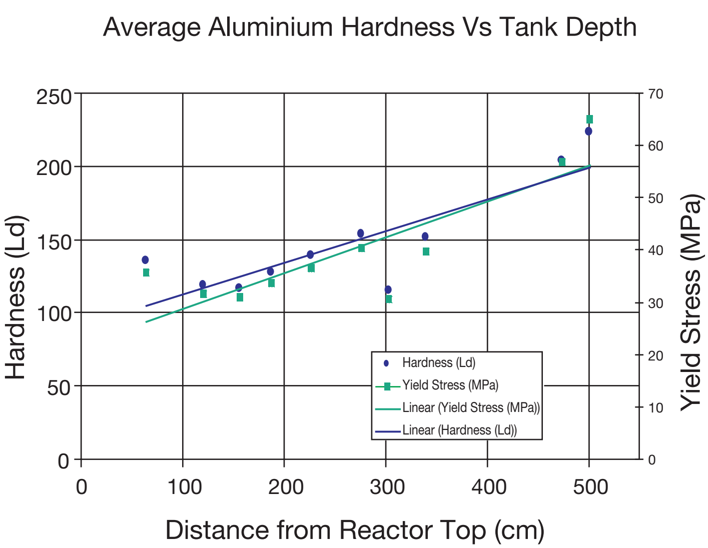


FIG. 2. Reactor tank liner hardness as a function of the depth from the top.

test at the low thickness area and several suspected areas showed that it could be some defect from original manufacture. A sample of the replication test is shown in Fig. 3.

The radiation exposure in the NDT locations surrounding the reactor tank during NDT activities is shown in Table 1. With this condition, the radiation doses acceptable for personnel are the following: 12 persons got greater than 10 m rad and the highest is 42 m rad (one person).

5. CONCLUSION

From the inspection results, it can be concluded that the tank wall thickness is predominantly in original condition. A few thickness readings were found to be low, however, the origin of these low readings is uncertain and could indicate some defect from original manufacture. Even if leakage occurs in the belt or wall, the leakage would not cause a sudden loss of water. It would be a very small and slow leak. The small defect located in the upper belt weld will not cause a fracture and is an original manufacturing defect most likely confined to the weld cap, and this is a hot tear from the welding operation.

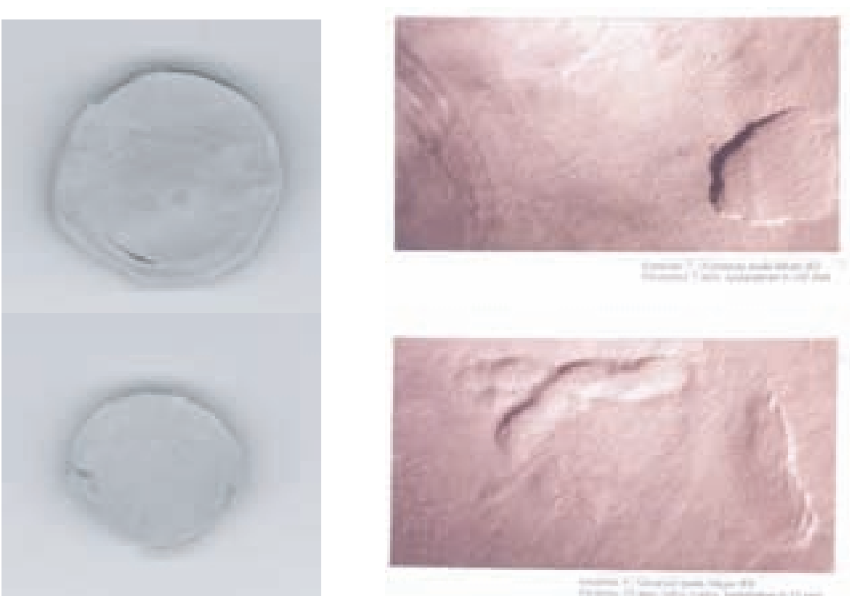


FIG. 3. A sample of the replication test (negative and positive film).

TABLE 1. THE RADIATION EXPOSURE AROUND NDT AREAS OF THE KARTINI REACTOR TANK

No.	Location (reactor water tank is drained up to 2 m deep)	Radiation exposure (m rad/h)
1	Reactor deck	0.2
2	Upper part of reactor tank	0.5
3	Above the reactor core	10.0
4	At the test area (without lead shield)	7.0
5	At the test area (with lead shield)	3.0

The general conclusion is that the reactor core structure and tank liner is still feasible for continued safe operation. However, based on some of the recommendations gained from this work, as well as to monitor and access ageing and corrosion effects on the tank liner, a more detailed comprehensive inspection is needed in the near future.

REFERENCES

[1] SYARIP, S., TEGAS SUTONDO, CHAERUL SALEH, "Inspeksi tangki reaktor Kartini dengan metode NDT sebagai upaya peningkatan keselamatan operasi" (in Indonesian), paper presented at the Seminar Teknologi Dan Keselamatan PLTN Serta Fasilitas Nuklir VII, Bandung, Indonesia, 19 February 2002.

[2] ASME, Visual Examination Standards, SD-2563 (Identical with ASTM Specification D2563-70), 1977.

[3] SYARIP, S., "Ageing investigation and upgrading of components/systems of Kartini research reactor", Utilization of Research Reactors (Proc. Workshop), Rep. JAERI-Conf98-015.

DEVELOPMENT OF MODERN SAFE SYSTEMS OF WORK AT THE IMPERIAL COLLEGE REACTOR CENTRE AND THEIR APPLICATION TO NEUTRON DETECTOR TESTING AND NUCLEAR TRAINING COURSES

D.S. BOND, L. HOWARD
Centronic Limited,
New Addington, Croydon
Email: dave.bond@imperial.ac.uk

H.J. PHILLIPS, S.J. FRANKLIN, N.J. CHAPMAN
Imperial College Reactor Centre,
Imperial College of Science, Technology and Medicine, London

United Kingdom

Abstract

The CONSORT design reactor is owned by, and licensed to, Imperial College of Science, Technology and Medicine, and has been in continuous safe operation since 1965. Today, it is the only civil research reactor left in the United Kingdom. CONSORT is designated a low power research reactor and is rated at 100 kW. The paper concentrates on the issues that have been addressed in ensuring that worker doses are maintained as low as reasonably practicable (ALARP), and describes how robust systems have been implemented to ensure that all assessments are in line with this principle. The principles of risk assessment are applied to all operations within the reactor centre and the framework leading to the establishment of the safe system of work will be outlined and discussed. Two case studies are described in detail as examples, showing the importance to the industry of ensuring that a system is in place to allow the work to continue and summarizes the experience gained in the past few years at Imperial College Reactor Centre. These experiences will provide useful information that managers of similar facilities may wish to consider in formulating their own arrangements. The CONSORT reactor provides an open beam tube facility for the calibration and periodic testing of neutron flux detectors (primarily fission chambers and ion chambers). The first case study outlines the detector testing programme that takes place on the CONSORT reactor using this facility. The second case study relates to the use of this integrated approach to safety management in the teaching environment. With ‘safe systems of work’ in place, it is now possible to reinstate the popular student experiment of directly

viewing the Cerenkov radiation emitted from the critical CONSORT reactor core. This particular activity had been discontinued for some years on ALARP grounds.

1. INTRODUCTION

Imperial College, in the United Kingdom, operates the smallest of the nuclear licensed sites in the Imperial College Reactor Centre (ICRC). (Since January 2003, the University has operated under the title of Imperial College London, although the University's legal entity remains Imperial College of Science Technology & Medicine—ICSTM.) The ICRC houses the CONSORT reactor which has been in continuous safe operation since the first criticality was established on 9 April 1965. The reactor is a low power, light water moderated research reactor, operating with an aluminium-uranium MTR design fuelled core and a full power capability of 100 kW. Although the operation of the reactor is different from, and involves risks of much lower magnitude than, nuclear power plants and fuel cycle facilities, the activities are governed by the same 36 standard licence conditions of the UK Nuclear Installations Act 1965 [1]. The UK nuclear industry operates under the requirements laid down in this legislation and is regulated by the Nuclear Installations Inspectorate (NII), a department of the UK Health and Safety Executive (HSE).

In addition, the site requires the assessment of 'reasonably forseeable' emergencies under the Radiation Emergency Preparedness and Public Information (REPPIR) regulations [2], and control of the work with ionizing radiation under the Ionizing Radiation Regulations (IRR) [3] and the Radioactive Substances Act (RSA) [4]. These regulations and nuclear site licence conditions lead to more clearly defined sets of procedures and assessments of competency than may be required in other parts of the college or other similar academic institutions.

The Imperial College Reactor Centre is a small department comprising ten permanent members of staff. These members of staff have been recruited either from positions within the nuclear industry, or as a result of their expertise gained in other areas of radiation work, and this experience in radiological issues is considered an essential part in ensuring that all staff members consider the ALARP principle as a primary objective in all their areas of work and study. The reactor is typically operated for a maximum period of seven hours each working day with the key applications for the facility being:

(1) Neutron activation analysis (NAA);
(2) Detector testing and calibration;

(3) Radioisotope production for industry and academia;
(4) Tracers and pharmaceutical development;
(5) Research and teaching.

This paper fully describes how a complete and consistent approach to the ALARP principle, including dose and risk assessments, has been developed to sit within a standard set of documents which allow work to be carried out safely and effectively within the requirements of the UK Ionising Radiation Regulations, 1999, and associated regulations.

2. OVERVIEW OF THE CONSORT FACILITY

In common with other research reactors around the world, the CONSORT reactor was designed to provide training to nuclear operators, as well as a source of neutrons for research, analytical and teaching purposes. This includes the provision of a number of beam tubes and penetrations into the biological shielding. By definition, these penetrations provide potential shielding weaknesses and their use requires a robust mixture of both procedurally based management control coupled with automated interlocks and safeguards. The layout of the reactor facility is shown in schematic form in Fig. 1, and this shows the location of the zero-degree beam tube facility that is used by Centronic Limited as part of their standard detector testing programme. This programme requires the use of a broad range of well thermalized neutron fluxes over 11 orders of magnitude. The background behind the detector testing programme is discussed in detail in Section 4.

3. THE ‘SAFE SYSTEMS OF WORK’ ASSESSMENT PROCESS

All aspects of operations on a UK licensed site are subject to a high degree of regulatory inspection and audit, and this is particularly true in the areas of radiological protection and discharge of radioactive waste. In order to comply with the site licence requirements, it is necessary for all operators to have a comprehensive and well structured set of documented procedures and protocols for the control, monitoring and management of all aspects of operation. Training and record keeping are also covered extensively by the licence. These requirements have led to a highly structured approach to the production of ‘safe systems of work’. A safe system of work at the ICRC comprises the following:

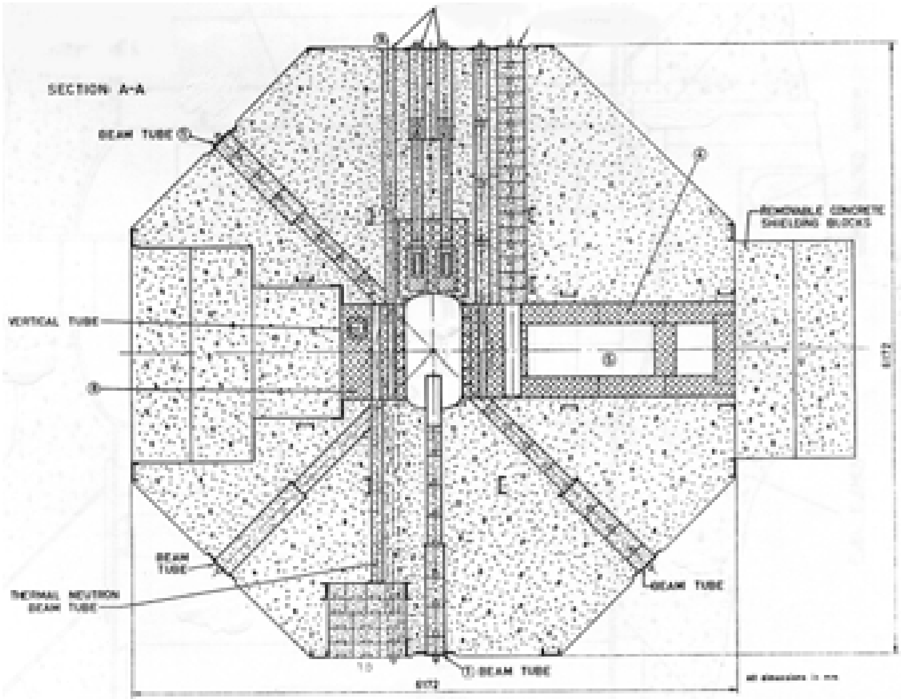


FIG. 1. Layout of the CONSORT irradiation facilities, including the zero degree facility.

— Proposed method statement;
— Risk assessment;
— Written system of work;
— Permit to work.

All standard operations carried out on the site are covered by documented procedures and assessments that are formally issued as a series of standard operating procedures. These procedures undergo peer review and controlled issue and distribution in line with best documentation practice. Tasks and work that is proposed on the licensed site that does not form a part of standard procedure clearly require the same degree of control and assessment in order to adequately comply with the site licence and statutory requirements. These proposals may come from internal members of staff who need to carry out a non-routine operation (routine operations are covered by the standard operating procedures) or from contractors or students proposing experiments or project work.

3.1. Development of safe systems of work

The approach adopted at the Reactor Centre to producing safe systems of work is based on a standard approach of carrying out a preliminary risk assessment of the planned task or operation, taking into account both radiological, nuclear and conventional safety hazards. This initial risk assessment is reviewed and any disproportionately hazardous or high dose-rate aspects of the process further analysed to establish alternative techniques or methods of achieving the task objective. For straightforward tasks that are not especially hazardous, the risk assessment is sufficient to control the operation and provides all the information and instructions for the work to be carried out safely. Operations with a radiological risk, particularly those involving non-classified workers in controlled areas, have additional written controls in the form of a 'written system of work' which details both the radiological and conventional hazards and mitigating safeguards, and will usually include 'hold points'. These 'hold points' will state dose-rate limits to ensure that dose-uptake for the task as a whole cannot be exceeded. The most hazardous tasks (from both radiological and conventional viewpoints) are controlled through the use of permits to work. In all cases, operations which involve the use of contractors, require careful assessment of the contracting organization and their plans for the work (method statements, etc.).

Contractor assessment is carried out in accordance with a standard operating procedure, which requires detailed information on the contractor's health and safety record, insurance arrangements, training of staff, certification, accident record, etc., to be made available. Only when this assessment has been examined and completed to the satisfaction of the Reactor Manager will the contractors be authorized to carry out work on the licensed site. This process is, of course, applied proportionately, depending on the nature and duration of the task and, in some cases, scaffolding or control instrumentation modification, for example, the process will be detailed and extensive. In the case of modifications to safety systems, the contractor will need to be assessed by the full Reactor Centre Nuclear Safety Committee and judged suitably qualified and experienced prior to any work being undertaken. This process allows the creation of procedures to enable personnel to work with an open beam tube within a controlled area, yet still ensure that the doses accrued by those workers are maintained ALARP.

The use of hold points in higher risk operations is core to the structure of the written systems of work. These hold points ensure that the work may only proceed once the Reactor Manager (RM), Reactor Supervisor, Radiation Protection Supervisor (RPS) or the Director of Reactor Operations and Safety (DOS), has reviewed the situation and authorized the work to proceed past that point. In the unlikely event that tasks need to be undertaken where there is a possibility that dose-uptake could approach the internally set dose restraint objective for Reactor Centre staff, then the written system of work is prepared in consultation with the Reactor Centre's Radiation Protection Advisor (RPA). A high degree of procedure and management control has ensured that these situations are extremely rare.

3.2. Preparation of a safe system of work

The process for the generation of a safe system of work is summarized by the flow chart shown in Fig. 2.

The process starts with the submission of a method statement from either internal personnel, contractors or students. This proposed method statement is subjected to an initial risk assessment by the Reactor Safety Officer (RSO). This assessment is reviewed by the Reactor Manager and modified in order to provide a final assessment for the process (it will be necessary to prepare an additional modification proposal if the work involves modification to reactor plant). Once this final risk assessment is complete, a decision is taken as to whether a permit to work and written system of work is required. For many low risk operations, the risk assessment and its associated defined actions is sufficient to adequately control the task. In these cases, all workers will read and signify understanding and compliance by signing onto the resulting action sheet. For other tasks either more hazardous, unusual or involving work in controlled areas by non-classified workers, a written system of work is always required. The function of the permit to work is to ensure that all other interested parties who may be affected by the work are duly notified and aware of the possibility for the work affecting their area. In addition, it allows a formally managed review of work or hazard conditions to be carried out, where the work is taking place over an extended period where the hazards and associated conditions may change during the work.

4. CASE STUDY 1: USE OF THE FACILITY FOR NEUTRON DETECTOR TESTING BY CENTRONIC LIMITED

Centronic Limited, based at Croydon in the UK, was founded in 1945 as 20th Century Electronics Ltd and was initially a manufacturer of Geiger-Müller and Cathode Ray Tubes. 20th Century Electronics grew along with the nuclear industry in the 1940s and 1950s, working closely with the Atomic Energy Research Establishment (AERE) of the United Kingdom Atomic Energy

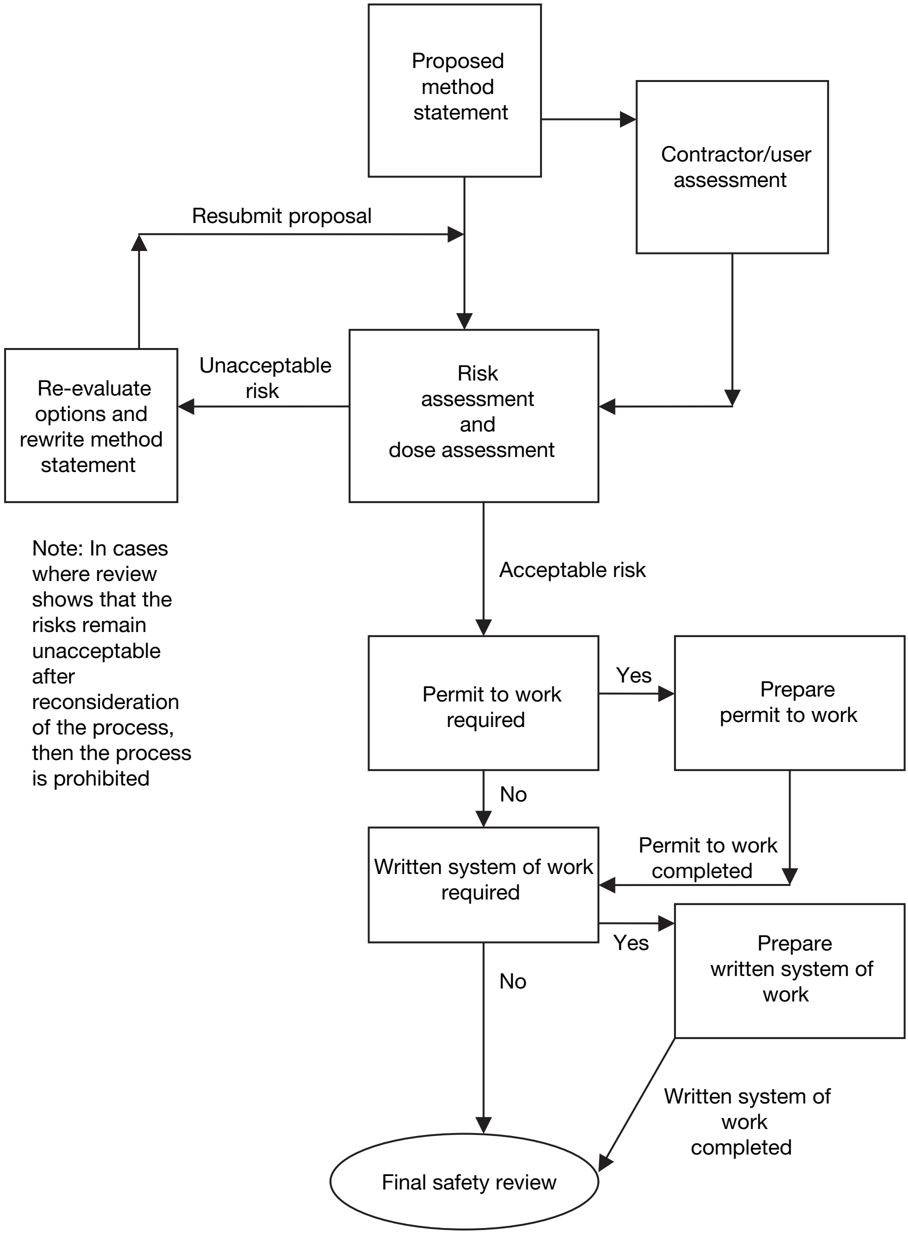


FIG. 2. Process for the preparation of the safe system of work.

Authority (UKAEA). The company became the world's first commercial supplier of the ^{10}B isotope and the first European manufacturer of BF_3 neutron detectors. The capability of producing the ^{10}B isotope led to 20th Century Electronics manufacturing ionization chambers for use in UKAEA nuclear reactors being developed at that time. The company, therefore, became involved at the early stages of the UK nuclear power generation industry. Since then, 20th Century Electronics Ltd or Centronic Limited, as it has been known since 1978, has become a supplier of radiation detectors (e.g. BF_3 counters, boron ion chambers and fission chambers) to both research and commercial nuclear reactors worldwide.

4.1. Historical background and justification for testing

Centronic Limited is a core supplier of detectors used in the neutron flux monitoring and safety systems of many of the nuclear plants and establishments around the world. For this reason, it is important that a suitable testing and validation facility is maintained. Some of the applications for which Centronic Limited manufactures reactor control detectors (e.g. AGR and CANDU reactors) require that the detectors are tested in a high thermal neutron flux (e.g. 10^{8}–10^{9} n.cm^{-2}s^{-1}) as part of their acceptance tests. It is also necessary, on occasion, to carry out proving tests of new detector designs at similar flux levels. These tests provide a measurement of the neutron sensitivity of the detector and may also include a determination of its neutron saturation characteristics. To determine the neutron saturation characteristic, it is necessary to irradiate the device in as high a neutron flux as possible, consistent with obtaining a reasonable extrapolation to the operating saturation characteristic while avoiding excessive activation of the construction materials of the device. Suitable levels of thermal neutron flux have only been obtainable at research reactors. Initially, Centronic Limited routinely used the BEPO and JASON reactors as a source of an appropriate neutron flux with the HERALD and PLUTO reactors being used for some special test requirements. From the late 1960s, routine testing was undertaken at the GLEEP reactor. This testing was transferred to the NESTOR reactor in the late 1980s, following the closure of GLEEP. At the time of the closure of NESTOR in 1994, the only two operational research reactors remaining in the UK were JASON and CONSORT. Centronic Limited chose to develop a test capability at CONSORT, a fortuitous decision with JASON ceasing operation only a few years later and leaving CONSORT as the only civil research reactor in the UK.

4.2. Test geometry and reactor conditions

The tests carried out by Centronic Limited require insertion of the neutron detector into a well-thermalized neutron flux, generally in the range 10^5–10^9 n·cm^{-2}s^{-1}. The level of flux required is largely dependent on the neutron sensitivity of the device being tested and the mode of operation of the detector (i.e. pulse, Campbelling or d.c.).

The most suitable source of thermal neutrons from the CONSORT reactor is obtained by use of a beam tube of 25.4 cm inside diameter in the zero degree face of the reactor which runs parallel to, and approximately 75 cm away from, the central axis of the reactor core. This beam tube is approximately 3.5 m in length, with the final metre penetrating into the thermal column alongside the reactor core. The beam tube diameter contains a step which is bridged by the insertion of a polypropylene guide tube inserted to the full depth of the beam tube after the removal of the shield plugs and before the testing commences. The neutron detector undergoing testing is then manually inserted into position within the polypropylene tube by means of a graduated push rod and separate loading guide tube mounted on a hydraulic lifting table in order to minimize manual handling problems and to ensure that the operation is carried out in the minimum time. The sensitive length of the neutron detector is generally positioned in the middle of the metre long thermal column section, since this is where the neutron flux profile is flattest. The centre of the sensitive length of each detector tested is, therefore, positioned in line with the centre of the reactor core.

In order to calibrate the neutron sensitivity of a device being tested, up to 12 manganese foils are positioned in established standard positions around and along the sensitive length of the detector. Following irradiation for a carefully recorded time, the detector is removed from the beam tube and the activation foils removed for off-line counting using a NaI detector system. The flux is determined from the activity produced in these foils using an internally developed and established computer code. Verification of the accuracy of the foil counting is periodically verified by additional cross comparisons with foils counted at the UK's national standards laboratory, the National Physical Laboratory.

4.3. Development of the safe system of work for the detector testing programme

Once the zero degree beam tube had been identified as providing the required range of thermal neutron fluxes, it was necessary to review further the configuration of the facility and develop the standard procedures that would form the safe system of work with the primary safety objective of ensuring that dose levels to workers would be maintained ALARP.

The use of the hydraulic detector loading trolley ensures that the time spent in loading the detectors is kept to the absolute minimum. This loading arrangement is shown in Fig. 3.

The work includes the operation of the reactor with an open beam tube. The reactor operating rules [5] limit the power to a maximum of 2 kW whenever any beam tube is unblocked, in order to ensure that dose rates in the vicinity of the open beam tube are kept within acceptable limits. The operating rules ensure that operations are at all times within the bounds defined in the reactor operational safety case [6]. These operating rules are agreed with the regulator (NII), any breach of these rules will result in regulatory enforcement procedures which could include legal action. The assessment process described above has been applied to the process in its entirety and this has resulted in the development of procedures based on dose-rate surveys of the open beam tube. The results of these surveys have been included into specific design aspects of the equipment and process in order to ensure that the workers' dose-uptake is ALARP.

The area around the reactor face (zero degree) housing the beam tube is surrounded by additional concrete biological shielding and access is gained via a concrete labyrinth. This area is a permanently designated controlled area and consequently a written system of work is required for contractors working in

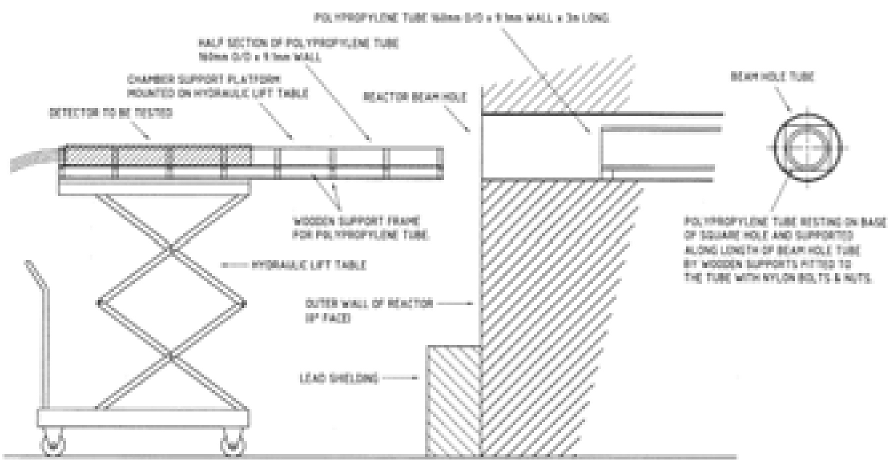


FIG. 3. The detector loading trolley shown with a detector in position prior to loading into the open beam tube.

this area. As the detector testing is an ongoing standard process, the written system of work has been prepared in the form of standard reactor centre procedures and incorporated into the controlled distribution and document review process in line with all standard operating procedures. This ensures that review is automatically carried out at appropriate intervals.

The Centronic Limited personnel working on this project are classified radiation workers under the UK IRR99 regulations. The classified radiation worker status ensures that the dose records for these workers are appropriately managed and that the radiation passbook system which operates within the UK allows the Reactor Centre to ensure that all doses received by the workers during the detector testing process is adequately recorded. In addition, the Radiation Protection Supervisors agree the dosimetry requirements for the workers while working at the Reactor Centre. The contractors are issued with neutron Albedo badges that have been selected for the CONSORT neutron spectrum and these are worn by the contractors at all times in addition to their own TLD dosimetry and Reactor Centre finger TLDs and Electronic Personal Dosimeters (EPDs). The readings from the EPDs are logged in the workers' passbooks at the end of the working day and the readings from the finger TLDs and neutron Albedo badges are reported to the Centronic Approved Dosimetry Service so that the values may be incorporated in personal dose records.

During the process of detector testing, a number of detectors will be inserted in turn into the beam tube. This is done manually using a hydraulic loading trolley as shown in Fig. 3. This equipment has been specifically designed and modified in order to ensure that the time spent at the task is minimized, and that the operatives spend minimal time in the vicinity of the neutron beam. This procedure has been developed on the basis of detailed dose mapping of the beam with the reactor at set powers between 2 W and 2 kW in order to accurately determine the expected dose-uptake, and in order to allow the provision of management control and monitoring of the operation. It is not practical to bring the reactor to shutdown before loading each detector, as this would not only be disproportionately time consuming but also would be incompatible with the use of the calibration foils fixed to each detector for flux assessment purposes. In addition, the reactor does not have an internal shutter for this beam tube and it is not possible to retrospectively install a suitable device. It has been very important under these conditions to ensure that a thorough assessment has been made which fully addresses the following key points and to ensure that the associated procedures are continually reviewed, and that the workers are working in full accordance with those procedures:

(1) Dose-uptake to the contractors involved with loading the detectors and also to Reactor Centre staff removing and reinstating active beam plugs.
(2) Contamination control associated with items that have been exposed to the internal surfaces of the beam tube.
(3) Manual handling of heavy items.
(4) Use of the crane in unloading shielding (statutory UK regulations apply).
(5) Monitoring and coordination of dose records requires the agreement of the RPAs for both organizations.
(6) Transfer of fissile material between organizations (EURATOM requirements associated with fission chamber movements).
(7) The Radioactive Material Road Transport Regulations [7] apply to the movement of the fission chambers in transit between the Reactor Centre and Centronic Limited.

5. CASE STUDY 2: VIEWING OF CERENKOV RADIATION

For many years, it was a normal part of student work at the Reactor Centre to view the Cerenkov radiation emitted from the critical reactor core. This is possible as the reactor tank is covered by Perspex lined tread-plates through which it is possible to view the core assembly at power. Although this experiment was a popular activity with most students and visitors, the activity of directly viewing the phenomenon was discontinued as it was felt that the process could give rise to unnecessarily high dose-rates. This prohibition was implemented without carrying out a detailed risk assessment. Considerable effort was expended in attempting to provide indirect methods of viewing the radiation. These methods included the use of a large mirror suspended above the open reactor tank. The mirror was angled so as to be viewable from an area away from the reactor top. This method has clearly undesirable risks associated with the positioning of items over the open tank; in addition, it did not provide a satisfactory image. Another process that was trialled was the use of an underwater camera which allowed the picture to be viewed on a screen in the reactor control room. Although this was clearly more acceptable from a safety viewpoint, the resulting images were poor in relation to direct viewing and the process was abandoned. Figure 4 shows the Cerenkov radiation emitted from the CONSORT reactor core at a power level of 100 kW.

Many years later, the experiment of viewing the Cerenkov radiation directly was again proposed. By this time, the Reactor Centre had developed the risk assessment and safe system of work procedures described in this paper

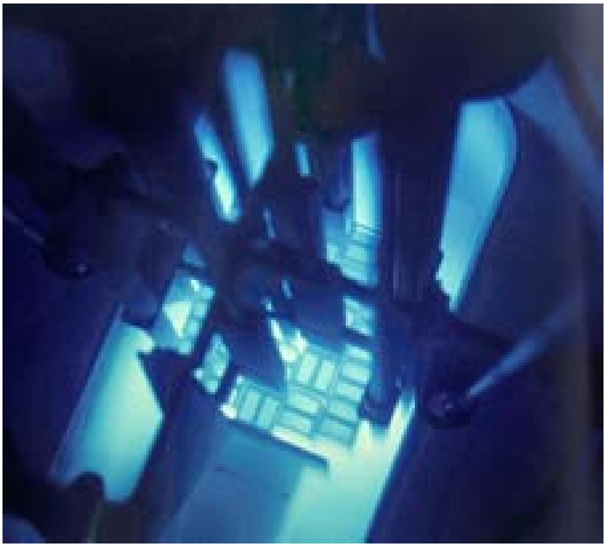


FIG. 4. The Cerenkov radiation emitted from the CONSORT reactor critical core at a reactor power level of 100 kW.

which allowed a complete and structured assessment to be carried out. The resulting procedure and safe system of work were based on the principles of dose assessment, consideration of competing hazards and associated benefits. Tests were carried out in order to establish the minimum reactor power required to give a worthwhile viewing. Consequently, a much reduced reactor power level of 10 kW was established as the optimum, based on the time required to satisfactorily view the effect. The result of this assessment process was that when following best radiological protection principles, each observer would accrue a dose-uptake of around 1 uSv during the viewing. It has been successfully shown that the direct viewing experiment can be carried out successfully in line with the ALARP principle, and that the conventional hazards that may be encountered in the reactor top area actually outweigh the limited dose-uptake received. This can be considered justified when the overall educational benefits of the process are considered. The direct viewing of the Cerenkov radiation at 10kW has now been added to the standard suite of reactor experiments available for students to perform, and this is one of the most popular activities during the experimental days at the reactor.

6. CONCLUSIONS

It has been shown that, with the development of modern risk assessment processes, it is possible for organizations to ensure that adequate levels of control can be applied to practices that may initially seem problematic. The Imperial College Reactor Centre has developed these procedures into a comprehensive system for the production of safe systems of work. This has a number of very obvious benefits, principally in the demonstration that the ALARP process has been followed. This is helpful in gaining the confidence of the regulators that the facility is being managed and operated in a responsible and appropriate manner. In addition, it allows the development of important work programmes that would otherwise not be possible. The example described of Centronic Limited's ongoing neutron detector testing programme shows that the development of these procedures allows for wider improvements to the safety of the industry in providing a test bed for detectors used within the nuclear industry.

REFERENCES

[1] Nuclear Installations Act, 1965, UK legislation.
[2] Radiation (Emergency Preparedness and Public Information) Regulations, 2001, UK legislation.
[3] Ionising Radiation Regulations, 1999, UK legislation.
[4] Radioactive Substances Act, 1993, UK legislation.
[5] ICRC 105/03/01, Operating Rules, S. Franklin, December 2002.
[6] ICRC 190 Series CONSORT Safety Case, (various).
[7] The Radioactive Material Road Transport Regulations 2002 No. 1093, June 2002, UK Legislation.

SAFETY CULTURE ASSESSMENT PROGRAMME
Statistical analysis of a survey conducted at the IEA-R1 Brazilian research reactor

P.S.P. OLIVEIRA, R.N. SAXENA
Instituto de Pesquisas Energéticas e Nucleares (IPEN),
São Paulo, Brazil
Email: patricia@ipen.br

Abstract

The paper describes the statistical analysis of a survey conducted in September 2002 among the employees of the IEA-R1 Brazilian research reactor to evaluate the current status of the safety culture in this organization. A questionnaire was prepared consisting mainly of statements about safety issues. A total of 34 individuals participated in the survey representing the personnel of the Operation and Maintenance Division, the Irradiation Service Division, as well as the technicians in the Radiation Protection Division. The statistical analysis of the survey was developed into three principal steps. In the first step, descriptive techniques were used to estimate the parameters of the sampling distribution of the responses to each question of the questionnaire. In the second step, the aspects of safety culture to be investigated were defined by grouping these questions into issue areas. The final step of the analysis consisted of the calculation of the employees' satisfaction level regarding the safety culture aspects previously mentioned.

1. INTRODUCTION

IEA-R1 is a 5 MW pool-type reactor, cooled and moderated by light water, and it uses graphite and beryllium as reflectors. First criticality was achieved on 16 September 1957 and the reactor has been operating regularly and safely for almost 46 years. The reactor building is located within the premises of IPEN/CNEN-SP, one of the Brazilian institutes for energy and nuclear research, inside the campus of the University of São Paulo.

The safety culture assessment and enhancement programme of the IEA-R1 research reactor was launched by senior management in 2002. A survey was conducted in order to evaluate the main aspects of the safety culture according to its employees' attitudes, opinions and perceptions. The statistical analysis to be presented in this paper shows the quantitative results related to the indicators adopted to identify the main organizational problems and to plan actions, which would result in the improvement of the safety culture.

2. TYPE OF SURVEY CONDUCTED AT THE IEA-R1 RESEARCH REACTOR

The survey method used was a quantitative written questionnaire previously developed at the Australian Nuclear Science and Technology Organisation (ANSTO) and then adapted by the management of the IEA-R1 research reactor to be applied in this organization. The questionnaire was composed of five parts, as follows:

(1) In the introductory paragraph, the objective of the survey was described highlighting the importance of the employees' contribution concerning the results to be obtained. It was also mentioned that the respondents would remain anonymous so that they would express critical views without fear of consequences.

(2) In the second part, there were 43 statements (closed multiple choice questions). A five-point scale allowed the respondents to indicate the extent to which they agree or disagree with each statement. The adopted scale was as follows: strongly disagree, disagree, no opinion, agree and strongly agree. The employees could reveal their opinion in each question by choosing one of these possible alternatives. It is relevant to make some observations in relation to the interpretation given to some of these responses:
   - In the case where the employee had not answered one of the questions, it is possible that he/she had forgotten to answer it or that he/she considered the issue did not pertain to him/her;
   - In the case where the employee had chosen the response "no opinion", it is possible to infer that either he/she had preferred to adopt a neutral position with respect to the issue or he/she considered that it did not pertain to him/her.

(3) In the third part, there were two questions regarding the respondent's personal information in order to characterize the observed sample: working area and working time at the IEA-R1 reactor site.

(4) In the fourth part, there were questions regarding work accidents at the IEA-R1 reactor site.

(5) There was a final open question in which the respondent could make any further comments.

The survey was carried out in September 2002. The questionnaire was answered by 34 people, involving only part of the staff at the IEA-R1 reactor, more specifically, those who work at the Operation and Maintenance Division, the Irradiation Service Division, as well as the technicians of the Radiation

Protection Division. The respondents had a few hours to fill in the questionnaire and to give it back to the nominated reactor management staff member.

3. ANALYSIS METHODOLOGY

3.1. Descriptive statistical analysis of the responses to each question of the questionnaire

Initially, a descriptive statistical analysis of the responses to each question of the questionnaire was carried out so as to classify and to summarize some numerical data. The descriptive analysis consisted of the calculation of the sampling distribution of the responses, including a table of frequencies and a bar graph. The JMP software of SAS Institute [1] was used to perform this task. An example of this analysis is shown in Fig. 1 and Table 1.

3.2. Definition of the safety culture aspects to be assessed in the survey

In the following phase of the analysis, 43 statements presented in the second part of the questionnaire were grouped according to the subject they were related to. For each group, composed of one or more statements, a safety culture aspect was defined. The safety culture aspects, nominated variables 1 to 14 to be analysed in the survey, resulted from this grouping task as shown in Table 2.

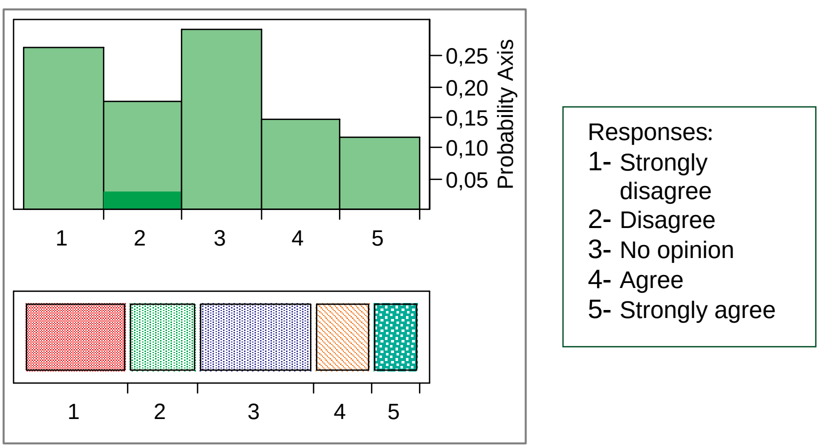


FIG. 1. Sampling distribution of the responses to Question 7: We have lots of breaches of safety procedures around here.

TABLE 1. FREQUENCY DISTRIBUTION OF THE RESPONSES TO QUESTION 7: WE HAVE LOTS OF BREACHES OF SAFETY PROCEDURES AROUND HERE

Response	Count	Probability
1	9	0.265
2	6	0.176
3	10	0.294
4	5	0.147
5	4	0.118
Total	34	1.000

3.3. Descriptive statistical analysis of the safety culture aspects

After defining the safety culture aspects to be analysed in the survey (see Table 2), new bar graphs were plotted to compare the results of each specific group of statements. In Fig. 2, an example of this graph is shown for variable 7: compliance with regulations and procedures.

3.4. Methodology for deriving the satisfaction level of the respondents in relation to the safety culture aspects

In order to assess the respondents' opinions regarding the safety culture aspects described in Section 3.2, a method based on the Likert–Attitude Scoring Technique [2] was adopted. The method consisted of getting an attitude score of each respondent with respect to each variable. This score was calculated summing up the respondent's points from the statements associated with each variable. The respondent's points in each statement were given following the criteria shown in Table 3.

After summing up the points of the statements of each group listed in Table 2, an attitude score of the respondent was obtained in relation to each variable. Besides the attitude score, the respondent's concept regarding the variable was evaluated according to the scale shown in Table 4.

The values x_1, x_2, x_3 and x_4 should be defined according to the number of statements grouped to compose the variable. The scale adopted in this paper is presented in Table 5.

TABLE 2. DEFINITION OF THE SAFETY CULTURE ASPECTS TO BE ANALYSED IN THE OPINION SURVEY CONDUCTED AT THE IEA-R1 RESEARCH REACTOR

Variable/ safety culture aspect	Group of statements of the questionnaire
1. Priority to safety/ importance given to safety related issues	*Question 10*. In general, I think too much attention is paid to safety in our work.—*Positive statement* *Question 13*. People in my work unit are more concerned about safety than people are in other units.—*Positive statement* *Question 35*. Safety improvements are never implemented in my unit because there are not enough resources allocated to this area.—*Negative statement* *Question 36*. The department assigns a high priority to safety as a rule.—*Positive statement*
2. Top management's commitment to safety	*Question 20*. Management gives a consistent message about safety.—*Positive statement* *Question 43*. If it is a matter of safety, the management endeavours to implement it.—*Positive statement*
3. Employee's attitude towards safety	*Question 2*. In general, there's a good safety attitude in my department.—*Positive statement* *Question 16*. In case my immediate superior gives an order, which although in my opinion compromises safety, I try to obey it without questioning.—*Negative statement* *Question 26*. Everybody works safely in my workplace.—*Positive statement*
4. Commitment and responsibility of the employees	*Question 9*. When a safety rule is violated, I'd rather ignore it so as not to compromise my fellow workers.—*Negative statement* *Question 12*. We generally report violations of safety practices.—*Positive statement* *Question 23*. When I find an error in a procedure, I always report it.—*Positive statement*
5. Assessment of the safety level in the organization	*Question 17*. The general level of safety of my fellow workers in this organization is very high.—*Positive statement* *Question 22*. The general level of safety in my work unit or department is very high.—*Positive statement* *Question 32*. My own level of safety performance is very high.—*Positive statement* *Question 41*. There is still a long way to go to attain a good level of safety culture in our department.—*Negative statement*

TABLE 2. DEFINITION OF THE SAFETY CULTURE ASPECTS TO BE ANALYSED IN THE OPINION SURVEY CONDUCTED AT THE IEA-R1 RESEARCH REACTOR (cont.)

Variable/ safety culture aspect	Group of statements of the questionnaire
6. “Absence of safety *versus* production” conflict	*Question 8*. Management often wants short cuts regarding safety for increased efficiency.—*Negative statement* *Question 25*. Management in my workplace is more concerned with production than with people’s safety.—*Negative statement* *Question 29*. Safety works until we are busy, then other things take priority.—*Negative statement* *Question 30*. If I worried about safety all the time, I would not get my job done.—*Negative statement*
7. Compliance with regulations and procedures	*Question 7*. We have lots of breaches of safety procedures around here.—*Negative statement* *Question 14.* As far as safety goes, I find it hard to keep up with all the rules.—*Negative statement* *Question 34.* I try to keep up with the safety rules set in my department.—*Positive statement*
8. Quality and adequacy of documentation and procedures	*Question 3*. Our safety procedures are too strict.—*Negative statement* *Question 27.* The safety rules and procedures in my workplace really work.—*Positive statement* *Question 38.* In case of accidents, the procedures adopted are correct.—*Positive statement* *Question 39.* People who work following the safety procedures will always be safe.—*Positive statement*
9. Openness and communications	*Question 5*. Management would appreciate if we gave suggestions regarding the safest way to do things.—*Positive statement* *Question 15*. There is good communication about safety between those ‘higher up’ and those ‘lower down’ in this organization.—*Positive statement* *Question 18*. When I raise a safety issue, people often do not want to talk about it.—*Negative statement* *Question 21*. Management accepts our safety suggestions.—*Positive statement*

TABLE 2. DEFINITION OF THE SAFETY CULTURE ASPECTS TO BE ANALYSED IN THE OPINION SURVEY CONDUCTED AT THE IEA-R1 RESEARCH REACTOR (cont.)

Variable/ safety culture aspect	Group of statements of the questionnaire
10. Training	*Question 1*. It is easy for a new staff member to learn to do things the safe way.—*Positive statement* *Question 4*. Some of my fellow workers keep making the same dangerous mistake.—*Negative statement* *Question 6*. Some new staff members just don't see the need to do things safely.—*Negative statement* *Question 11*. When it comes to safety, most of the people who work around here do not really know what they should be doing.—*Negative statement* *Question 33*. I am properly skilled to do my tasks completely safely.—*Positive statement*
11. Notions on risk prevention	*Question 31*. Accidents will happen no matter what I do.—*Negative statement* *Question 37*. Not all accidents are preventable; some people are just unlucky.—*Negative statement*
12. Working conditions regarding safety	*Question 24*. Management checks equipment to make sure it is safe to use.—*Positive statement* *Question 40*. All things considered, I am quite satisfied with my job safety.—*Positive statement* *Question 42*. In general, my working conditions enable me to do job my job safely.—*Positive statement*
13. Motivation and job satisfaction	*Question 19*. What happens to this department is really important to me.—*Positive statement*
14. Safety management	*Question 28*. Our safety committee is very effective.—*Positive statement*

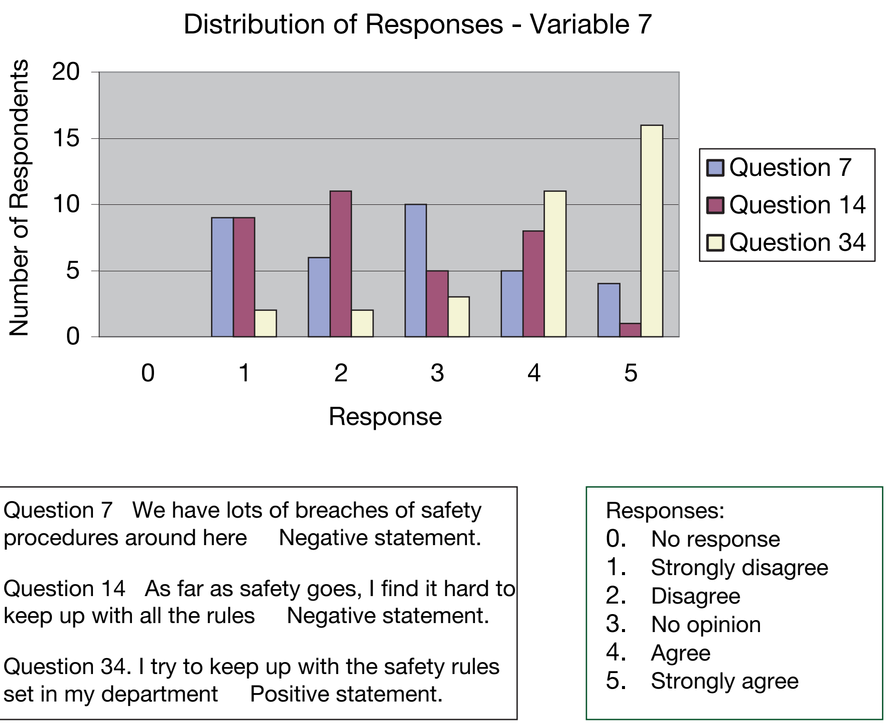


FIG. 2. Distribution of the responses to Questions 7, 14 and 34 (variable 7: compliance with regulations and procedures).

TABLE 3. SCORING CRITERIA FOR THE RESPONSE TO EACH STATEMENT

Response	Criteria 1 Positive statement	Criteria 2 Negative statement
No response	0	0
Strongly disagree	1	5
Disagree	2	4
No opinion	3	3
Agree	4	2
Strongly agree	5	1

TABLE 4. RESPONDENT'S EVALUATION ACCORDING TO THE ATTITUDE SCORE

Attitude score	Respondent's evaluation
Less than x_1 points	Very unsatisfied
From $x_1 + 1$ to x_2 points	Unsatisfied
$x_2 + 1$ points	Indifferent
From $x_2 + 2$ to x_3 points	Satisfied
From $x_3 + 1$ to x_4 points	Very satisfied

To illustrate the steps described in Table 5, consider the example of variable 7: compliance with regulations and procedures. Suppose an employee had answered "agree" to Question 7 (Negative statement), "disagree" to Question 14 (Negative statement) and "agree" to Question 34 (Positive statement). According to Table 3, he/she would have got 2 points for Question 7, 4 points for Question 14 and 4 points for Question 34, resulting in 10 points. Consequently, his/her attitude score in relation to variable 7 is 10, which means, based on Table 5, that he/she would be "satisfied" with respect to the safety culture aspect, known as "compliance with regulations and procedures". The global satisfaction level in relation to each variable was calculated summing up the proportion of employees of the sample who were "very satisfied" and "satisfied" with the safety culture aspect studied. Thus, the final evaluation of the variable could be obtained following the criteria determined in Table 6.

4. RESULTS OF THE SURVEY

4.1. Sample characterization

The sample drawn for the survey carried out at the IEA-R1 reactor consisted of 34 employees whose areas of work are presented in Table 7. Based on the responses to the questions of the third part of the questionnaire, it was possible to verify that 50% of the employees have been working more than 7.5 years for the IEA-R1 research reactor organization. The maximum observed work time was 24 years and the minimum was one year. The mean work time of the employees resulted in 10.1 years.

TABLE 5. SCALE ADOPTED TO EVALUATE THE RESPONDENTS' CONCEPTS ACCORDING TO THE ATTITUDE SCORE

Number of statements	Attitude score				
1 statement	1 point	2 points	3 points	4 points	5 points
2 statements	≤3 points	4 ≤ x ≤ 5 points	6 points	7 ≤ x 8 ≤ points	9 ≤ x ≤ 10 points
3 statements	≤5 points	6 ≤ x ≤ 8 points	9 points	10 ≤ x ≤ 12 points	13 ≤ x ≤ 15 points
4 statements	≤7 points	8 ≤ x ≤ 11 points	12 points	13 ≤ x ≤ 16 points	17 ≤ x ≤ 20 points
5 statements	≤9 points	10 ≤ x ≤ 14 points	15 points	16 ≤ x ≤ 20 points	21 ≤ x ≤ 25 points
Concept	Very unsatisfied	Unsatisfied	Indifferent	Satisfied	Very satisfied

TABLE 6. CRITERIA FOR THE FINAL EVALUATION OF THE SATISFACTION LEVEL OF THE RESPONDENTS IN RELATION TO THE VARIABLES ANALYSED

Satisfaction index in relation to the variable analysed	Evaluation
85% ≤ X	Very good
75% ≤ X < 85%	Good
65% ≤ X < 75%	Satisfactory
50% ≤ X < 65%	Regular
X < 50%	Unsatisfactory

TABLE 7. DISTRIBUTION OF THE RESPONDENTS ACCORDING TO THE WORK AREA

Work area	Number of respondents	%
Customer assistance and administrative areas	3	8.82
Irradiation service	6	17.65
Maintenance division	6	17.65
Operation division	10	29.41
Radiation protection	8	23.53
Other areas	1	2.94
Total	34	100.00

4.2. Final results

The final results of the survey are summarized in Table 8 and plotted in a ‘radar-type’ graph, as shown in Fig. 3.

Concerning the questions related to work accidents at the installation, three employees (8.82% of the respondents) declared they had been involved in at least one event for the period of time they had been working for the IEA-R1 research reactor organization. Among these employees, only one had mentioned that he/she had had a 20-day lay-off as a consequence of the accident.

TABLE 8. SUMMARY OF THE RESULTS OF THE ASSESSMENT OF SAFETY CULTURE ASPECTS AT IEA-R1 RESEARCH REACTOR

Safety culture aspect	Satisfaction level (%)	Global evaluation
1. Priority to safety/importance given to safety related issues	64.70	Regular
2. Top management commitment to safety	61.76	Regular
3. Employee's attitude towards safety	79.41	Good
4. Commitment and responsibility of the employees	76.47	Good
5. Assessment of the safety level in the organization	58.82	Regular
6. "Absence of safety versus production" conflict	67.65	Satisfactory
7. Compliance with regulations and procedures	64.71	Regular
8. Quality and adequacy of documentation and procedures	61.77	Regular
9. Openness and communications	61.77	Regular
10. Training	61.77	Regular
11. Notions on risk prevention	85.30	Very good
12. Working conditions regarding safety	67.64	Satisfactory
13. Motivation and job satisfaction	88.24	Very good
14. Safety management	38.23	Unsatisfactory

5. CONCLUSIONS

(1) The senior managers of the IEA-R1 reactor took the initiative of launching a safety culture assessment and enhancement programme at this organization and were involved in the elaboration of the survey instrument as well.

(2) The experience concerning the application of a survey method for the assessment of the safety culture at the IEA-R1 reactor revealed that the results might be useful to identify organizational problems and to plan actions seeking improvement of safety in the installation.

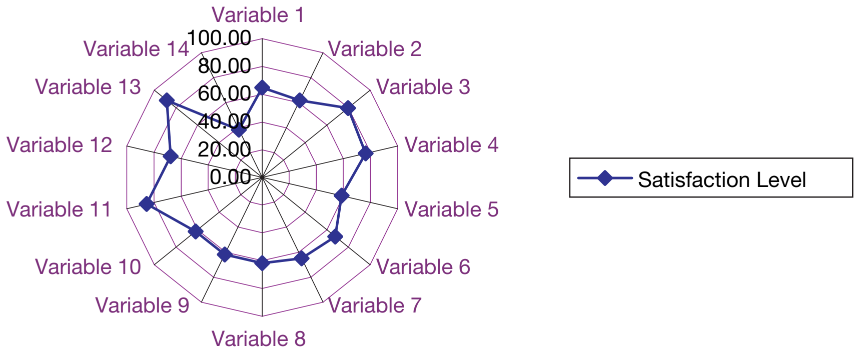


FIG. 3. Summary of the results of the assessment of safety culture aspects at IEA-R1 research reactor.

(3) The sample drawn in the survey was representative of the population of IEA-R1 reactor employees. However, purposely, it did not include all the divisions of the organization. In the future, it is possible to have all the other areas of the IEA-R1 reactor organization participate in order to get more precise statistical results.

(4) Based on the opinions of IEA-R1 employees, the survey identified the safety aspects still not adequate for the accomplishment of the activities in the installation. Following a descending order of satisfaction level, these aspects are:
- Compliance with regulations and procedures (variable 7: 64.71%);
- Priority to safety/importance given to safety related issues (variable 1: 64.70%);
- Training (variable 10: 61.77%);
- Openness and communications (variable 9: 61.77%);
- Quality and adequacy of documentation and procedures (variable 8: 61.77%);
- Top management commitment to safety (variable 2: 61.76%);
- Assessment of the safety level in the organization (variable 5: 58.82%);
- Safety management (variable 14: 38.23%).

(5) On the other hand, the safety culture aspects considered adequate by the same group of employees are:
- Motivation and job satisfaction (variable 13: 88.24%);
- Notions on risk prevention (variable 11: 85.30%);
- Employees' attitude towards safety (variable 3: 79.41%);
- Commitment and responsibility of the employees (variable 4: 76.47%);
- "Absence of safety versus production" conflict (variable 6: 67.65%);

— Working conditions regarding safety (variable 12 67.64%).

(6) It is important to mention that:
- — It would have been better if the safety culture issues to be investigated in the survey had been defined before the selection and development of the questionnaire used to collect data;
- — Some authors recommend that each safety culture aspect should be assessed through at least three statements of the questionnaire. Nevertheless, the ideal format is to have from five to eight statements covering each aspect. Therefore, the aspect corresponding to question 28 named "safety management", needs to be better analysed in the next survey;
- — The aspect named "motivation and job satisfaction" was also assessed based on one statement only (question 19). It is possible to consider a more detailed investigation in a future survey;
- — The aspect named "assessment of the safety level in the organization" may be analysed better in a future survey because statements 17, 22, 42 and 41 were not covering this issue specifically.

(7) After concluding the statistical analysis presented in this paper, a report was elaborated and became available to IEA-R1 reactor managers and employees. Besides that, a lecture was delivered in order to feed back the results to the group involved in the survey.

(8) A follow-up stage was already initiated consisting of a scheduled seminar with a view to increase the understanding of the concept of safety culture among the participants of a focus group. The seminar has been planned by a team of professionals, composed of IEA-R1 reactor senior managers, quality assurance personnel and specialists in probabilistic safety analysis. Other purposes of the seminar programme included to look at potential responses to survey findings; to highlight the need for improvements to be made in the management of safety; and to identify the most appropriate solutions.

(9) Finally, before planning a future survey, it is advisable to conduct interviews to collect some more qualitative information concerning safety aspects at the IEA-R1 reactor organization.

REFERENCES

[1] SAS INSTITUTE Inc., JMP® for Windows and Macintosh, The Statistical Discovery Software™, Version 4 Release 4.0.4.

[2] LIKERT, R., A technique for the measurement of attitudes, Arch. Psych. **140** (1932) 5–53.

BIBLIOGRAPHY

INTERNATIONAL ATOMIC ENERGY AGENCY, Developing Safety Culture in Nuclear Activities, Practical Suggestions to Assist Progress, Safety Reports Series No. 11, IAEA, Vienna (1998).

INTERNATIONAL NUCLEAR SAFETY ADVISORY GROUP, Safety Culture, Safety Series No. 75-INSAG-4, IAEA, Vienna (1991).

BLANK

IAEA-CN-100/58

IMPROVING THE SAFE OPERATION AND NEW OPEN POLICY AT THE VR-1 REACTOR

L. SKLENKA, K. MATEJKA, M. KROPÍK
Faculty of Nuclear Sciences and Physical Engineering,
Department of Nuclear Reactors,
Czech Technical University in Prague,
Prague, Czech Republic
Email: sklenka@troja.fjfi.cvut.cz

Abstract

The VR-1 reactor is operated for training of university students and nuclear power plant personnel, R&D, and information services for non-military nuclear energy use. During the last four years, many improvements in operation have been achieved. Some of them are important from the safety related point of view. In early 2003, a new Web portal with on-line information from the operation of the reactor was launched. The Web portal brings new features in the way to opening information about operation of the VR-1 reactor to the public.

1. INTRODUCTION

The operation of the VR-1 training reactor was started in 1990 by the Department of Nuclear Reactors of the Faculty of Nuclear Sciences and Physical Engineering, Czech Technical University in Prague. The VR-1 reactor is a pool-type light water reactor based on enriched uranium (36%). Its thermal power is up to 5 kW. The moderator of neutrons is light demineralized water, which is also used as a reflector, a biological shielding and a coolant. Heat is removed from the core by natural convection. The pool disposition of the reactor facilitates access to the core, setting and removing of various experimental samples and detectors, and easy and safe handling of fuel assemblies. The control rods have an integral performance. In their structure, the rods are the same. They differ only in function (safety, compensation or control), according to the connection with the control and safety system. The absorber is cadmium. A neutron source is used to start up the reactor. It ensures a sufficient level of the signal at the output of the power measuring channels from the deepest sub-criticalities, thus it guarantees a reliable check of the power during the reactor startup. The reactor is equipped with several experimental devices, e.g. horizontal, radial and tangential channels used to take out a neutron beam.

The VR-1 reactor is utilized particularly for training of university students and nuclear power plant staff. The training on the VR-1 reactor is oriented to the reactor and neutron physics, dosimetry, nuclear safety and control of nuclear installations. Students from technical universities, as well as from universities of natural science come to the reactor for training. Scientific research has to respect reactor parameters and requirements of the so-called clean reactor core (free from a major effect of the fission products). Research at the VR-1 reactor is mainly aimed at the preparation and testing of new educational methodologies, investigation of reactor lattice parameters, reactor dynamics study, research in the field of control equipment, neutron detector calibration, etc.

2. OPERATION DOCUMENTATION AND NEUTRONICS CALCULATIONS REVIEWING

The new Czech Atomic Act issued in 1997 and updated in 2002 requests reviewing all safety and operation documentation within five years from the date of releasing the Atomic Act. The majority of the VR-1 reactor documentation was reviewed and updated or new documentation was created. According to requirements of the Atomic Act, the Czech regulatory body requirements and IAEA recommendations, the following were reviewed:

— Operational limits and conditions;
— Quality assurance (QA) programmes and procedures;
— Inner emergency plan, emergency preparedness and emergency exercises;
— Operation staff qualification and training procedure, operating instructions and procedures;
— Radiation protection and environmental monitoring procedure, waste management procedure.

Two of them (QA programmes and procedures, and emergency preparedness and emergency exercises) were significantly innovated. The new procedure for decommissioning was created. This preliminary version provides aims and methodology for potential decommissioning of the reactor only. The next safety analysis report will be prepared 10 years after the last SAR and after a full upgrade of the control and safety system (2005–2006). At the end of 2003, work on the safety analysis by the PSA method will be finished.

Operation documentation reviewing for the reactor was very useful and brought at the same time new aspects and views on nuclear safety and cultural safety in the operation and utilization of the VR-1 reactor.

In the last four years, neutronics calculation techniques were upgraded, too. Old procedures based on WIMS and DIFER codes were replaced by NJOY and MCNP codes with ENDF/B- 6 and JEFF 2.2 nuclear data libraries. Today, neutronics calculation, evaluation and comparison with experiments are more accurate and exact and can give a new opportunity in the design of the core configuration.

3. UPGRADE OF THE CONTROL AND SAFETY SYSTEM

The present control and safety system (instrumentation and control—I&C) of the VR-1 training reactor was developed in the mid-1980s. The system is digital; it utilizes 8-bit microcomputers with software written in the assembly language. Even if the present control and safety system fully meets the demands that are put on it, its technical design is obsolete to a certain extent at the present time. There are also difficulties with maintenance because of a lack of spare parts. Furthermore, during development and manufacturing, some new internationally imposed demands on quality and the qualification (e.g. recommendations and standards of the IAEA, IEC and IEEE) had not been or could had not have been considered. Therefore, it was decided to upgrade the existing control and safety system with the aim of applying the latest available techniques and technology observing the recommendations and standards mentioned.

The replacement of the control and safety system started in 2001. Because of the frequent utilization of the VR-1 training reactor during the academic terms, it was decided to carry out the upgrade of the control and safety system gradually during holidays so as not to affect the training at the reactor. The completion of the whole control and safety system replacement is assumed in the year 2004 or 2005.

The plan of the replacement consists of four stages. Each stage will be independent and after its completion, the reactor has to remain functional. Moreover, the necessary changes to the upgrade stages just carried out should be minimized with respect to the oncoming stages.

The plan has been scheduled as follows:

(1) Mid-2001: human–machine interface and control room upgrade (done).
(2) Mid-2002: control rod drivers and safety circuits upgrade (done).

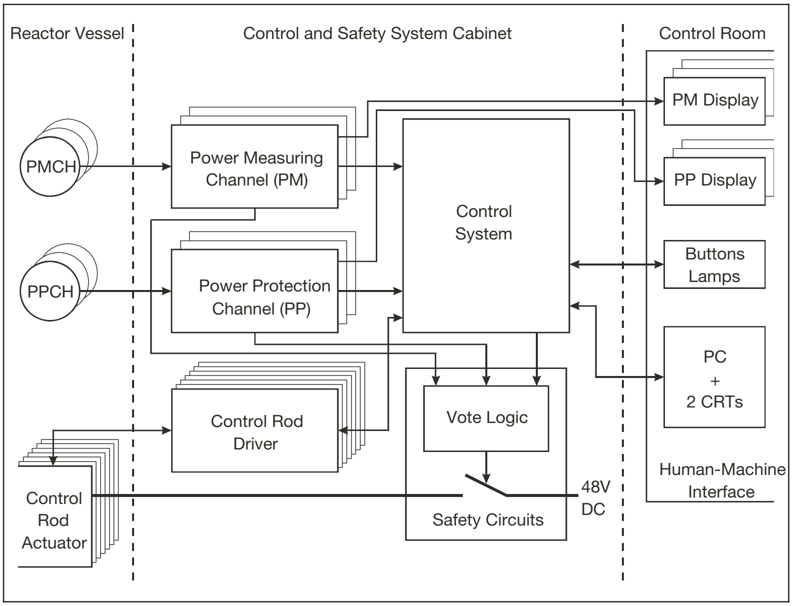


FIG. 1. Block diagram of the upgraded control and safety system.

(3) Mid-2003: control system upgrade (done, in trial operation).
(4) Mid-2004 (2005): power measuring and power protection channels upgrade.

3.1. New human–machine interface

The human–machine interface replacement as the first stage of the control and safety systems upgrade was carried out in mid-2001. The aim of the upgrade was to improve ergonomic and aesthetic properties of the operator's desk and the control room, to enhance the operator's comfort and, thus, to increase the utilization of the reactor and nuclear safety. The new human–machine interface is connected to an Internet server with guaranteed safety of the data transfer.

3.2. New control rod drivers, motors and safety circuits

The control rod drivers, motors and safety circuits were changed during 2002. The original motors of the rods are not available any more. That caused a

FIG. 2. New human–machine interface.

maintenance problem. The rod drivers were also very difficult to maintain because of the unavailability of spare parts and awkward electronic circuits. The original safety circuits suffered ageing of utilized relays. There were problems with the quality of contacts: the contacts corroded, lost conductivity, produced heat because of increased transitional resistance and deteriorated the functionality of the relays. These relays are not produced any more, and the unused spare ones suffered the same problem of ageing and corrosion.

The rod motors were replaced with the new ones that provide the required properties and dimensions. Necessary mechanical changes on the control rod mechanism, induced by the utilization of the new motor, were done by the äkoda Company. High quality connectors were utilized for the connection of the cables to the motors. PLC Simatic S7-200 equipped with a proper power electronic board serves as a motor driver. Appropriate software to control the PLC was developed. The PLCs communicate with the control system via RS485 (ProfiBus) lines. New safety circuits utilize high quality relays with forced contacts to guarantee high reliability of their operation. The safety circuits are installed in a 19 in. rack for an easy installation in new cabinets of the new control and safety system.

This stage of the I&C replacement was seriously complicated by the flood in Prague in August 2002. There was no electricity in the reactor laboratory until October 2002 and all activities of installation had been done utilizing diesel generators.

3.3. New control system

The control system replacement was carried out in mid-2003. The new control system receives data from the power measuring and power protection channels and compares them with safety limits, and it also controls the safety circuits. It calculates the average values of the important variables (power, velocity), and sends data and system status to the human–machine interface. Next, it receives commands and button inputs from the operator's desk and carries them out according to the reactor operation mode. Finally, it serves as an automatic power regulating system.

The information about the control system and the nuclear reactor operational status was enlarged substantially in comparison with the old system. Furthermore, complex tests and control rods diagnostics were added.

The control system is based on the industrial personal computer (PC) of the Nexcom Company mounted in a 19 in. crate with a redundant power supply system. The operating system of the PCs is the Microsoft Windows XP with the real time support RTX of the VentureCom Company. The computer is equipped with 8 RS232 lines for communication with the power measuring and power protection channels, with the RS485 (Profibus) line for communication with the Simatic control rod and I/O PLCs and with the Ethernet line for data transfer to the human–machine interface.

It was necessary to adapt the original power measuring and power protection system for the RS232 communication with the new control system. The communication units, originally developed for the human–machine interface upgrade, were utilized with new PLA firmware and control software. Furthermore, the power measuring and power protection system software was modified in an appropriate way.

Because of the importance of the control system to nuclear safety, a high quality of the delivered hardware and software is requested. Intensive verification and validation is carried out during the manufacturing and after the delivery.

FIG. 3. Control computer.

3.4. New power measuring and power protection system

The power measuring and power protection system, together with the safety circuits and the control rods are the most important items for nuclear safety. In the project study, possible alternatives available on the market were evaluated, with respect to functional, quality and qualification requirements.

The preferred choice is a system based on the SPINLINE 3 from the Schneider Group Company. This system is qualified for protection systems of nuclear devices and is utilized in many nuclear power plants. The SPINLINE 3 technology will also be used for an upgrade of the control and safety system in the Czech nuclear power plant Dukovany.

The SPINLINE 3 is a modular and digital solution, and utilizes the NERVIA network for safe and secure data communication within the safety I&C system. The system meets the international standards governing the design and manufacture of equipment used in nuclear installations

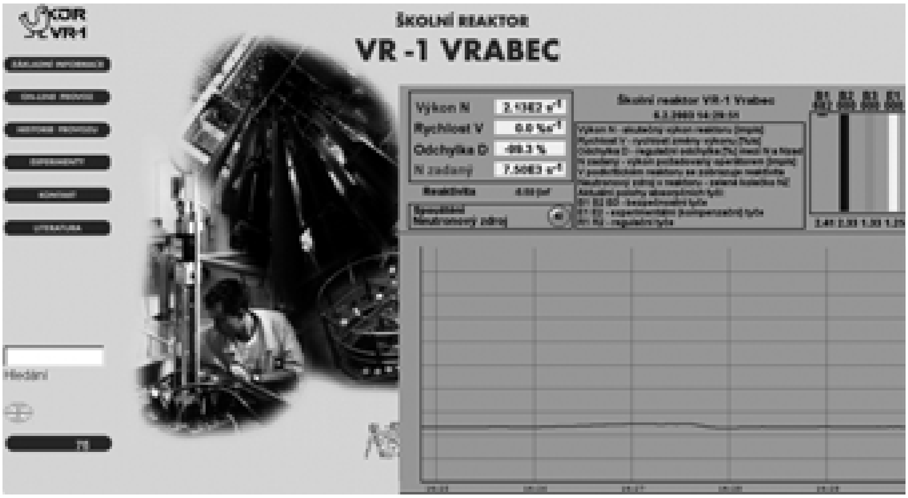


FIG. 4. Actual state of the VR-1 reactor on the Internet.

(e.g. IEC 61226 category A & B functions and associated systems and equipment, IEEE 603 class 1E systems). The software development methodology for the SPINLINE 3 is based on IEC 60880 standard. The SPINLINE 3 is very expensive and the Faculty has to find sponsors to finance the purchase and installation of the SPINLINE 3 system. The final decision will be made according to the funds available.

4. VR-1 REACTOR ON-LINE ON THE INTERNET

State-of-the-art information technologies allow on-line access to information about the operation of any facilities to experts, customers, users, as well as the public. Today, all nuclear installations are in the focus of the public and, therefore, open information is necessary from the operation of any of the nuclear installations. As the Internet is currently the most common, convenient and cheap information channel for the release of non-stop information, the Web portal Vrabec (Sparrow) was installed on the Internet with on-line information from the training reactor VR-1 operation. A bilingual (Czech and English) web site is intended for:

— Students and teachers who take part in the training and teaching process at the reactor as a source of information on reactor experiments which are performed at the reactor;

— The regulatory body in the Czech Republic (State Office for Nuclear Safety) and international organizations (IAEA, EURATOM, etc.) as a tool which can help them to fulfil all aspects of the safe operation in the frame of non-military use and the Non-proliferation Treaty;
— The public as a source of information on construction and safe operation of the small nuclear installation located in the university campus in the Prague municipality.

The Web portal Vrabec consists of four main modules. The first is the Basic information module, which includes basic reactor parameters, a description of the reactor, nuclear fuel, control rods, neutron source, control and safety system, reactor use and principles of nuclear safety, radiation protection, physical protection and emergency preparedness.

The on-line operation module consists of Web cameras and direct on-line output from the operator's desk (human–machine interface) (see Fig. 5). Through three Web cameras located in the reactor hall, control room and reactor vessel, visitors can see the actual situation in the reactor hall and the control room. Visitors can study the actual state of the reactor through on-line output from the operator's desk to the Internet server. The Web interface is the

FIG. 5. Web cameras.

same as at the operator's desk and it can display basic operation parameters, such as reactor power, velocity and deviation, position of control rods, three types of on-line graphs, actual core configuration, reactivity margin calculation, etc.

The previous operation module can display the operation of the reactor in the past. Visitors can select some predefined history and study all basic reactor parameters and graphs as for the on-line operation.

Training and teaching is a significant part of the reactor's operation. The last module, experiments, describes the all important and often used experiments performed at the reactor. Students and teachers can find in this module the methodology for each experiment as a short description or in full in an attached pdf file. In the case of previous the operation, the module includes a history of operation associated with certain experiments. Thus, history is part of the experiment description, too.

The Web portal Vrabec with on-line information from the training reactor VR-1 operation brings new potential in the aspect of educational process at the Department of Nuclear Reactors at the Czech Technical University, and it will be a useful tool for anyone wishing to find up to date information on the VR-1 reactor operation.

REFERENCES

[1] MATEJKA, K., et al., Training Reactor VR-1, Czech Technical University, Prague (2003).

[2] KROPÍK, M., MATEJKA, K., Upgrade of the VR-1 Training Reactor I&C, Enlarged Halden Programme Group Meeting, Lillehammer (2002).

[3] SKLENKA, L., et al., "Training reactor VR-1 on-line on the Internet" (Proc. CTU Workshop Prague, 2003), Czech Technical University, Prague (2003).

[4] KROPÍK, M., MATEJKA, K., SKLENKA, L., CHÁB, V., "New human machine interface for VR-1 training reactor", Conf. on Nuclear Energy in Central Europe, Krajinska Gora, 2002.

[5] KROPÍK, M., MATEJKA, K., SKLENKA, L., Control Room and VR-1 Visualisation System Upgrade, Grant Final Report, FSNPE, Czech Technical University, Prague (2002).

RESEARCH REACTOR UTILIZATION

(Session 2)

Chairpersons

H. BÖCK
Austria

R.N. SAXENA
Brazil

M. RAMANADHAM
India

Co-Chairperson

N. RAMAMOORTHY
IAEA

BLANK

IAEA-CN-100/85

INDIAN EXPERIENCE IN NEUTRON BEAM UTILIZATION
In-house and collaborative programmes

M. RAMANADHAM, S.K. PARANJPE*, S.L. CHAPLOT,
R. MUKHOPADHYAY
Bhabha Atomic Research Centre,
Mumbai, India
Email: ramu@magnum.barc.ernet.in

Abstract

India operates a National Facility for Neutron Beam Research around its 100 MW research reactor, Dhruva. The facility in the reactor hall includes a single-crystal diffractometer, two powder diffractometers, a high-Q diffractometer, a polarization analysis spectrometer, a triple-axis spectrometer, a filter detector spectrometer and a quasi-elastic scattering spectrometer, while two small-angle neutron scattering instruments and a polarized neutron reflectometer are operational in the Guide-Tube Laboratory. In addition, a neutron radiography facility is also available. Our strength lies in the in-house development of all the instruments, including the detectors and the electronics. About 35 research groups from various universities and other institutions are engaged in various collaborative projects using the national facility. Various international collaborations have continued over the last four decades with RCA partners in Asia and with countries elsewhere.

1. INTRODUCTION

Neutron beam research was initiated in India when Apsara, the first research reactor in Asia, attained criticality at the then Atomic Energy Establishment Trombay (AEET), Mumbai, during the late 1950s. The work continued at CIRUS, a 40 MW research reactor during the 1960s, 1970s and 1980s. Presently, a National Facility for Neutron Beam Research (NFNBR) is operational at the Solid State Physics Division (SSPD), Bhabha Atomic Research Centre (BARC), Mumbai. NFNBR has been built around Dhruva, a

* Present address: Department of Nuclear Sciences and Applications, International Atomic Energy Agency, P.O. Box 100, Wagramer Strasse 5, A-1400 Vienna, Austria.

natural uranium fuelled, heavy water moderated and cooled research reactor. The neutron flux is about 1.8×10^{14} neutrons/cm^2/s, when Dhruva is operated at its maximum power of 100 MW.

2. NEUTRON SCATTERING FACILITIES

The layout plan of the neutron beam lines at the Dhruva reactor and the neutron scattering instruments in the reactor hall and guide laboratory are shown in Fig. 1 and a photograph of the panaromic view of the neutron scattering facilities at Dhruva is shown in Fig. 2.

At present, a four-circle single-crystal diffractometer, two powder diffractometers, a high-Q diffractometer, a polarization analysis spectrometer, a triple-axis spectrometer, a filter detector spectrometer, and a quasi-elastic scattering spectrometer are located inside the reactor hall on various beam ports. Two small-angle neutron scattering instruments are operational in the adjacent Guide-Tube Laboratory (GTL). In addition, a polarized neutron reflectometer has recently become operational, and a spin-echo spectrometer is under installation, both inside GTL. Two neutron guide tubes, G1 and G2 (length: 21 m and 35 m; radius of curvature: 1916 m and 3452 m; characteristic wavelength: 3.0 Å and 2.2 Å, respectively) transport neutron beams in to GTL

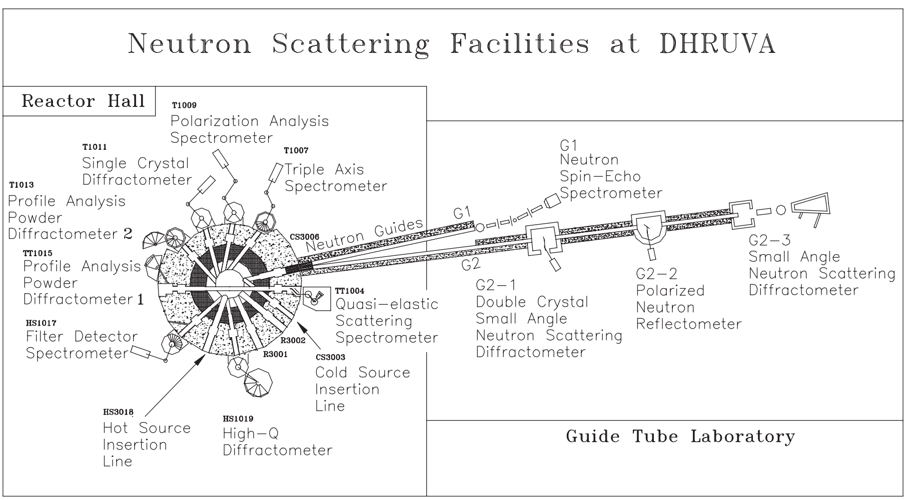


FIG. 1. The layout plan of the neutron scattering facility at Dhruva, BARC, Trombay.

FIG. 2. Panaromic view of the neutron instruments at Dhruva. The instruments seen are, from left, triple axis spectrometer, polarization analysis spectrometer, single-crystal diffractometer, multi-position-sensitive detector (PSD) based powder diffractometer, on the four tangential beamtubes, a single PSD based powder diffractometer at the end of a through-tube and filter detector spectrometer at the extreme right. The hi-Q diffractometer and Quasielastic spectrometer are on the opposite side and not seen in the photo. The guide tube laboratory is on the left of the reactor hall, the shielding of the guide is seen.

from the reactor hall. Average flux at the breaks, provided on the guides to accommodate various instruments, is about 10^7 neutrons/cm^2/s. A neutron interferometer, originally installed at CIRUS, is being upgraded, and plans to build a prompt gamma ray neutron activation analysis facility, in collaboration with the Radio-Chemistry Division (RCD), are currently underway. The Apsara reactor houses the neutron radiography facility, which will be upgraded and installed at CIRUS reactor in the near future. NFNBR develops and fabricates neutron detectors, for both in-house use and supply to other users in the country, using BF_3 and He^3 gases. Liner position-sensitive detectors, using He^3 gas have been developed and extensively used at BARC during the past decade. A two-dimensional position-sensitive detector has also been fabricated and tested recently.

CIRUS has been refurbished recently. With the availability of CIRUS, it is expected that the NBR activities will be boosted. It is proposed to have a time of flight (rotating crystal based) quasielastic spectrometer, one ultra small-angle instrument based on channel-cut double crystal and one beamline for radiography facility and one stress analysis instrument.

All the neutron instruments, guide tubes, detectors, etc., have been designed and developed in-house by neutron beam researchers in SSPD. Fabrication, installation and testing of them, as well as the development of instrument control and data acquisition systems, have been carried out in collaboration with various divisions in BARC. These facilities are continuously upgraded and maintained to facilitate research on materials of scientific interest and technological importance.

3. NATIONAL AND INTERNATIONAL COLLABORATION

Our NBR facilities are made available to the Indian scientific community through bilateral and multilateral collaborations, most of which are sponsored by the Inter-University Consortium for the Deparment of Atomic Energy Facilities (IUC-DAEF). Young scientists are trained by arranging annual workshops and schools on various aspects of neutron scattering in collaboration with the IUC-DAEF. International collaboration involving India in neutron beam research is over four decades old. The first such collaboration was initiated by the India-Phillipines-IAEA Agreement. Subsequently, the experience and expertise, available at BARC, were made accessible to other Asian countries, such as Indonesia, Korea, Bangladesh, etc. Scientists from BARC visited some of these countries in the past to set up neutron beam research facilities, and train local scientists, some of whom subsequently visited BARC and worked at CIRUS/Dhruva with support from IAEA and other agencies.

Neutron beam researchers from BARC have been using advanced neutron scattering facilities in the United Kingdom, France, Germany, Switzerland and Japan to carry out experiments requiring higher resolution, higher neutron intensity and extreme sample environment. Thus, the main aim of neutron beam research in India, all along, has been to develop excellent facilities, use them to carry out research on frontiers of science, share the facilities and expertise with scientists from India and other Asian countries, and also use the advanced facilities available abroad.

4. RESEARCH ACTIVITIES

Some of the important neutron beam research activities at BARC are described below.

Inorganic hydrates were studied initially by single-crystal neutron diffraction technique in order to classify the lone-pair coordination of hydrogen bonded water molecules in these crystal structures [1]. Subsequently, structures of amino acids, small peptides and other bio-molecules were investigated to obtain high precision data on the hydrogen atom positions and hydrogen bond parameters [2, 3]. These data were extensively used in interpreting the X ray protein structures. Recently, phase transitions in the triglycine family of hydrogen bonded ferroelectrics have been interpreted on the basis of the structural studies [4].

Powder neutron diffraction studies, using a variety of sample environments, were carried out on ferrites having disordered magnetic structure, manganites showing CMR behaviour, cobaltates showing coexistence of ferro- and antiferromagnetism [5], titanates and zirconates exhibiting ferro/antiferroelectric properties [6], and alloys of transition metals, rare-earths and actinides showing exotic magnetic properties [7].

A combination of experimental neutron scattering studies at BARC and elsewhere, and theoretical lattice dynamical calculations has resulted in understanding the vibrational and thermodynamic properties of a variety of materials, including many organic and inorganic compounds [8], high T_c superconductor [9], intermetallic hydrides, material having negative thermal expansion [10, 11], number of minerals of geophysical interest [12] etc.

The quasielastic neutron scattering (QENS) technique is used to study the stochastic dynamics in a variety of systems [13]. The results of the recent study of the alkyl chain dynamics in monolayer protected metal nano-clusters are quite interesting. In these substances, monolayers of surfactants are formed spontaneously by self-assembly of the long-chain alkyl thiol molecules on a nano-sized metal core [14]. The QENS method was also used to study the diffusivity of various guest molecules in Zeolite cages [15].

Small-angle neutron scattering (SANS) studies, carried out at BARC, include pore morphology in sintered ZrO_2 - 8 mole% Y_2O_3 seramic, pore surface roughening in a variety of rocks, etc. [16]. One of the major contributions of SANS studies is the exploration of multiple scattering to investigate the very large inhomogeneities, which are otherwise not accessible to low resolution SANS experiments [17].

Micellar formation using a variety of surfactants under various conditions was also investigated using SANS [18]. These studies include those on Gemini surfactants, multi-head group surfactants and others [19]. The role of additives in the formation of micellar structures was also investigated.

Using high-Q diffractometer, structural aspects of liquids (cluster formation in hydrogen-bonded liquids, such as methyl alcohol, t-butyl alcohol, etc.) [20] and amorphous substances were investigated. Some of the amorphous substances studied were semi-conducting chalcogenide glasses (Ge-Se glasses)

[21], phosphate glasses (rare-earth oxide-phosphatre, lead-iron phosphate, etc) and oxide glasses [22].

The neutron reflectometer is useful to study surfaces and multi-layered structures. Recently, the structure and magnetic properties of the ultra-thin multilayer of Fe-Ge have been investigated [23] using both polarized and unpolarized neutrons.

Other than the in-house research, the neutron scattering facility is available to all researchers from different universities and other institutions in India. There are, at present, about 35 ongoing projects sponsored by the Inter-University Consortium for the Department of Atomic Energy Facilities (IUC-DAEF) [24] largely involving small-angle scattering, powder diffraction studies of crystalline and amorphous materials and also quasi-elastic and inelastic scattering. The IUC-DAEF is also in the process of developing a beamline that would have contributions from many university users.

5. PERSPECTIVE

In this paper, we have given a glimpse of our available facilities for neutron beam research and current activities involving national and international collaboration. We are presently working on plans to upgrade many of the spectrometers by the use of neutron focusing optics, multiple one-dimensional position sensitive detectors and two-dimensional PSD, and also to add a couple of new facilities. These developments would enable not only a greater throughput of the experimental data, but also enhance the range of research application; and thereby sustain a high level of research at the frontiers of condensed matter science.

REFERENCES

[1] CHIDAMBARAM, R., SEQUEIRA, A., SIKKA, S.K., J. Chem. Phys. **41** (1964) 3616.

[2] RAMANADHAM, M., SIKKA, S.K., CHIDAMBARAM, R., Pramana **1** 247 (1973); M. RAMANADHAM, M., CHIDAMBARAM, R., Advances in Crystallography (R. SRINIVASAN, Ed.), Oxford & IBH Publishing Co., New Delhi (1978), p. 81.

[3] CHITRA, R., CHOUDHURY, R.R., RAMANADHAM, M., Appl. Phys. **A 74 [S]**, S1576 (2002).

[4] RANJAN-CHOUDHURY, R., CHITRA, R., RAMANADHAM, M., J. Phys., Condens. Matt. **15** 1 (2003).

[5] PARANJPE, S.K., YUSUF, S.M., Neutron News **13** (3), 36 (2002); YUSUF, S.M., SAHANA, M., DÖRR, K., RÖβLER, U.K., MUELLER, K.H., Phys. Rev. **B 66** 064414.1 (2002); SATHE, V.G. et al., J. Phys., Condens. Matt. **8** 3889 (1996).

[6] KRISHNA, P.S.R., et al., Appl. Phys. Lett. **62** 231 (1993); RANJAN, R., et al., J. Phys., Condens. Matt. **11** 2233 (1999).

[7] YUSUF, S.M., et al., Phys. Rev. **B 53** 28 (1996).

[8] CHAPLOT, S. L., Indian J. Pure Appl. Phys. **32** 560 (1994); RAO, K.R., et al., Science **236** (1987) 64; MITTAL, R., CHAPLOT, S.L., KOLESNIKOV, A.I., LOONG, C.K., JAYAKUMAR, O.D., KULSHRESHTHA, S.K., Phys. Rev. **B 66** 174304 (2002).

[9] CHAPLOT, S.L., REICHARDT W., PINTSCHOVIUS, L., PYKA, N., Phys. Rev. **B 52** 7230 (1995).

[10] MITTAL, R., CHAPLOT, S.L., SCHOBER ,H., MARY, T.A., Phys. Rev. Lett. **86** 4692 (2001).

[11] CHAPLOT, S.L., MITTAL, R., Phys. Rev. Lett. **86** 4976 (2001).

[12] CHAPLOT, S.L., et al., Eur. J. Mineral **14** 291 (2002).

[13] MITRA, S., MUKHOPADHYAY, R., Curr. Sci. **5** 653 (2003).

[14] MITRA, S., NAIR, B., GOYAL, P.S., PRADEEP, T., MUKHOPADHYAY, R., J. Phys. Chem. B **106** 3960 (2002); MUKHOPADHYAY, R., MITRA, S., PRADEEP, T., TSUKUSHI, I., IKEDA, S., J. Chem. Phys. **118** 4614 (2003).

[15] SAYEED, A., MITRA, S., ANILKUMAR, A.V., MUKHOPADHYAY, R., YASHONATH, S., CHAPLOT, S.L., J. Phys. Chem. B **107** 527 (2003); MUKHOPADHYAY, R., SAYEED A., MITRA S., ANILKUMAR, A.V., YASHONATH, S., CHAPLOT S.L., Phys. Rev. E **66** 061201 (2002); SAHASRABUDHE, A., MITRA, S., TRIPATHY, A.K., MUKHOPADHYAY, R., GUPTA, N.M., J. Phys. Chem. **B 106** 10923 (2002).

[16] SEN, D., et al., J. Alloys Compounds **340** 236 (2002); SEN, D., et al., J. Mater. Sci. **37** 941 (2002).

[17] MAZUMDER, S., et al., Curr. Sci. **81** 257 (2001).

[18] JOSHI, J.V., ASWAL, V.K., BAHADUR, P., GOYAL, P.S., Curr. Sci. **83** 47 (2002).

[19] ASWAL, V.K., GOYAL, P.S., Chem. Phys. Lett. **357** 491 (2002); ASWAL, V.K., GOYAL, P.S., Chem. Phys. Lett. **364** 44 (2002); SONI, S.S., SASTRY, N.V., ASWAL, V.K., GOYAL, P.S., J. Phys. Chem. **B 106** 2606 (2002); PAL, O.R., GAIKAR, V.G., JOSHI, J.V., GOYAL, P.S., ASWAL, V.K., Langmuir **18** 6764 (2002); SONI, S.S., SASTRY, N.V., PATRA, A.K., GOYAL, P.S., J. Phys. Chem. **B 106** 13069 (2002); ASWAL, V.K., et al., J. Phys. Chem. **B 102** (1998) 2469; Phys. Rev. **E 61** (2000) 2947.

[20] NATH, P.P., SARKAR, S., KRISHNA, P.S.R., JOARDER, R.N., J. Appl. Phys. **A 74** [S], S348 (2002); JOARDER, R.N. et al., Phys. Lett. **253** (1999) 207; JOARDER, R.N., et al., Phys. Lett. **253** 207 (1999).

[21] RAMESH, N., et al., J. Non-Cryst. Solids **240** 221 (1998).

[22] SHIKERKAR, A., et al., J. Non-Cryst. Solids **270** 234 (2000).
[23] SINGH, S., BASU, S., VEDPATHAK, M., KODAMA, R.H., CHITRA, R., GOUD, Y. (communicated).
[24] INTER-UNIVERSITY CONSORTIUM FOR DEPARTMENT OF ATOMIC ENERGY FACILITIES (INDIA) Annual Report, IUC-DAEF (2002).

GEOCHRONOLOGY BY THE ^{40}AR/^{39}AR METHOD AT THE SERNAGEOMIN LABORATORY, SANTIAGO, CHILE

C. PÉREZ DE ARCE, S. MATTHEWS,
Servicio Nacional de Geología y Minería (SERNAGEOMIN),
Santiago

J. KLEIN
Comisión Chilena de Energía Nuclear (CCHEN),
Santiago

Chile

Abstract

An $^{40}Ar/^{39}Ar$ geochronology laboratory has been in operation at the Servicio Nacional de Geología y Minería (SERNAGEOMIN) in Santiago, Chile, since 1998. This laboratory has the capacity to date potassium-bearing geological materials with ages ranging from the age of the solar system (4600 million years), into the historical record (2000 years). The youngest sample analysed to date in this laboratory is a sanidine crystal with an age of 12 000 years. Samples are irradiated at the Comisión Chilena de Energía Nuclear (CCHEN) nuclear reactor, using the Fish Canyon sanidine (FCs) standard (28.03 million years) as a neutron flux monitor. The most important factor in obtaining precise and accurate ages is the measurement of neutron flux (J factor), which depends on the use of high quality geological standards and a relatively homogeneous distribution of J factors across the sample holder discs. A SERNAGEOMIN biotite standard (METRI), with an age of 12.88 ± 0.12 million years, is currently being developed as a complement to the FCs standard. The large (20%) variation of J factors across the sample discs, due to a steep gradient in neutron flux in the irradiation site near the margin of the reactor, will be reduced by the implementation of a motorized sample rotation system, currently under testing. Changes in isotope production ratios, due to the gradual replacement of the enriched fuel elements by less enriched fuel, are monitored using artificial standards such as kalsilite glass and calcium fluoride. Problems of ^{39}Ar loss from fine grained samples by recoil effects are being addressed by the implementation of a sample encapsulation technique. The successful development of the SERNAGEOMIN $^{40}Ar/^{39}Ar$ laboratory has been possible only with the close working relationship which exists with the CCHEN .

1. INTRODUCTION

Potassium has three naturally occurring isotopes: ^{39}K (93.2581%), ^{40}K (0.01167%) and ^{41}K (6.7302%). ^{40}K undergoes a branched decay to ^{40}Ca (89.5%) and ^{40}Ar (10.5%). The ^{40}K to ^{40}Ar decay has a half-life of 1.25 x 10^9 years, and is thus suitable for the study of the age of potassium-bearing geological materials which have the capacity to retain radiogenic argon.

The decay constants for the ^{40}K to ^{40}Ar decay are:

$\lambda_{\beta\_} = 4.962 \times 10^{-10}$ a^{-1} $\qquad$ $\lambda_{e+e'} = 1 \times 10^{-10}$ a^{-1}

Thus, for the combined decay constant, λ: $\lambda = 5.543 \times 10^{-10}$ a^{-1}

The age of the sample (t) is given by: $t = (1/\lambda) \times \ln [1 + (\lambda/\lambda_{e+e'}) \cdot (^{40}Ar^*/^{40}K)]$

The potassium-argon dating method requires the separate analysis of potassium (by chemical methods) and argon isotopes (mass spectrometry). Corrections must be made for the presence of atmospheric argon, which has a $^{40}Ar/^{36}Ar$ ratio of 295.5 ± 0.5. The absolute ^{40}Ar content of the sample is obtained by the addition of a pure ^{38}Ar spike of known volume to the extracted gas. At the Servicio Nacional de Geología y Minería (SERNAGEOMIN), a potassium-argon laboratory has been operating since 1980. Over this 20 year

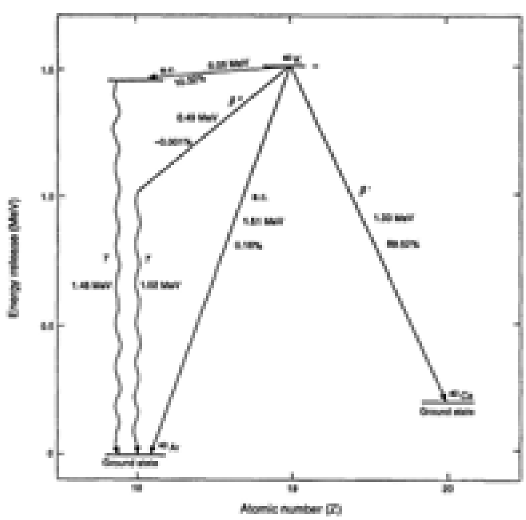


FIG. 1. Decay scheme for ^{40}K, illustrating the dual decay to ^{40}Ca (89.5%) and ^{40}Ar (10.5%). The ^{40}K to ^{40}Ar branch is dominated by electron capture. From Ref. [8], adapted from Refs [1, 2, 4, 6].

period, about 7000 K-Ar analyses have been performed in this laboratory. In 1998, an $^{40}Ar/^{39}Ar$ laboratory was constructed [9], the first of its kind in South America (there is now a second laboratory in Brazil).

2. THE ^{40}AR/^{39}AR DATING METHOD

$^{40}Ar/^{39}Ar$ dating requires the conversion of a proportion of the ^{39}K into ^{39}Ar by neutron irradiation. The measurement of $^{40}Ar/^{39}Ar$ ratios by mass spectrometry permits the calculation of the age of the material, following a series of corrections. Both the K-Ar and $^{40}Ar/^{39}Ar$ methods require the presence of stable mineral phases in the sample which contain potassium and have reasonable Ar retention properties. The most common useful minerals for K-Ar and $^{40}Ar/^{39}Ar$ geochronology are the micas (sericite, muscovite, biotite), amphiboles, feldspars, some sulphate minerals (e.g. alunite, jarosite) and manganese oxides (e.g. cryptomelane, coronadite todorokite). Under ideal conditions, fine grained volcanic groundmass samples may also be analysed. The method requires pure separation and cleaning of minerals, commonly with acids, followed by irradiation in which a proportion of ^{39}K is converted to ^{39}Ar. Samples are placed in orifices in pure aluminium discs (Fig. 2) which are covered with cadmium foil and irradiated within aluminium cans. The cadmium foil serves as a shield against thermal neutrons.

Standards of known age are placed in the orifices with the samples. At the SERNAGEOMIN laboratory the Fish Canyon sanidine (FCs) standard is used, with an age of 28.03 million years [10]. By analysing these standards, it is possible to characterize the distribution of neutron flux (J factor) across the sample holder:

$$J = \exp\lambda t - 1)/(^{40}Ar^{*}/^{39}Ar_{K})$$

where:

t is the age of the standard in years,
$^{40}Ar^{*}$ is the radiogenic ^{40}Ar, and
$^{39}Ar_{K}$ is the ^{39}Ar produced by neutron irradiation from ^{39}K.

Samples are heated in an evacuated sample chamber using a CO_2 laser, either to fusion or in steps of increasing temperature. The liberated gases are cleaned in an extraction line (Fig. 3), and pure noble gases are introduced into a mass spectrometer, which measures the concentrations of the five isotopes of argon (40, 39, 38, 37 and 36).

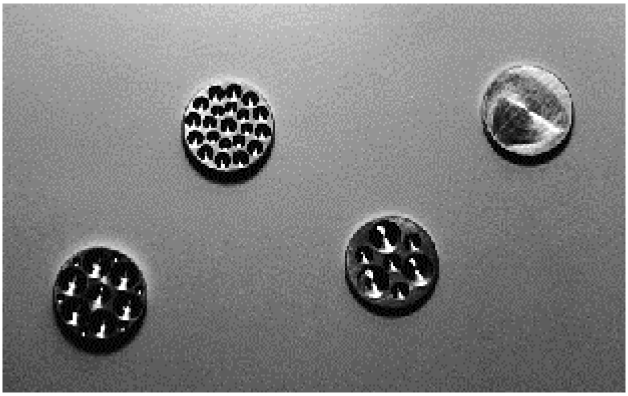


FIG. 2. Aluminium discs used for sample irradiation. These discs are 18 mm in diameter and contain small orifices. Samples and radiation monitors are placed in the orifices, and the discs are covered with an aluminium foil gasket, then a cover (top right). The assembly is then packaged in aluminium foil. The 21-hole disc (top left) is the most frequently used for normal samples.

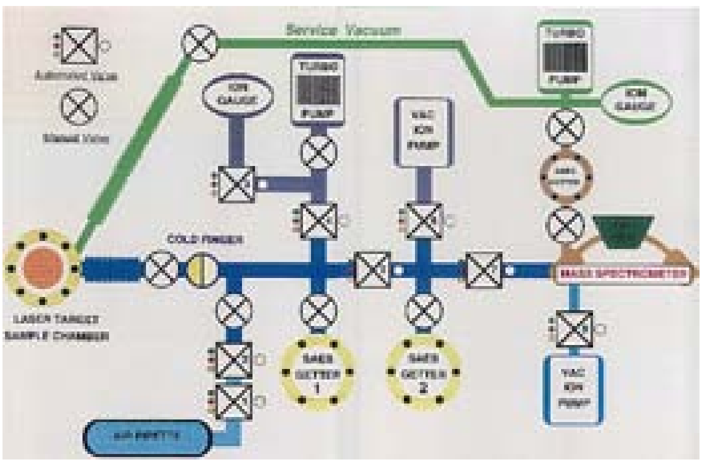


FIG. 3. Diagram of the fully automated ultrahigh vacuum extraction line used for argon release and purification. At left is the sample chamber which has a laser-transparent window in the top cover. Liberated gases are expanded into the extraction line, cleaned using the cold finger (-136^{a} C) and getters, and the pure noble gases are expanded into the mass spectrometer, at right. The service vacuum is used exclusively to maintain the seal on the sample chamber window. While the argon isotopes are being analysed, the extraction line is evacuated using the turbo and ion pumps.

The calculated age of the sample is a function of the $^{40}Ar^{*}/^{39}Ar$ ratio:

$$T = (1/\lambda)\cdot\ln[1 + J\cdot(^{40}Ar^{*}/^{39}Ar_{K})]$$

Corrections are required for atmospheric argon ($^{40}Ar/^{36}Ar = 295.5$) and isotopes produced by the irradiation of K, Ca and Cl. These calculations are carried out using the concentrations of the other Ar isotopes, and the known production ratios for the reactor measured using standard materials.

Samples are analysed either by total fusion (focused laser at high power), or by step-heating experiments, in which the laser is defocused using an integrator lens. Laser power is incrementally increased, and the aliquots of gas which are released in each step are analysed individually. The advantage of this method is that it provides information about the distribution of isotopes in different sites within the mineral structure, and permits the detection of argon loss, excess argon, components with different ages, and so on. These data are presented in terms of step-heating spectra, in which individual steps are plotted as boxes on a plot of cumulative percent ^{39}Ar released against apparent age (assuming that all non-radiogenic argon has an atmospheric composition). The vertical thickness of each box represents the 2-sigma errors in the apparent age. In the spectrum presented in Fig. 4, the 'integrated age' of the sample represents the age calculated after summing all of the gas released; this age is equivalent to a total fusion age. For this sample, a 'plateau age' has also been

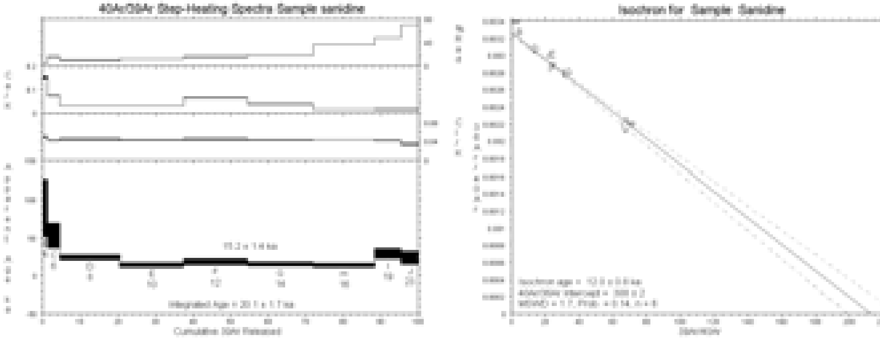


FIG. 4. Step-heating spectrum and inverse isochron plot for a very young sanidine sample ($KAlSi_3O_8$). The U-shaped spectrum indicates the presence of excess argon ($^{40}Ar/^{36}Ar >$ 295.5). In the inverse isochron plot, the data form a mixing line between the radiogenic component (x intercept) and an extraneous, or inherited component (y intercept) with a $^{40}Ar/^{36}Ar$ ratio of 308 ± 2. The correct age for this sample, calculated from the x intercept, is 12 000 ± 1600 years (2 sigma error). Samples younger than 30 000 years can normally only be dated using very potassium-rich phases such as sanidine.

calculated. For a sample to have a valid plateau age, it must show at least three consecutive steps whose ages are concordant within their 2-sigma errors, and the total ^{39}Ar released in the plateau steps must exceed 50% of the ^{39}Ar released during the whole experiment. In this sample, a plateau is present for steps E to H. The radiogenic argon content ($^{40}Ar^*$) for each step is calculated using the ^{36}Ar released:

$$^{40}Ar^* = {}^{40}Ar_t - ({}^{36}Ar \times 295.5)$$

The Ca/K and Cl/K ratios are calculated using the $^{37}Ar/^{39}Ar$ and $^{38}Ar/^{39}Ar$ ratios, respectively.

It is very common for samples to contain a non-radiogenic component with a $^{40}Ar/^{36}Ar$ ratio higher than 295.5. Such samples are said to contain excess ^{40}Ar, and produce U-shaped or steadily decreasing age spectra. Plateau ages for such samples are commonly too old. In order to test for the presence of excess ^{40}Ar, the data are plotted on an inverse isochron plot ($^{36}Ar/^{40}Ar$ against $^{39}Ar/^{40}Ar$). In these plots, the data describe a mixing line between the radiogenic component (x axis intercept) and the inherited, or non-radiogenic component (y axis intercept). The excess argon–corrected age is calculated using the x axis intercept value.

The $^{40}Ar/^{39}Ar$ dating method is entirely comparative (ages obtained are relative to the accepted ages of the available standard materials, most of which have been dated absolutely by high precision K-Ar methods), so that validation of the laboratory using these same standards is rather a circular procedure. Even so, standards are frequently analysed as unknowns in order to maintain a constant check on analytical accuracy.

A SERNAGEOMIN standard mineral, METRI biotite, is being tested both in Chile and in foreign laboratories as a possible international Ar/Ar standard, since many of the available international standards are becoming exhausted. This biotite was obtained from a hornblende-biotite granodiorite intrusion near Puerto Montt, and has been used for some time as an internal standard in the SERNAGEOMIN K-Ar laboratory. Preliminary results of repeated analyses of this standard are presented in Fig. 5. These ages are calculated using the Fish Canyon sanidine standard as a flux monitor, with an assumed age of 28.03 million years. The analyses comprise a series of total fusion ages carried out on one to three biotite grains (each around 0.5 mm across), and one step-heating analysis. The weighted mean of all of the data gives an age of 12.88 ± 0.12 million years. A biotite $^{40}Ar/^{39}Ar$ standard is useful when irradiating white mineral samples such as feldspars, where it is not possible to place a Fish Canyon sanidine in the same orifice as the sample (the sample and standard are difficult to distinguish).

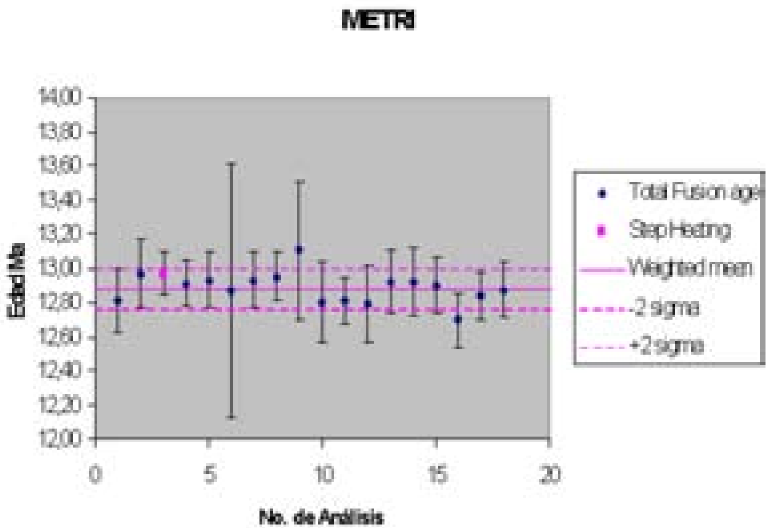


FIG. 5. Preliminary data for the SERNAGEOMIN biotite standard METRI. Most of the data are total fusion analyses, with one step-heating analysis included. The individual 2-sigma error bars depend mainly on the amount of gas released in each analysis. This standard shows excellent reproducibility with a weighted mean age of 12.88 ± 0.12 million years, calibrated to an age of 28.03 million years for the Fish Canyon sanidine standard.

3. USES AND APPLICATIONS OF $^{40}Ar/^{39}Ar$ GEOCHRONOLOGY IN CHILE

The $^{40}Ar/^{39}Ar$ method is of great use in high precision dating of small geological samples, over the full range of geological time (thousands to billions of years). Most samples analysed at the SERNAGEOMIN laboratory are either of internal origin (ongoing SERNAGEOMIN projects) or related to academic research and industry (mining exploration and development). The method is of particular use in Chile for the following types of investigations:

(a) Regional geology; dating of intrusive and volcanic igneous rocks within the areas of regional mapping projects.
(b) Environmental geology; dating of young volcanic rocks for the assessment of eruption cyclicity and volcanic hazards; precise dating of volcanic horizons within young sedimentary sequences for seismic hazard assessment or climate-change studies, etc.

(c) Economic geology; exploration for mineral deposits within ore provinces based on age, fine-scale studies of individual mineralizing events within mineral deposits.

4. IMPORTANCE OF THE CCHEN REACTOR FOR THE SERNAGEOMIN LABORATORY

The SERNAGEOMIN $^{40}Ar/^{39}Ar$ geochronology laboratory has an analytical capacity of about 20 samples per month (step-heating analysis), or 40 samples per month (total fusion analysis). At present, the laboratory is operating at full capacity, and all samples are irradiated at the RECH-1 research reactor operated by the CCHEN at the nominal power of 5 MW. The advantages of irradiating samples at the CCHEN reactor compared with an external reactor are numerous:

(1) Location: The reactor centre is close to the SERNAGEOMIN laboratory so that samples can be transported easily and at low cost, minimizing the problems associated with the transport of radioactive materials and reducing the turnaround time to about one month.
(2) Professional relationship: Constant discussion of analytical problems and methodology is of great assistance when designing irradiation schedules for different sample materials. Monitoring of the radiation environment at the geochronology laboratory is undertaken with the constant advice of CCHEN staff. The development of new sample irradiation techniques is facilitated by this relationship.

5. LIMITATIONS AND DEVELOPMENT OF IRRADIATION METHODOLOGY

5.1. J factor variation and sample rotation

The principal limitation related to the irradiation method is the variation and characterization of neutron flux (J factor) across the sample disc. Typically, a 15–20% variation of J factors is obtained, resulting in a 1–2% variation across individual sample orifices (Figs 6 and 7).

Standards analysis and data processing under these conditions result in a 0.5 to 1% analytical uncertainty due to this variation. In high quality samples, this J factor uncertainty is the dominant cause of total analytical uncertainty.

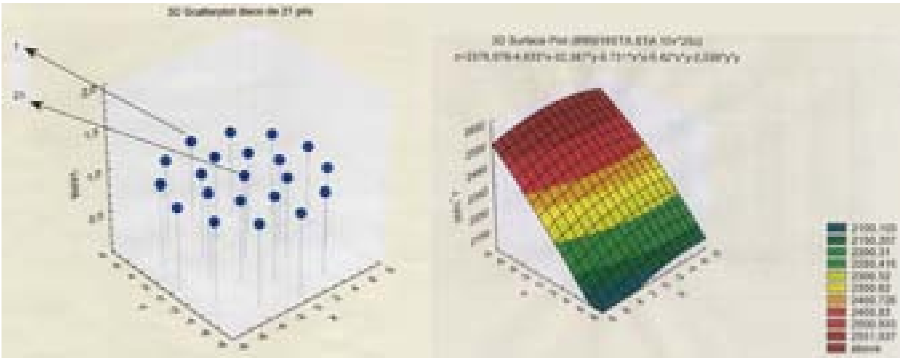


FIG. 6. Typical J factor calculation for a 21-hole irradiation. The holes are numbered clockwise from the top, with 1–12 on the outside, 13–20 in the inner ring, and 21 in the centre. Each sample hole is assigned x-y coordinates (left), and the J factors calculated using Fish Canyon sanidine (FCs) as a flux monitor. Not all sample holes contain FCs, since many samples are feldspars and are difficult to distinguish from the sanidine standard. A smoothed J surface is calculated for the disc, using a 3D quadratic fit to the data (right). For this irradiation, a variation of around 20% is obtained across the width of the disc.

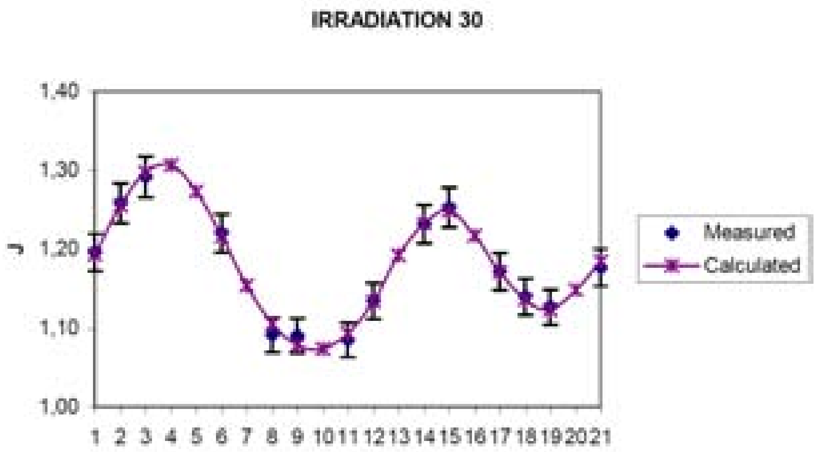


FIG. 7. Comparison of measured values of J (blue diamonds) and smoothed fit to the data (purple asterisks) for a 21-hole sample disc. 2% uncertainties are assumed for the measured values, taking into consideration the possible non-central positioning of the FCs grains within the sample holes. This approach permits accurate calculations of J for holes which do not contain FCs standard grains. The J factor uncertainty used in age calculations is the average percentage difference between measured and calculated values.

Absolute analytical precision is considerably better (0.1–0.2%), thus a reduction in the neutron flux variation across the sample discs would greatly improve the analytical uncertainties for many samples.

Typically, neutron flux variations are reduced by rotating the sample discs during irradiation (Fig. 8). A motorized sample rotation system has recently been completed and is currently being tested by CCHEN staff.

5.2. Isotope production ratios and changes of fuel type

In addition to the production of $^{39}Ar_K$, $^{37}Ar_{Ca}$ and $^{38}Ar_{Cl}$, other Ar isotopes are produced during irradiation. Examples are $^{40}Ar_K$, $^{40}Ar_{Ca}$ and $^{36}Ar_{Cl}$. The production ratios of these isotopes depend upon the energy distribution of neutrons in the irradiation position. Corrections must be made for these neutrogenic isotopes, using specific standards to measure isotope production ratios.

The production ratio $(^{40}Ar/^{39}Ar)_K$ is measured using a synthetic kalsilite (potassium aluminium silicate) glass. This glass is fabricated under high vacuum conditions in order to limit its atmospheric argon content. Since it has a zero age, there is no radiogenic ^{40}Ar present.

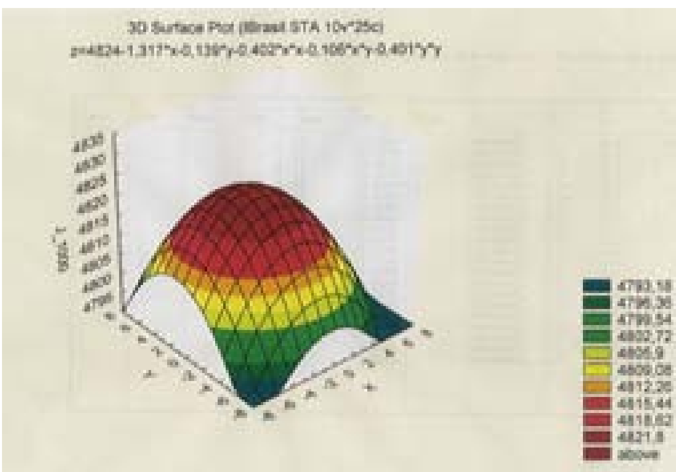


FIG. 8. Smoothed J factor distribution across a 21-hole disc which was rotated during irradiation in the IPEN/CNEN IEA-R1 reactor, Brazil. The hump-shaped distribution is the product of neutron absorption by the aluminium disc, but the difference between the maximum and minimum J factor is of the order of 0.7%. This result may in part reflect the location of the disc close to the centre of the reactor, but illustrates the improvement in uncertainties obtained by rotating the sample disc.

The production ratio $(^{40}Ar/^{37}Ar)_{Ca}$ is measured using synthetic Ca-Si glass or fluorite.

These production ratios need normally be measured only during the early stages of laboratory development, since most reactors are very stable and have fairly constant neutron energy distributions. However, recently it was found that very young samples (20–100 thousand years) began to give negative ages, due to the assignment of too much ^{40}Ar to the irradiation of potassium. New measurements were made of the $(^{40}Ar/^{39}Ar)_K$ production ratio and this was found to be 10 times smaller than the original value obtained four years ago. This discrepancy may be partly the result of a low quality first measurement due to high system blanks, but we attribute a large part of the change in production ratios to the gradual replacement of high enrichment fuel by less enriched fuel in the reactor. Since this process is ongoing at the reactor, frequent measurements of production ratios will need to be made, especially when analysing such young samples. The effect of these changing production ratios on samples over 300 thousand years old is minimal, and most samples analysed in the laboratory have ages ranging from 1 million to 300 million years.

5.3. Recoil effects and sample encapsulation

The energy transferred to a potassium isotope when it absorbs a neutron may cause the resultant argon isotope to move from its crystallographic site into another, less bound, site or to be lost completely from the crystal. This process is known as 'recoil', and can be a serious problem when analysing fine-grained materials. The effective recoil distance is of the order of 1–2 m, so that samples which have crystal sizes in this range can suffer from recoil effects. Two main effects are observed:

(a) ^{39}Ar loss by recoil. In this process, a significant proportion of the ^{39}Ar produced by irradiation is completely lost from the crystal and the sample presents a spectrum which climbs rapidly towards an age which is much too high [3, 5, 7, 11].

(b) ^{39}Ar recoil redistribution. This process commonly affects mica crystals which have interlayer alteration phases. Due to the sheet-like nature of mica minerals, it is common for K in interlayer sites to be replaced by other elements such as Fe or Mg. Thus, thin (2–3 m) layers of vermiculite or other clays may form within the biotite crystal. When irradiated, a proportion of the ^{39}Ar produced recoils into these clay sites. During the step-heating process, the clays tend to degass first, producing apparent ages which are too young. At higher temperatures, biotite layers with a deficit of ^{39}Ar are degassed, producing apparent ages which are too high. The net result is a staircase-like spectrum, which may terminate in a false plateau which is too old (Fig. 9).

The problems of recoil redistribution have no easy solution. The most effective way to avoid these effects is to reject samples which have fine-grained intergrowths or evidence for interlayer alteration, using optical microscopy and X ray crystallography. However, it is frequently very useful to analyse fine-grained materials such as illite or glauconite, which can provide important geochronological information on alteration processes. In such cases, it has been demonstrated that encapsulation of samples in silica glass tubes under high vacuum, prior to their irradiation, can effectively trap the lost gas. During sample analysis, the tube is punctured using a focused laser and the gas it contains is analysed. This gas commonly consists of almost pure ^{39}Ar, with an apparent zero age. The addition of this component to the Ar obtained during step-heating of the sample has been shown to provide reasonable total gas ages for fine-grained samples.

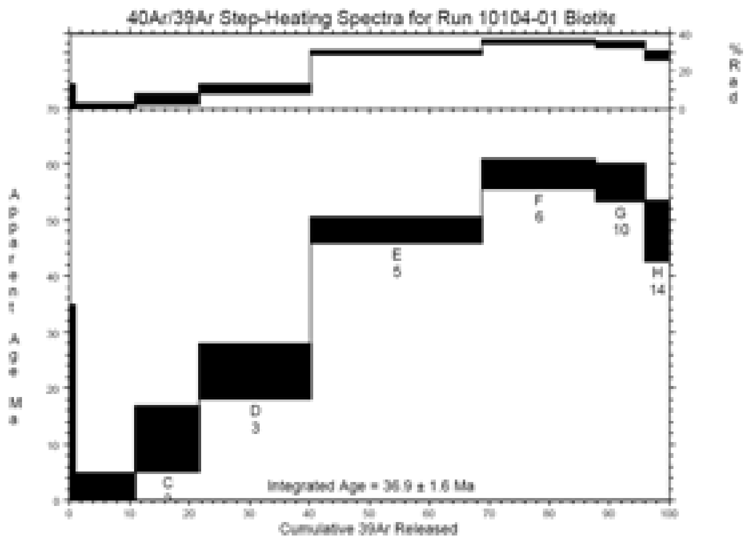


FIG. 9. $^{40}Ar/^{39}Ar$ step-heating spectrum for a biotite sample with interlayer vermiculite alteration. The staircase-like spectrum is typical for such samples and reflects the recoil of ^{39}Ar from the biotite layers into the vermiculite interlayers. The integrated age obtained for such samples commonly, although not always, approximates their true age.

In the SERNAGEOMIN laboratory, recoil effects are frequently encountered in slightly altered or fine-grained materials (Fig. 10). A future development of encapsulation techniques for such samples will require not only the study of evacuation methods, but also the redesign of sample holders for irradiation and analysis.

6. CONCLUSIONS

The single most important factor in the precise and accurate measurement of the ages of geological materials by the $^{40}Ar/^{39}Ar$ method, is the correct measurement of J factors. This requires well dated, homogeneous geological standards and consistent irradiation conditions. Additionally, the isotope production ratios in the reactor must be well characterized, particularly when analysing very young samples. Since the inauguration of the SERNAGEOMIN $^{40}Ar/^{39}Ar$ geochronology laboratory, the close working relationship with the CCHEN has proved vital in the development of a suitable irradiation methodology, the standardization of sample processing throughout the entire analytical chain and the implementation of safety norms for the manipulation of (minimally) radioactive samples. Present and future development and improvement of the method, such as the implementation of sample rotation during irradiation (currently under testing) and possible changes in sample holder design for encapsulated samples, are dependent upon this relationship.

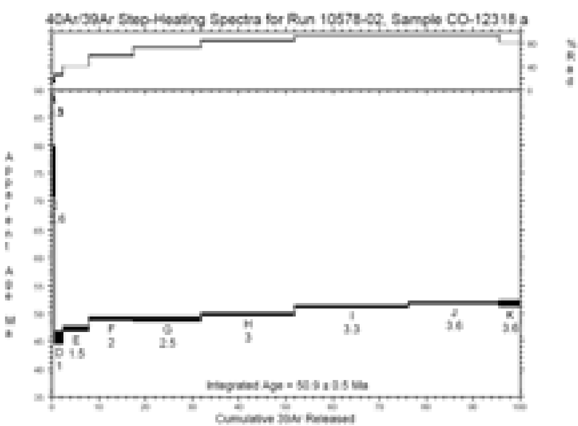


FIG. 10. Sample of fine-grained alunite-quartz mixture showing both ^{39}Ar loss (steps A to C) and redistribution (steps D to K) by recoil effects. The true age of this sample is around 49 Ma.

REFERENCES

[1] BECKINSALE, R.D., GALE, N.H., A reappraisal of the decay constants and branching ratio of ^{40}K, Earth and Planet. Sci. Lett. **6** (1969) 289–294.

[2] DALRYMPLE, G.B., LANPHERE, M.A., Potassium-argon Dating, Freeman, San Fransisco (1969).

[3] DONG, H., HALL, C.M., PEACOR, D.R., HALLIDAY, A.N., Mechanisms of argon retention in clays revealed by laser ^{40}Ar-^{39}Ar dating, Science **267** (1995) 355–359.

[4] FAURE, G., Principals of Isotope Geology, Wiley, New York (1977).

[5] FOLAND, K.A., HUBACHER, F.A., AREHART, G.B., ^{40}Ar/^{39}Ar dating of very fine-grained samples: An encapsulated–vial procedure to overcome the problem of ^{39}Ar recoil loss, Chem. Geol. **102** (1992) 269–276.

[6] GARNER, E.L., MURPHY, T.J., GRAMLICH, J.W., PAULSEN, P.J., BARNES, I.L., Absolute isotopic abundance ratios and the atomic weight of a reference sample of potassium, Journal of Research, National Bureau of Standards **79A** (1975) 713–725.

[7] HESS, J.C., LIPPOLT, H.J., Kinetics of Ar isotopes during neutron irradiation: 39Ar loss from minerals as a source of error in ^{40}Ar/^{39}Ar dating, Chem. Geol. **59** (1986) 223–236.

[8] MCDOUGALL, I., HARRISON, T.M., Geochronology and thermochronology by the ^{40}Ar/^{39}Ar method, 2nd ed., Oxford University Press, Oxford (1999).

[9] PÉREZ DE ARCE, C., BECKER, T., ROESCHMANN, C., El nuevo sistema de datación ^{40}Ar/^{39}Ar equipado con láser de CO_2 en el SERNAGEOMIN, IX Congreso Geológico Chileno, Puerto Varas, 2000, Actas vol. 1. (2000), p. 675.

[10] RENNE, P., et al., Intercalibration of astronomical and radioisotopic time, Geology **22** (1994) 783–786.

[11] SMITH, P.E., EVENSEN, N.M., YORK, D., First successful ^{40}Ar-^{39}Ar dating of glauconies: Argon recoil in single grains of cryptocrystalline material, Geology **21** (1993) 41–44.

FiR 1 REACTOR IN SERVICE FOR BORON NEUTRON CAPTURE THERAPY (BNCT) AND ISOTOPE PRODUCTION

I. AUTERINEN, S.E.J. SALMENHAARA
Technical Research Centre of Finland (VTT),
Otaniemi, Espoo, Finland

Abstract

The FiR 1 reactor, a 250 kW Triga reactor, has been in operation since 1962. The main purpose to run the reactor is now the Boron Neutron Capture Therapy (BNCT). Although BNCT dominates the current utilization of the reactor, it also has an important national role in providing local enterprises and research institutions in the fields of industrial measurements, pharmaceuticals, electronics, etc. with isotope production and activation analysis services. The whole reactor building has been renovated, creating a dedicated clinical BNCT facility at the reactor. Close to 30 patients have been treated since May 1999, when the licence for patient treatment was granted to the responsible BNCT treatment organization. The treatment organization has a close connection to the Helsinki University Central Hospital.

1. INTRODUCTION

A research project to carry out clinical application of boron neutron capture therapy (BNCT) was established in Finland in the early 1990s. It was motivated both by the need to create new uses for the Finnish research reactor FiR 1 and by ideas to start research and production of new boron carriers for BNCT in Finland. As the primary requirement, the suitability of the FiR 1 nuclear reactor, located at and operated by VTT (Technical Research Centre of Finland), was evaluated. The basic design performed by VTT showed that an epithermal neutron beam suitable for BNCT of glioma could be constructed, even with world top performance characteristics [1] despite its low 250 kW power. The other epithermal neutron beams for BNCT had been created on rather high power research reactors: the 3 MW BMRR at BNL (United States of America), the 5 MW MITR-II at MIT (USA) and the 45 MW HFR at JRC Petten (Netherlands). In Japan, multimode BNCT beams, which include also epithermal modes, have been created at the KURRI (5 MW) and JRR-4 (3.5 MW) research reactors.

Soon also medical, medical physics and chemistry disciplines joined the project [2–4]. Now the project involves scientists from different departments of University of Helsinki (HU), Helsinki University Central Hospital (HUCH), Technical Research Centre of Finland (VTT), Finnish Radiation and Nuclear Safety Authority (STUK) and of the University of Turku and other Finnish universities. After gaining enough support both from the medical community, as well as from private and state financing sources, a decision was made in 1994 to construct a BNCT facility at the FiR 1. The aim of this project has been to start BNCT treatments of malignant brain tumours in Finland.

The reactor is located within the Helsinki metropolitan area, where there areabout one million inhabitants, at Otaniemi, Espoo, about 6 km from the largest hospital of Finland, the Helsinki University Central Hospital. The FiR 1 reactor, a 250 kW Triga Mark II research reactor, was taken in operation in 1962. It functioned as a training and research reactor for neutron activation analysis, isotope production, and neutron physics until the mid 1990s. In 1996, an epithermal neutron beam was constructed based on a new neutron moderator material Fluental™ developed at VTT [1, 5]. After successful demonstration of a high purity epithermal beam, the patient irradiation room was constructed by cutting partly into the original concrete shielding of the reactor (Fig. 1). The Fluental™ moderator was shortened to create, at that time, the highest intensity and best purity epithermal neutron beam for BNCT. The whole reactor building was renovated, including construction of irradiation simulation and monitoring rooms, and a laboratory for boron analysis, creating a dedicated clinical BNCT facility at the reactor site [3].

The first patient was treated with BNCT at FiR 1 in May 1999. Patients are treated in collaboration with the Helsinki University Central Hospital, VTT and the Boneca Corporation. The Finnish BNCT multispeciality team consists of radiation therapists and clinical oncologists, neurologists, neurosurgeons, radiologists, pathologists, radiation physicists, chemists, pharmacists, nurses and the nuclear reactor facility personnel. The BNCT facility has been licensed for clinical use and is being surveyed by the Radiation and Nuclear Safety Authority (STUK). The FiR 1 neutron beam is well characterized and particularly well suited for BNCT because of its low hydrogen-recoil and incident gamma doses, and its high intensity and penetrating neutron spectrum characteristics [6–9].

The main purpose of the existence of the reactor is now BNCT. The BNCT work dominates the current utilization of the reactor: three or four days per week are reserved for BNCT purposes and the rest for other purposes, such as isotope production and neutron activation analysis.

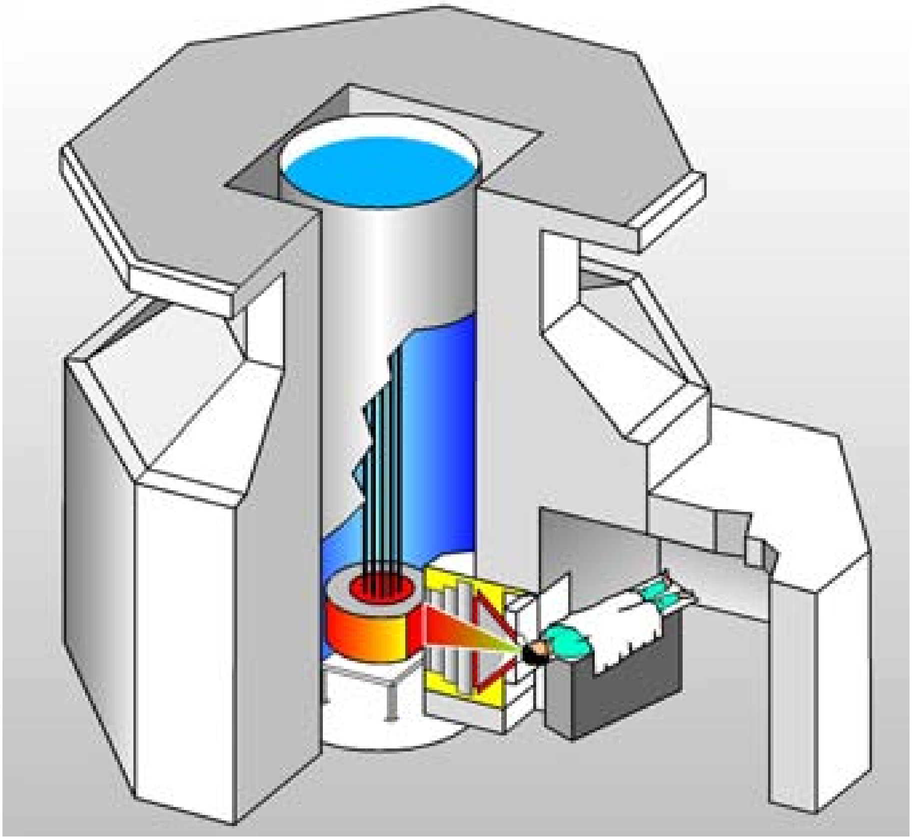


FIG. 1. BNCT facility at the FiR 1 reactor.

2. FIR 1 RESEARCH REACTOR

The Finnish Research Reactor 1 (FiR 1) operated by the VTT is a 250 kW Triga II open tank reactor with a graphite reflector and a core loading of 15 kg U containing 3 kg ^{235}U (20% enrichment) in the special Triga uranium-zirconium hydride fuel (8–12 w% U, 91% Zr, 1% H). The advantages of the Triga design for BNCT include a large flux-per-Watt feature and inherent safety of the Triga fuel. Due to the strong and fast negative temperature coefficient of the reactivity of the Triga fuel and easy operation of this type of a relatively low power reactor, FiR 1 is a safe neutron source for a clinical BNCT facility. The reactor has a good safety and availability record over the last 41 years.

The old training and research reactor facility has undergone major changes in becoming a BNCT clinic. First of all, an epithermal beam facility has been constructed in the radiation shield of the reactor and around that, an irradiation room with a patient positioning system. Rooms for BNCT facility control, patient preparation, boron analysis and dosimetry work have been created. Offices and meeting rooms have been rebuilt. A new entrance for BNCT personnel and patients has been opened for easy access to the BNCT facility. The top of the reactor tank was separated from the reactor hall in order to confine contamination in case of a leakage from irradiation samples or fuel elements. The ventilation of the building, emergency power supply system, heat exchangers and the secondary cooling circuit of the reactor, including cooling towers, were completely redesigned and rebuilt. The reactor core loading and control rod operations were optimized for maximum epithermal beam output.

3. THE BNCT FACILITY

The clinical facilities for BNCT are located on the ground floor of the reactor building with easy access from the driveway through the new BNCT facility entrance. There are the patient preparation and treatment simulation room, the irradiation room, the BNCT monitoring room, the boron analysis laboratory and room for physical dosimetry equipment and laboratory work.

Patient positioning on the treatment coach relative to the neutron beam aperture is performed in the treatment simulation room using a beam aperture simulator. Then the coach with the patient is rolled into the irradiation room into the same position relative to the beam aperture. A crosshair laser system provides an identical coordinate system for patient positioning both in the simulation room and in the irradiation room.

3.1. The epithermal neutron beam

The epithermal neutron field at FiR 1 is produced by replacing the original thermal column graphite in the thermal column cavity starting from the very bottom with the patented Fluental™ neutron moderator (Fig. 2) [10–12]. The thermal neutron load from the graphite reflector of the reactor core is removed with a boral plate. By placing the wide moderator closer to the reactor core, more neutrons are caught into it. These neutrons have a wide energy spectrum that is then compacted to the epithermal range due to collisions and

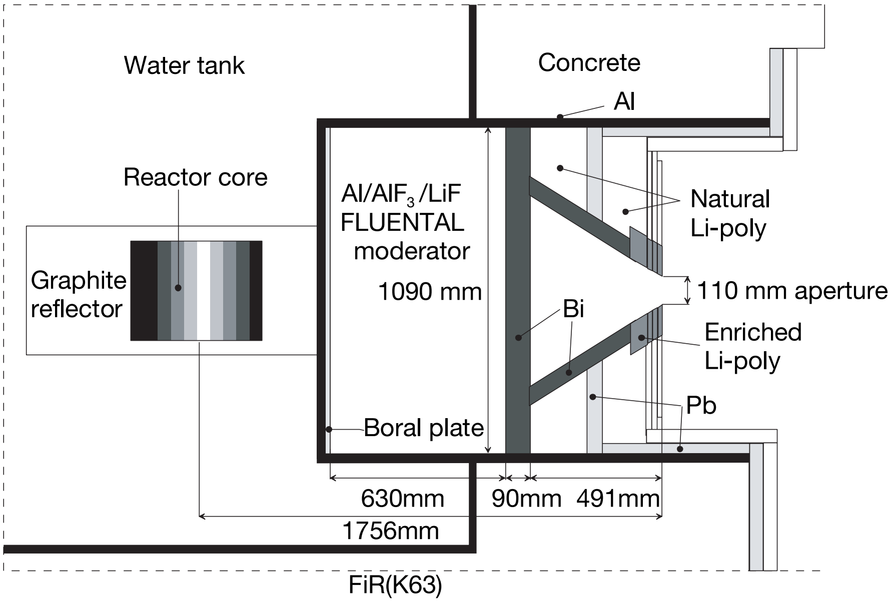


FIG. 2. The epithermal neutron irradiation facility at FiR 1 in its current configuration with a 63 cm thick Fluental moderator. For patient treatments, normally 14 cm and 11 cm diameter beams are used.

capture reactions in the moderator. The bismuth shield passes through neutrons but attenuates efficiently the gammas originating from the reactor core and neutron activated structures.

A conical beam collimator enables the use of the epithermal field for human irradiations. As a compromise between flexible patient positioning and favourable field distribution in the target, a cone with a 60 full angle was found to be optimal [13]. The diameter of the beam exit aperture can be selected from 20 cm down to 8 cm in steps of 3 cm by the number of aperture collars in place.

The neutron fluence intensity decreases rapidly with the distance from the beam port. Therefore, the patient head is required to be kept in contact with the beam port, causing an additional constraint to patient positioning. Several cubic meters of the concrete biological shield had to be cut away in order to allow positioning of the patient close enough to the reactor core in proper bodily orientation. A limiting factor in the removal was the stability of the original biological shield. A structural strength analysis performed by Olli Majamki (IVO Power Engineering Ltd) showed that the thermal column could be widened only so far that a suspending arch is still formed over the irradiation position. Therefore, the cutting was limited to halfway of the side faces.

3.2. The irradiation room

A heavy concrete shielded therapy room was built around the irradiation position (Fig. 1) [14]. Criteria for the construction were easy and clean assembly at the reactor, and the possibility of removing part of the ceiling for maintenance work at the beam, for example, when changing moderator thickness or bismuth collimator. No load was allowed to put on the biological shield of the reactor so the large beam lengths were required. Also optimized shielding for both gamma and neutrons was looked for. Therefore heavy concrete was selected as the basic material and by casting the concrete into steel tubes, good structural strength was combined with high density (4.35 g/cm^3). The achieved high density ensured that with the 80 cm wall and roof thickness, the dose rate levels outside are well within the design objectives and actually lower than outside the original biological shield. The neutron field in the irradiation room is depressed by a factor of 200 by lining the inner walls and the ceiling with in house made 30 mm thick lithium plastic elements. The floor is made of steel plate box elements filled with the same lithium plastic mixture. Lithium was selected as the neutron absorbing isotope throughout the beam collimator and irradiation room to enable background free boron prompt gamma monitoring in the future. The complications created by the tritium production from the lithium have been manageable and no tritium contamination of personnel has been detected. The irradiation room has both incoming and outgoing ventilation ducts. The ventilation can be balanced so that the room has either over- or under-pressure compared with the reactor hall.

3.3. Easy operation for patient irradiations

A BNCT irradiation is started by manual reactor startup and terminated by manual scram. During the irradiation, the reactor is operated at a full 250 kW thermal power. The reactor power is maintained at the specified level by the reactor automation, but the initial starting procedure has to be done manually by the reactor operator. The power is raised on request by the BNCT facility operator. The length of the irradiation is controlled based on the beam monitor units given by the beam monitor system. When the specified amount of beam monitor units is reached, a manual reactor scram is initiated by the BNCT operator.

A rather rapid operation is achieved without beam shutter. Startup after closing the irradiation room door takes 3 min. The patient can be removed at the latest 5 min after reactor scram without uncomfortable doses to personnel.

The medical team, together with the reactor personnel, have been trained for incidents and emergency situations with the reactor. Emphasis has been on the communication between the reactor staff and the BNCT facility. If the situation allows, the shift supervisor of the reactor will consult the persons responsible for patient irradiation, the radiation oncologist and the medical physicist, before making decisions about shutting down the reactor or evacuation of the reactor building in case of emergency.

4. LICENSING

In May 1999, STUK gave to the BNCT facility at FiR 1 the licence to perform BNCT radiotherapy, complying with experimental protocols accepted by the ethical committees. The licence also required an inspection and approval by the municipal health care authorities, as well as an approval by the regional governmental medical authority. The facility met all the requirements as designed without any needs for improvements.

The operating licence of the reactor was renewed at the beginning of 2000. The reactor is now explicitly licensed to be used also as part of a BNCT treatment facility. The reactor personnel has now the right to judge the benefits of a completed patient irradiation by, for example, continuing the running of the reactor even in an emergency situation such as fuel element leakage against risks in radiation and reactor safety.

The operating licence is held by the VTT, an independent government research organization under the Ministry for Trade and Industry. The management organization of the reactor can be seen in Fig. 3. A special management position, the BNCT Manager, was set up for managing the BNCT facility.

The radio therapy licence is held by Boneca Corporation. Boneca Corporation is a firm owned by the Clinical Research Institute at Helsinki University Central Hospital, VTT and Sitra, the Finnish National Fund for Research and Development. The management organization for the radiotherapy licence is shown also in Fig. 3.

As can be seen, the two organizations overlap and are partly included in each other's licensing documents. The VTT is responsible for the maintenance, operation and safety of the reactor and for the radiation safety of the BNCT irradiation facility. Boneca Corporation is responsible for the patient irradia-

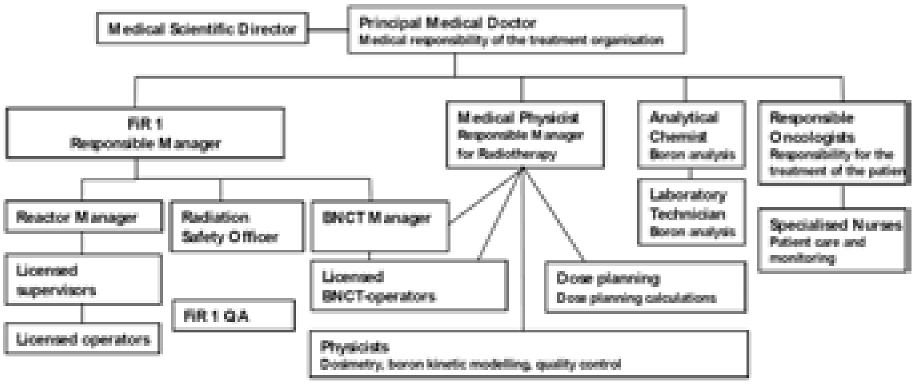


FIG. 3. The management organization for the radiotherapy licence.

tions, especially for the dose the patients are receiving, e.g. dosimetry, treatment planning, boron determination and patient positioning into the neutron field.

5. CLINICAL TRIALS

Two clinical trials are currently running at the FiR 1 BNCT facility [15]. Since May 1999, over 20 patients with glioblastoma, until now an incurable brain tumour, have been treated with boronophenylalanine (BPA)-based BNCT within a context of a prospective clinical trial (protocol P-01). BPA-fructose is given at the BNCT facility as a two-hour infusion prior to neutron beam irradiation. Blood samples for monitoring whole blood boron concentration are analysed for blood boron concentration using inductively coupled plasma-atomic emission spectrometry (ICP-AES) in a dedicated analytical laboratory at the reactor [16]. The irradiation is given as a single fraction from two fields. The irradiation procedure typically lasts for about one hour. In another trial (protocol P-03), some patients with recurring or progressing glioblastoma following surgery and conventional cranial radiotherapy have already been treated with BPA-based BNCT. The conclusion has been that BPA-based BNCT has been relatively well tolerated both in previously irradiated and unirradiated glioblastoma patients. Efficacy comparisons with conventional photon radiation are difficult due to patient selection and confounding factors, such as other treatments given, but the results support continuation of clinical research on BPA-based BNCT. [15]

6. PRODUCTION OF ISOTOPES AND OTHER IRRADIATION SERVICES

Although the BNCT work dominates the current utilization of the reactor, one or two days per week are used for other purposes such as isotope production and neutron activation analysis. The main routinely produced isotope is Br-82, either in the form of KBr or ethylene bromide. They are used by customers for flow measurements in industry. Typical produced activity per irradiation capsule is around 40 GBq. Also other isotopes used in measurements in industry are produced. The volume of neutron activation analysis is dramatically smaller than in the 1970s and 1980s, when the reactor was operated close to daily only for activation analysis [17]. Occasionally, special isotopes or other irradiation services are produced for customers' R&D projects.

7. FINANCES

The expenditure of designing and accomplishing the construction work of the BNCT facility at FiR 1 was about 4 million euros. Half of that was caused by the renovation of the reactor hall and the other half has been spent on the research, development and construction of the epithermal beam. The epithermal irradiation facility was constructed by the VTT under contract with the Radtek Inc., Espoo. Radtek is a company formed to combine private capital and State technology development funding (TEKES-Technology Development Centre Finland, Sitra) for this purpose.

The basic maintenance cost of the reactor is about 400 thousand euros per year, including licensing administration. Also the operational costs of the reactor are moderate, as one operation shift includes only the reactor operator and the shift supervisor. They are not even fully occupied by the reactor operation but are allowed to perform other duties during the reactor shift. With an increasing number of patients, more reactor operators have to be involved causing stepwise increases in the operation costs. The radiation protection has one duty officer. If the demand so requires, the reactor can be made available for BNCT treatments from early morning until late evening, allowing irradiation of several patients in a day.

The reactor is considered at the VTT as a self-supportive service unit without financial support from VTT basic funding or other government sources. The funding is totally based on sales and contracts with the customers.

8. CONCLUSIONS

Close to 30 patients have been treated at FiR 1 since May 1999, when the licence for patient treatment was granted to the responsible BNCT treatment organization, Boneca Corporation. The VTT, as the reactor operator, has a long term contract with the Boneca Corp. to provide the facility and irradiation services for the patient treatments. The BNCT facility has been licensed for clinical use and is being surveyed by several national public health authorities, including the STUK. The treatments are given in collaboration with the Helsinki University Central Hospital, located only about 6 km from the reactor.

FiR 1 also has an important national role in providing local enterprises and research institutions in the fields of industrial measurements, pharmaceuticals, electronics, etc. with isotope production and activation analysis services.

ACKNOWLEDGEMENT

The contribution of all the personnel at the FiR 1 reactor and in the Finnish BNCT project for the development, construction, operation and maintenance of the BNCT facility, as well as the maintenance and operation of the reactor is acknowledged.

REFERENCES

[1] AUTERINEN, I., HIISMKI, P., Epithermal BNCT neutron beam design for a Triga II reactor, Advances in Neutron Capture Therapy (Proc. 5th Int. Symp. Columbus, OH, 1992) (SOLOWAY, A.H., BARTH, R.F., Eds), Plenum Press, New York (1993) 81–84.

[2] HIISMKI, P., AUTERINEN, I., FRKKIL, M., The Finnish BNCT Program, An Overview, CLINCT BNCT (Proc. Workshop, Helsinki, 1993) (AUTERINEN, I., KALLIO, M., Eds), Helsinki University of Technology series, TKK-F-A718, Helsinki University of Technology, Helsinki (1994) 14–24.

[3] SAVOLAINEN, S., AUTERINEN, I., KALLIO, M., et al., The Finnish boron neutron capture therapy program—an overview on scientific projects, Excerpta Medica, Int. Congress Series 1132, **1** (1997) 342–347.

[4] LEPPL, J., et al., Accumulation of 99m-Tc-LDL in human malignant glioma, Br. J. Cancer **71** (1995) 383–387.

[5] AUTERINEN, I., et al., Metamorphosis of a 35 years old Triga reactor into a modern BNCT facility, Frontiers in Neutron Capture Therapy (HAWTHORNE, M.F., SHELLY, K., WIERSEMA, R.J., Eds), Vol. 1, Plenum Press, New York, (2001) 267–275.
[6] SAVOLAINEN, S., et al., Dosimetry chain for the dogs irradiated in the epithermal neutron beam at the Finnish BNCT facility, Frontiers in Neutron Capture Therapy (HAWTHORNE, M.F., SHELLY, K., WIERSEMA, R.J., Eds), Vol. 2, Kluwer Academic/Plenum (2001) 1245–1250.
[7] SERN, T., AUTERINEN, I., SEPPL, T., KOTILUOTO, P., Spectrum measurements and calculations in the epithermal neutron beam at the FiR 1 BNCT facility, VTT Symp. 197 on TRIGA (Proc. 15th European Conf. Espoo, 1999) (SALMENHAARA, S., Ed.), Libella, Espoo (1999) 67–179.
[8] KOSUNEN, A., KORTESNIEMI, M., YL-MELLA, H., SEPPL, T., LAMPINEN, J., SERN, T., AUTERINEN, I., JRVINEN, H., SAVOLAINEN, S., Twin ionisation chambers for dose determinations in phantom in an epithermal neutron beam, Radiat. Prot. Dosim. **81** (1999) 187–194.
[9] WHEELER, F.J., et al., Boron neutron capture therapy (BNCT): Implications of neutron beam and boron compound characteristics, Med. Phys. **26** (1999) 1237–1244.
[10] HIISMKI, P., AUTERINEN, I., Patent FI-92890 (1995).
[11] HIISMKI, P., AUTERINEN, I., US Patent 5,703,918 (1997).
[12] AUTERINEN, I., HIISMKI, P., SERN, T., SEPPL, T., Creating an Epithermal Neutron Field for BNCT Using a Triga Reactor and an Aluminum Fluoride Composite as Neutron Moderator, Technologies for the New Century: 1998 ANS Radiation Protection and Shielding Division (Proc. Topical Conf., 1998), ANS (1998), Vol. II, pp. 77–83.
[13] SEPPL, T., SEREN, T., AUTERINEN, I., Source characterisation for the rtt_MC treatment planning program at FiR 1, Frontiers in Neutron Capture Therapy for Cancer (Proc. 8th Int. Symp., Los Angeles, 1998) Frontiers in Neutron Capture Therapy (HAWTHORNE, M.F., SHELLY, K., WIERSEMA, R.J., Eds) Vol. 1, Kluwer Academic/Plenum Publishers, New York (2001) 219–224.
[14] KOTILUOTO, P., HIISMKI, P., AUTERINEN, I., SEPPL, T., ASCHAN, C., Shielding design and calculations for the Finnish BNCT facility, Frontiers in Neutron Capture Therapy (HAWTHORNE, M.F., SHELLY, K., WIERSEMA, R.J., Eds) Vol. 1, Kluwer Academic/Plenum Publishers, New York (2001) 623–628.
[15] JOENSUU, H., KANKAANRANTA, L., SEPPL, T., et al., Boron neutron capture therapy of brain tumors: Clinical trials at the Finnish facility using boronophenylalanine, J. Neuro-Oncology **62** (2003) 123–34.
[16] LAAKSO, J., KULVIK, M., RUOKONEN, I., VHTALO, J., ZILLIACUS, R., FRKKIL, M., KALLIO, M., Atomic emission method for total boron in blood during neutron-capture therapy, Clin. Chem. **47** (2001) 1796–1803.

[17] ROSENBERG, R., LIPPONEN, M., VNSK, L., Neutron activation analysis of geological samples in free competition—a case history from Finland", Comparison of Nuclear Analytical Methods with Competitive Methods, IAEA Advisory Group Mtg, Oak Ridge, TN, 1986, IAEA-TECDOC-409, IAEA, Vienna (1987).

IAEA-CN-100/97

ROLE OF THE TAPIRO FAST RESEARCH REACTOR IN NEUTRON CAPTURE THERAPY IN ITALY
Calculations and measurements

G. ROSI
Italian Agency for New Technologies, Energy and Environment (ENEA), Rome
Email: giancarlo.rosi@casaccia.enea.it

S. AGOSTEO
Department of Nuclear Energy, Milan Polytechnic and INFN, Milan

K.W. BURN, L. CASALINI, E. NAVA, M. SAROTTO
Italian Agency for New Technologies, Energy and Environment (ENEA), Bologna

M. CORTESI, G. GAMBARINI, S. GAY
Department of Physics, Milan University and INFN, Milan

F. D'ERRICO
Department of Mechanical, Nuclear and Production Engineering, Pisa University, Pisa

Italy

Abstract

Thermal-neutron research reactors are currently the most common source of neutron beams for both research and clinical trials of neutron capture therapy (NCT). Neutron spectra suitable for NCT are typically produced either by beam filtering or spectrum shifting techniques. However, fast-neutron reactors are also being considered for NCT application as it is recognized that they may allow for improved beam quality. TAPIRO is a low power, high flux, highly enriched (93.5% ^{235}U) fast reactor. The power is 5 kW and the maximum neutron flux in the core is 3×10^{12} cm$^{-2}\cdot$s^{-1}. Both a thermal and an epithermal column have been designed and constructed, aimed at dosimetry and animal experiments. The configurations of the columns have been designed by means of Monte Carlo calculations. The columns have been characterized by means of measurements performed with activation techniques and thermoluminescence and gel dosimeters. Experimental results have shown good consistency with calculations. Moreover, they have confirmed the good quality of the beams obtainable with such a reactor. An epithermal column for clinical trials of patients with brain gliomas has been designed and is under construction. The treatment planning figures-of-merit in an anthropomorphic phantom look very satisfactory.

1. INTRODUCTION

Neutron capture therapy (NCT) is an experimental modality for cancer radiotherapy which is receiving increasing interest throughout the world. Its goal is that of delivering high linear energy transfer (LET) radiation to tumours at the cellular level. This objective is accomplished by selectively concentrating in the tumour tissue a suitable isotope having a high thermal neutron cross section and whose reaction products are heavy charged particles. The current isotope of choice is the naturally occurring ^{10}B (σ_{th} = 3837 b). The $^{10}B(n,\alpha)^{7}Li$ reaction yields high LET secondary particles having a range in tissue comparable with cellular dimensions. Current research aims at optimizing tumour selectivity of boron drugs, at improving the neutron beam quality and at measuring the actual boron distribution.

Boron neutron capture therapy (BNCT) is an interdisciplinary field involving chemistry, pharmacology, biology, physics and engineering. Experimentation on animals is mandatory. Research in BNCT is performed utilizing both thermal and epithermal neutron beams. In general, thermal neutron fields have less undesired secondary radiation components than epithermal fields. In radiobiology experimentation, thermal neutrons are more appropriate for irradiating cell cultures to investigate boron carrier compounds. Moreover, for BNCT treatment of small animals (either with skin melanoma or glioblastoma) thermal neutrons are required. Finally, thermal neutron fields are necessary in order to test new instruments and methods for radiation dosimetry. Experiments in the above mentioned areas have been carried out or are in progress at the TAPIRO thermal neutron facility. Epithermal neutrons penetrate more deeply into tissue, which acts as a moderator, and are thus used in non-superficial clinical applications.

At present, nuclear reactors available for NCT experimentation and trials are thermal research reactors, suitably modified in order to have the desired neutron flux and energy spectrum at the treatment position. To this end, fast neutrons coming from the reactor core are moderated by spectrum shifting or filtering techniques.

However, it is generally accepted that:

> "While the majority of nuclear reactors potentially available for NCT are thermal reactors, a few fast reactors are also found. Since the initial source of neutrons at the irradiation position is fast neutrons leaking from the core, a fast reactor can have much higher flux-to-power ratio than a thermal one of the same power. Indeed, it appears that a 5 kW fast reactor can produce sufficient epithermal neutrons for patient treatment. The low power and compact core of a fast reactor permit a very compact NCT facility…"([1], p. 9).

The TAPIRO fast research reactor, working at 5 kW maximum power, has been shown to produce a suitable epithermal neutron flux for patient treatment.

2. THE TAPIRO REACTOR

TAPIRO, shown in Fig. 1, is a small and very compact fast research reactor located at ENEA Casaccia, near Rome, Italy. The neutron flux at the core centre is 3×10^{12} cm-^{2}·s-^{1} at maximum power. The reactor has a cylindrical core (12.58 cm diameter and 10.87 cm height) made of 93.5% enriched uranium metal in a uranium-molybdenum alloy (22.2 kg ^{235}U) which is totally surrounded by a copper reflector. The cylindrical reflector is formed by two

FIG. 1. Picture of the TAPIRO reactor.

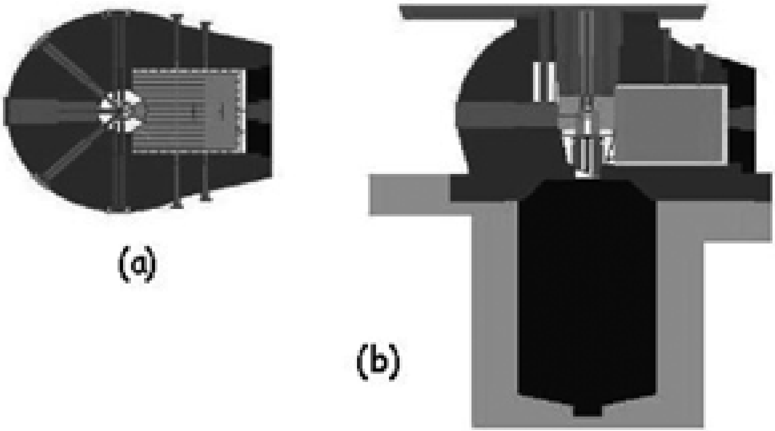


FIG. 2. Horizontal (a) and vertical (b) sections of the TAPIRO reactor.

coaxial parts: an inner one, up to 17.4 cm radius, and an outer one up to 40.0 cm radius. The height of the reflector is 72.0 cm. The reactor is surrounded by borate concrete shielding about 170 cm thick. Horizontal and vertical sections of TAPIRO are shown in Fig. 2.

The concrete biological shield has been removed from one azimuthal sector of the reactor to create a cavity for the thermal or epithermal columns. In the same sector, the outer part of the copper reflector has also been removed, and the resulting window is currently filled with alumina (Al_2O_3, density 1.3 g·cm^{-3}).

In order to achieve greater flexibility of use of the facility, the thermal and epithermal columns have been set up on two different trolleys, which can be separately driven into the cavity in the borate concrete shield, depending on the experimental requirement.

Since 1971, TAPIRO has been used for fast reactor shielding experiments, biological effect studies on animals and cells, neutron damage of electronic devices, etc. More recently, its utilization has been mainly devoted to research related to BNCT.

3. THE THERMAL COLUMN FOR THE BNCT EXPERIMENTAL PROGRAMME

The column has been designed [2] by means of Monte Carlo simulations with the MCNP-4B code [3] that models neutron and gamma transport in

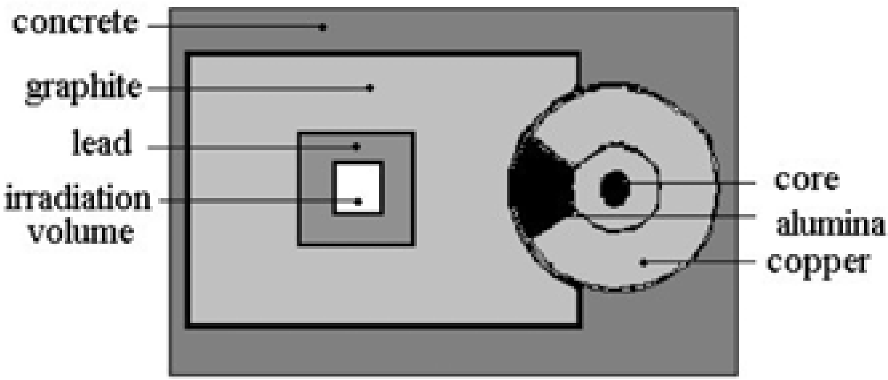


FIG. 3. Section of the thermal column for dosimetry experiments.

general geometries. Simulations were made to calculate the energy spectrum and the thermal neutron flux at various depths in the graphite moderator in order to determine the optimal configuration.

For the selected layout, thermal fluxes (< 0.4 eV) and neutron energy spectra have been calculated inside the irradiation volume both in air and in a cylindrical polyethylene phantom (16 cm diameter, 14 cm height). The energy spectra in air and in the middle of the phantom at the centre of the irradiation volume are shown in Fig. 4.

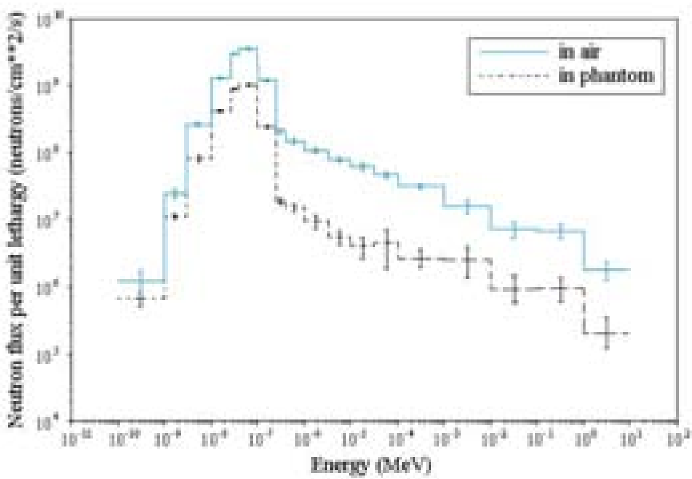


FIG. 4. Calculated energy spectra at the centre of the irradiation volume (in air and in the middle of the phantom).

Measurements of gamma absorbed dose and of thermal neutron fluence have been performed both in-air and in-phantom by means of gel dosimeters [4, 5] which can separate the various dose components (particularly useful in a mixed epithermal beam—see next section). Thermoluminescent dosimeters (TLD) have also been employed. Thermal neutron fluxes have been measured by means of activation detectors, utilizing the standard technique of bare and Cd-covered gold foils. In the centre of the irradiation cavity, the obtained gamma and fast neutron dose rates are respectively 0.057±0.006 Gy/min and 0.005±0.001 Gy/min, while the thermal neutron flux is nearly 5×10^{8} cm-^{2}·s-^{1} at maximum reactor power. In Fig. 5, the thermal neutron flux in-air along the beam axis in the irradiation cavity is shown as a function of distance into the cavity.

4. THE EPITHERMAL COLUMN FOR THE BNCT EXPERIMENTAL PROGRAMME

A collimated epithermal beam for BNCT research has been designed and set up at TAPIRO based on an initial configuration proposed in [6], then modified slightly as in [7]. The column has been constructed to perform BNCT dosimetry and radiobiology experiments as well as irradiations of medium size animals.

4.1. Design and realization of the column

Monte Carlo simulations were used in the design of the column, as is described in greater detail in the next section on the design of the column for

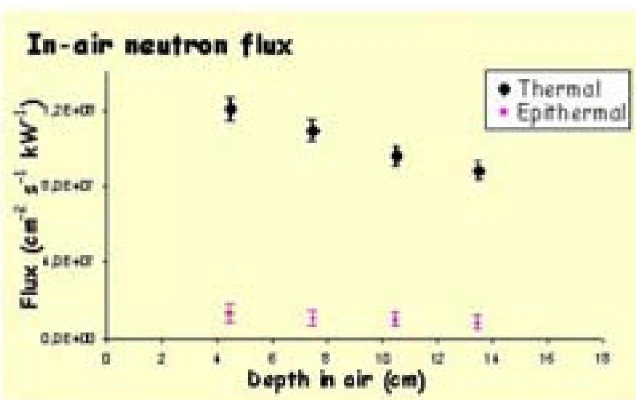


FIG. 5. In-air thermal neutron flux in the irradiation volume.

clinical trials. In addition to the Al_2O_3 in the window of the Cu reflector, compressed AlF_3 (1.85 g·cm^{-3}) was employed as moderating material. In this material, the cross-section resonances of each component element overlap well, especially at fast neutron energies.

The moderator is surrounded by nickel and followed by a thin Cd thermal neutron shield, a Pb gamma shield, and a Pb collimator. In Fig. 6, a section of the epithermal column is shown. The irradiation volume is a parallelepiped-shaped chamber, of 40 × 40 cm^2 cross-section and 70 cm depth. The collimator opening is 10 × 10 cm^2. In Fig. 7, the calculated spectrum at the collimator exit is shown.

4.2. Characterization of the column

The thermal and epithermal (0.4–10 keV) fluence rate of neutrons was measured at the collimator exit of the epithermal column. The measurements were performed with activation techniques using bare and cadmium-covered gold foils. The activity of the activated foils was assessed with a NaI(Tl) scintillator. The thermal and epithermal fluence rates (Table 1) were estimated from the reaction rate of the gold foils using the cross-section value at 0.025 eV and the resonance integral, respectively.

The uncertainties listed in Table 1 are both stochastic and systematic (normalization or bias uncertainties). The normalization uncertainties were estimated, accounting for the weight of the gold foils, the peak-efficiency of the

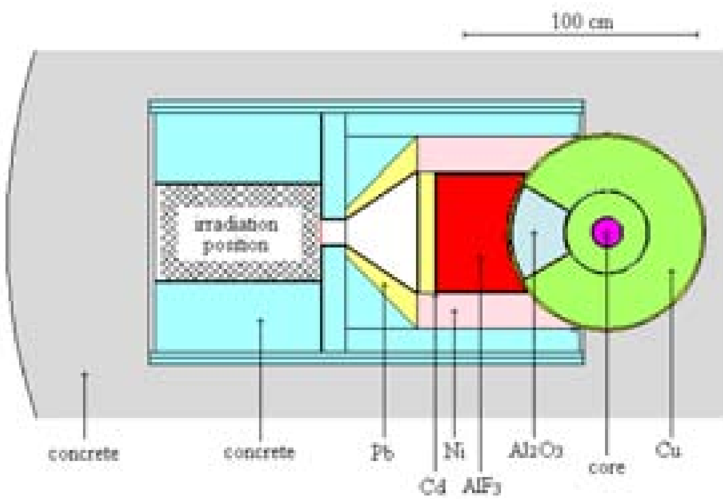


FIG. 6. Section of the epithermal column.

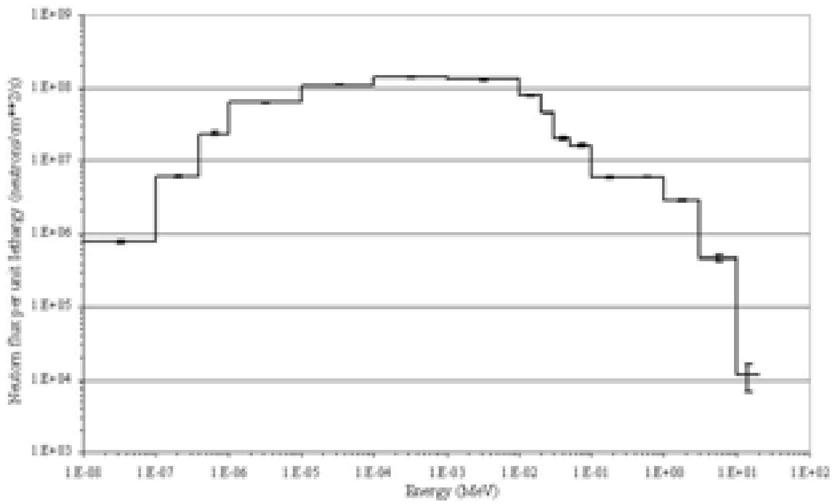


FIG. 7. Calculated spectrum at the collimator exit of the epithermal column.

NaI scintillator, the depression factors of thermal and epithermal fluence rates and the cadmium correction factor.

The gamma dose has been measured by means of CaF_2:Tm dosimeters (TLD-300) that are insensitive to epithermal and thermal neutrons. Gel dosimeters were employed to acquire images and profiles of the gamma dose and fast neutron dose.

The values of the measured and calculated neutron fluxes and the dose rate at the collimator exit per unit fluence are reported in Table 1.

In Fig. 8, gamma and fast neutron dose rates, measured with gel dosimeters and with TLD-300 at various positions along the beam axis, are reported versus the distance from the collimator opening. The gamma dose values turned out to be too high, in particular at the collimator opening and therefore a lead plug with a thickness of 5 cm has been introduced in the collimator mouth. After this variation, the gamma dose on the central axis of the column at the collimator opening has resulted in being $(2.6\pm0.4)\times10^{-13}$ ($Gy\cdot cm^2$).

The reliability of the Monte Carlo calculations of the high energy part of the neutron spectrum was verified by means of experimental fast neutron spectrometry. This was performed in air using an original instrument based on superheated drop (bubble) detectors [8]. The neutron spectrometer uses two detectors containing, respectively, emulsions of octafluorocyclobutane and dichlorotetrafluoro-ethane which are sequentially set at 25, 30, 35, and 40× C, and thus provide two series of nested threshold-responses covering the

TABLE 1. CALCULATED AND EXPERIMENTAL FREE BEAM PARAMETERS AT THE COLLIMATOR EXIT OF THE EPITHERMAL COLUMN (THE MC ERRORS ARE ONLY STOCHASTIC WHILE THE MEASUREMENT ERRORS ARE ALSO SYSTEMATIC)

	Monte Carlo	Experimental
Fluxes ($cm^{-2}\cdot s^{-1}$)		
Epithermal neutrons	$(9.61 \pm 0.07) \times 10^{8}$	$(7.45 \pm 0.15) \times 10^{8}$
Thermal neutrons	$(3.22 \pm 0.10) \times 10^{7}$	$(1.38 \pm 0.15) \times 10^{7}$
Dose/epithermal flux ($Gy\cdot cm^2$)		
D_{fast}/Φ_{epi}	$(5.38 \pm 0.07) \times 10^{-13}$	$(5.06 \pm 0.76) \times 10^{-13}$
$D\gamma/\Phi_{epi}$	$(3.74 \pm 0.13) \times 10^{-13}$	$(6.90 \pm 1.00) \times 10^{-13}$

0.1-1 MeV and 1-10 MeV intervals. Results of simulations and measurements were in close agreement (Fig. 9) and contributed to validate the computational procedures, while providing information useful for the radiation protection of future BNCT patients.

In order to further check the suitability of the TAPIRO epithermal beam, depth-dose profiles have been measured with gel dosimeters in a cylindrical polyethylene phantom with dimensions near those of a human head (16 cm diameter, 14 cm height). Fig. 10 shows gamma dose profiles in the phantom with its base placed at the collimator opening, before and after the introduction of the lead plug.

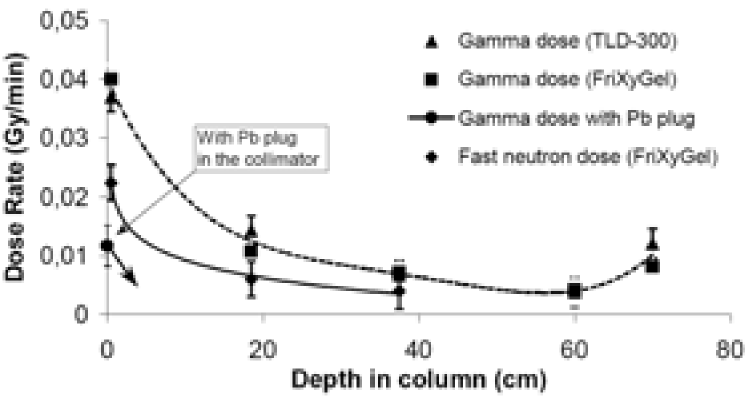


FIG. 8. Gamma and fast neutron dose rates measured with gel dosimeters and with TLD-300.

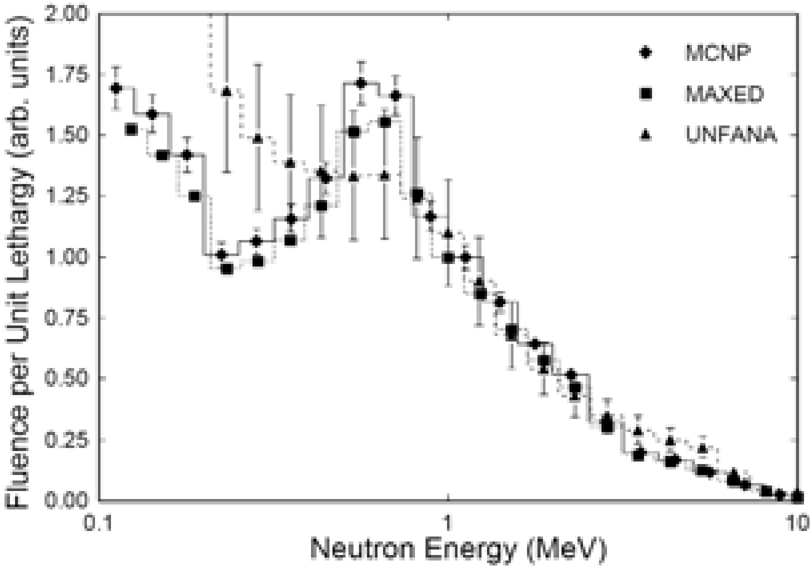


FIG. 9. Fast neutron spectra determined with superheated emulsions (unfolded with the MAXED and UNFANA codes) compared with MCNP results.

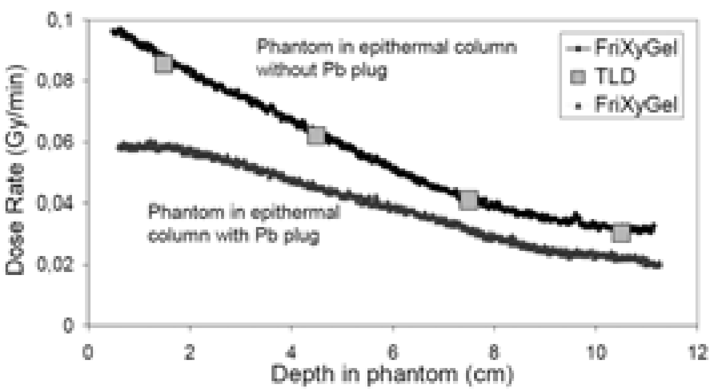


FIG. 10. Gamma dose profiles in the phantom with and without the lead plug at the collimator exit.

5. THE EPITHERMAL COLUMN FOR THE BNCT CLINICAL TRIALS (UNDER CONSTRUCTION)

For patient radiotherapy, the irradiation position at 119 cm from the core centre, used for BNCT experimentation, had to be moved a further 143 cm away from the core to outside the biological shield. Since the reactor power is relatively low (5 kW), careful design was necessary to achieve a sufficient neutron flux at the irradiation position. In the design process, coupling effects between the column layout and the reactivity of TAPIRO, due to the compact nature of the core, had to be taken into account. Stringent requirements were set on the beam quality in terms of fast neutron and gamma background. Design and optimization of an epithermal column for patient treatment with Monte Carlo can be extremely time consuming, particularly when neutron economy is so critical. Hence, variance reduction techniques in the Monte Carlo simulations played an important role.

Our approach was to use a Monte Carlo variance reduction optimizer developed in-house, called the DSA. This is described in Ref. [9], where an early design of an epithermal column is described, bearing no relation to the present one. The DSA ability to treat several responses simultaneously resulted in being particularly helpful in our applications. (These responses may be the epithermal neutron flux, the fast neutron dose and the gamma dose associated with the free beam at the collimator opening or, alternatively, the dose components at various depths in a phantom.)

The final column design [10] contains some interesting features, such as a thin (75 mm) nickel reflector, an epithermal neutron cavity, no thermal neutron or gamma shield and a long, unconventionally shaped, Pb collimator (see Fig. 11). The moderator is again compressed AlF_3 (1.85 g·cm^{-3}).

Data for the beam in air at the collimator opening and for treatment planning figures-of-merit in a realistic phantom 'ADAM' [11] are reported in Table 2. The beam quality appears adequate in terms of the current IAEA recommendations ([1], pp. 7–8). Moreover, the beam parameters compare well with those of other facilities presently utilized for BNCT trials. It should be noted that while the in-phantom figures-of-merit are a good measure of the therapeutic efficacy, they are strongly dependent on the particular hypotheses adopted (see Ref. [10]).

Some dose profiles from the various beam components are shown in Fig. 12. These results differ from those reported in Ref. [10] due to the use of an improved thermal neutron scattering model and correction of a dose conversion factor.

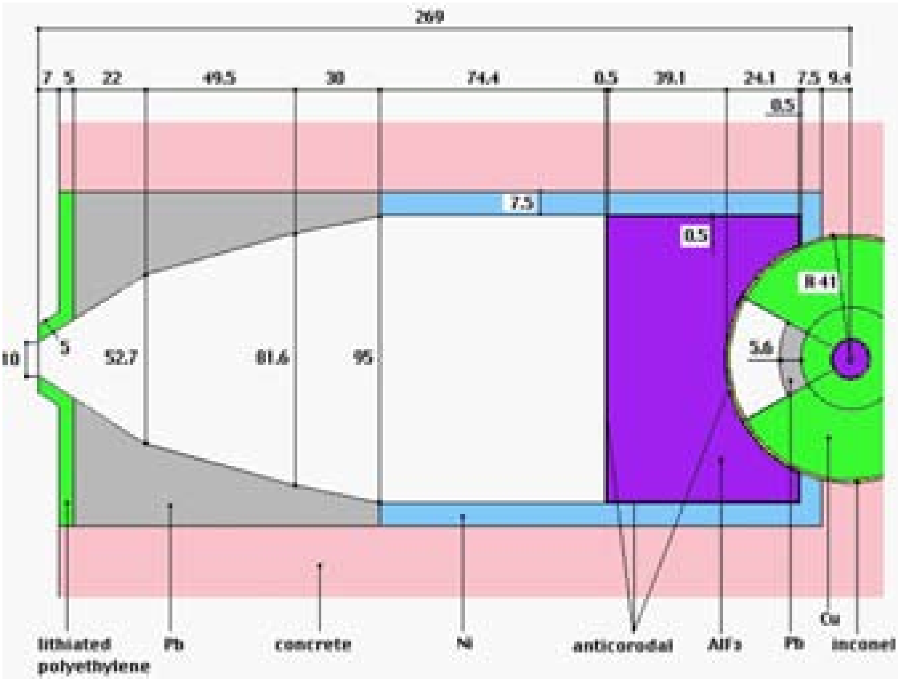


FIG. 11. Section of the epithermal column for patient irradiations (dimensions in cm).

TABLE 2. CALCULATED FREE BEAM PARAMETERS AND TREATMENT PLANNING FIGURES-OF-MERIT IN A REALISTIC PHANTOM 'ADAM' IN THE EPITHERMAL COLUMN FOR PATIENT IRRADIATIONS

Free beam parameters	
$\Phi_{n\ epith\ (0.4\ eV\ -\ 10\ keV)}$	8.0×10^{8} n·cm^{-2}·s^{-1}
Neutron dose in water (E > 10 keV) / $\Phi_{n\ epith}$	4.1×10^{-13} Gy·cm^{2}
γ whole body dose /$\Phi_{n\ epith}$	3.5×10^{-13} Gy·cm^{2}
$J_{n\ epith}$ / $\Phi_{n\ epith}$	0.73
In-phantom treatment planning figures-of-merit	
Advantage depth (in brain)	74 mm
Treatment time (max. healthy tissue dose 12.6 Gy Eq)	54 min
ADDR	0.2345 Gy Eq/min
Therapeutic depth (in brain)	52 mm
Peak therapeutic ratio (PTR)	4.25

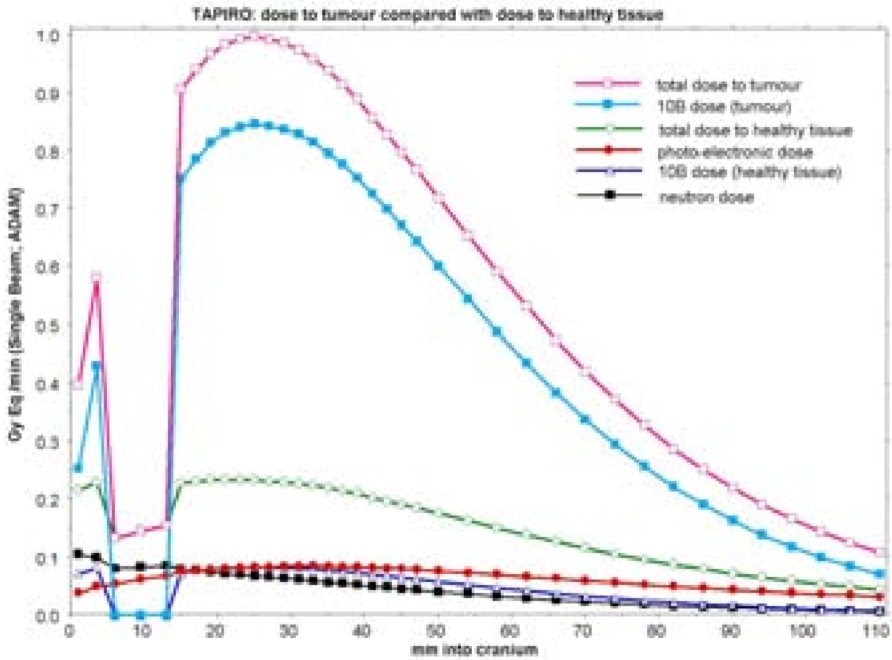


FIG. 12. Dose profiles from the various beam components.

6. CONCLUDING REMARKS

Recent BNCT experimentation at TAPIRO with thermal and epithermal columns has opened the way to an ensuing phase of clinical trials. The low neutron inventory at the TAPIRO reactor, due to a relatively low peak power, has been addressed. Projected therapy times for a single beam are around 50 min.

Upon construction of the new epithermal column, extensive investigations will be required for the characterization of the facility. In addition to the uncertainties usually associated with the design of a complex epithermal beam, two factors peculiar to TAPIRO might lead to discrepancies between calculation and measurement and, therefore, performances outside the design parameters:

— The singular reactor design and low power (only one other similar model has been built) imply that care must be taken in evaluating the absolute source strength;

— Large amounts of Cu are present; this is an unusual element in neutronics, and its neutron cross sections are affected by larger uncertainties compared with those of the more common structural elements, such as the components of steel.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Current Status of Neutron Capture Therapy, IAEA-TECDOC-1223, IAEA, Vienna (2001).

[2] NAVA, E., Characterization of a neutron beam designed for boron neutron capture therapy (BNCT), Master's thesis, Health Physics School, Physics Department, University of Bologna (1999) (in Italian).

[3] BRIESMEISTER, J.F. (Ed.), MCNP^{TM}, A General Monte Carlo N-Particle Transport Code, version 4B, LA-12625-M (1997).

[4] GAMBARINI, G., et al., Imaging and profiling of absorbed dose in neutron capture therapy, IEEE Transactions in Nuclear Science, 48 (2001) 780–784.

[5] GAMBARINI, G., et al., Fricke-gel dosimetry in boron neutron capture therapy, Radiat. Prot. Dosim. **101** (2002) 419–422.

[6] NAVA, E., Preliminary study for the implementation in Italy of boron neutron capture therapy (BNCT) for malignant gliomas, internal note, 30 October 2000, Dept. of Nucl. Eng., Pisa University, ENEA, Pisa (in Italian).

[7] BURN, K.W., NAVA, E., Preliminary feasibility study for the conversion of the TAPIRO reactor epithermal column from irradiation of guinea-pigs to patient treatment using BNCT, ENEA Technical Note, YT-SBD-0001 (2001) (in Italian).

[8] D'ERRICO, F., et al., Fast Neutron Spectrometry of BNCT Beams, Proc. 10th Int. Congr. Neutron Capture Therapy for Cancer (Essen, Germany, Sept 9-13, 2002).

[9] BURN, K.W., et al., Variance Reduction with Multiple Responses, Proc. Conf. Monte Carlo, 2000, Lisbon (2000) 687–695.

[10] BURN, K.W., et al., Design of an Epithermal Facility for Treating Patients with Brain Gliomas at the TAPIRO Fast Reactor at ENEA Casaccia, Proc. 10th Int. Congr. Neutron Capture Therapy for Cancer (Essen, Germany, Sept 9-13, 2002).

[11] KRAMER, R., et al., GSF-Bericht S-885 GSF, Forschungszentrum fur Umwelt und Gesundheit, Neuherberg, Germany (1982).

A COUPLED CALCULATION SYSTEM FOR OPTIMAL IN-CORE FUEL MANAGEMENT IN RESEARCH REACTORS

H. MAZROU
Laboratoire des Analyses de Sûreté,
Centre de Recherche Nucléaire d'Alger (CRNA),
Algiers
Email: mazrou_h@comena-dz.org

M. HAMADOUCHE
Département de mathématiques appliquées, Faculté des Sciences,
Université de Blida

Algeria

Abstract

The paper presents a coupled method to solve the problem of finding an optimal configuration of fuel elements in research reactor cores. Finding the optimal solution always requires a huge amount of calculations by traditional methods. Thus, in performing this work, the investigated way followed to overcome such difficulties, was a judicious combination of the artificial neural network (ANN) technique, together with the well known stochastic method which is simulated annealing (SA). It has been shown that the most distinguishing and attractive feature of such a system is the computational efficiency and an increasing probability in obtaining optimized solutions with reasonable error. Neural network offers a very fast core parameter prediction tool with reasonable accuracy, and the simulated annealing method offers a very effective searching procedure which avoids local minimum. A series of tests have been performed using a modified core configuration of the benchmark 10 MW IAEA low enriched uranium (LEU) research reactor and the result achieved is the optimum configuration of the studied core.

1. INTRODUCTION

In-core fuel management is one of the frequent safety headaches faced during the useful life of a nuclear reactor, because of the existence of a huge number of fuel configurations, arrangements and associated decisions to be taken, for the potential best configuration which satisfies established safety constraints.

In the present paper, an original computational strategy, inspired from Kim's paper [1], was proposed to solve adequately this problem. It requires two calculation stages involved by the use of coupled methods connected to each other.

In the first stage, an adaptive back-propagation network (BPN), is used to predict safety core parameters P_{max} and K_{eff}. The BPN receives the allowed configurations from a previous calculation using heuristics rules and thereafter predicts P_{max} and K_{eff} very quickly [2, 3]. The SA method, in a second stage, determines whether a current candidate is better than the reference one, based on the predicted results and consequently on the value of the objective function stated.

The most appropriate approach we have used for such optimization problems was to condense the multiple objectives into a single objective function [4], which is then optimized with respect to established safety constraints.

Thus, the objective function was developed based on two performance parameters: cycle length, which can be determined through the evaluation of the effective multiplication factor K_{eff}, and power peaking factor P_{max}. The system uses optimization of these two parameters to finds patterns in which K_{eff} is maximized, whether the value of P_{max} should be lower than the reference one given before. Using several heuristic rules in an adequate search algorithm, we tried to achieve a more effective and faster searching procedure. The overall strategy employed was validated for the IAEA 10 MW benchmark core reactor [5].

As for the optimization process, the system is solved on the basis of Schirru, et al. [6] who suggest that the solution to the nuclear reactor core reload problem (CRP) is the same as the solution to the classical travelling salesman problem (TSP). In the TSP, the objective is to find the minimum distance that completes a tour for a set of cities passing only once in each city and coming back to the starting one. Thus, in the present problem, the fuel assemblies correspond to the cities in the TSP and the position in the core is related to the order of the cities to be visited.

2. CORE PARAMETER PREDICTION BY ARTIFICIAL NEURAL NETWORKS

In the present work, the investigated global strategy followed was based upon the combination use of the well known back-propagation (BP) algorithm with a special adaptive learning rate procedure, in conjunction with an appropriate selection of configurations for the training database. The motivation in using such a computational procedure lies in the fact that it will let us use merely hundreds of configurations rather than thousands, in the learning stage, as it is usually required in such typical calculations to ensure reasonable predictions. Hence, as shown in Fig. 1, a suitable neural networks development strategy was tested, based on executing the following two main calculation stages, in a independent way:

(1) Learning stage;
(2) Prediction stage.

The first stage of computational procedure consists of creating suitable networks by applying appropriate learning rules using a desired database. The information required in the related database will contain coupled input values with the corresponding target output values. These values are used to train the

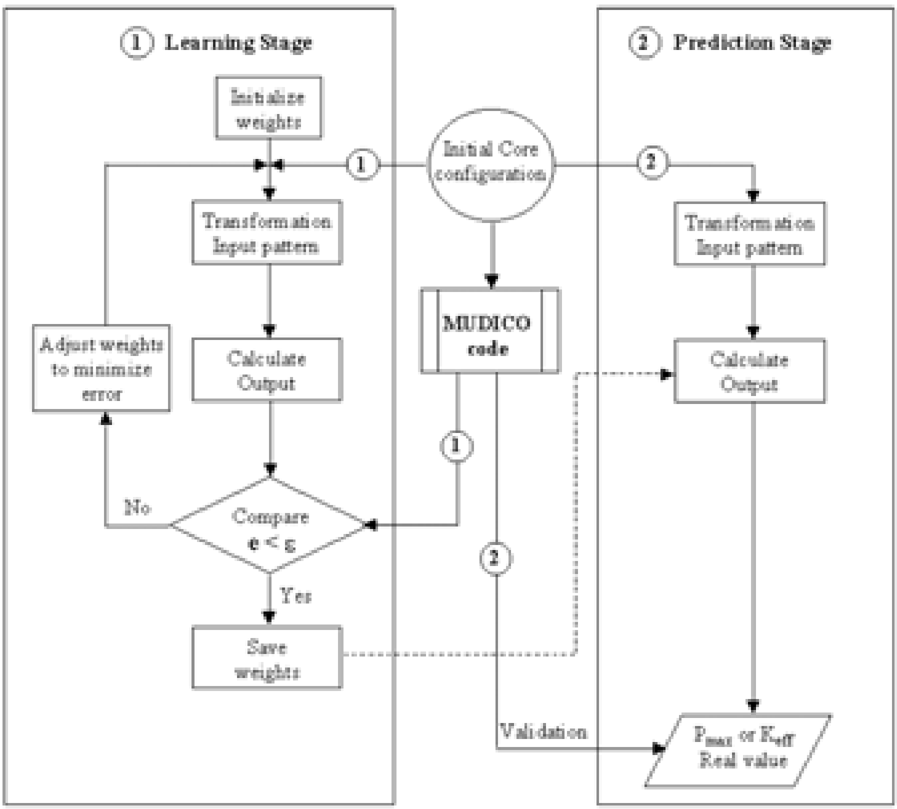


FIG. 1. Overall back-propagation computational strategy for core parameter prediction.

networks until the error reaches a desired value that is stated at the beginning of the learning process. It becomes evident that the quality of the obtained results will depend on how well knowledge is capitalized in this database. Hence, significant attention would be focused on how well this database has been created.

The second stage is the prediction one where the weights, from the interconnected neurons, have been adjusted to the desired error in the previous calculations stage. These weights will be used, in a global computational sequence, to predict the networks outputs when unseen data will be presented to the developed networks. This is the power of the network approach and one of the justified reasons for using it. The net is said to have *generalized* from the training data. This stage is necessary to test the performance of the developed neural network.

The standard back-propagation model was retained as the basic gradient descent algorithm to train the developed networks. The network weights are moved along the negative of the gradient of the performance function.

The main steps usually required in the process of developing neural network are as follows:

(1) *Create a database for training:* The core of the generic 10 MW IAEA LEU benchmark research reactor was selected to create the needed database for network training. A slight modification was introduced in its initial core arrangement, to ensure geometrical asymmetry in the core and thereafter to obtain the sufficient search space. Therefore, 30 positions were considered. In addition, for better representation of the various states of the reactor core during its useful life, several configurations have been taken into account and simulated, using fuel elements with different burn-up ranging from fresh fuel (FF) to 50% (5%, 10%, 25%, 30%, 45%, 50%). Finally, using several established heuristics rules, initial core configurations have been selected according to the following two requirements: the presence of fuel elements that influence both K_{eff} and P_{max} parameters; and the random generation of a finite set of initial configurations.

    The first criteria ensure an orientation in the research, whether the second is random and lets a diversity in the search space. At the end, a total of 800 configurations were generated, among which 700 were devoted to creating the training database, whether the remaining are considered to test the developed networks.

(2) *Construction of networks for training:* Two neural networks have been developed, based on the three layers (input-hidden-output) model. They contain 30 neurons as input, one neuron as output, whether the number

of hidden neurons was determined by increasing systematically their number during training until the minimum prediction error was reached. This result in a total of 280 hidden neurons, for the two developed networks.

(3) *Choosing a learning function:* An adaptive back-propagation algorithm based on Widrow-Hoff rules was used as the learning function. During this stage, the quadratic error E is minimized in the three-layered network according to the following formula:

$$\frac{\partial E}{\partial w_{ij}} = 0 \tag{1}$$

Adjusting the weights (w_{ij}) to minimize E at the (t+1)-th step is performed as follows:

$$w_{ij}^{t+1} = w_{ij}^{t} + \Delta w_{ij}^{t+1} \tag{2}$$

$$\Delta w_{ij} = -\eta_{ij} \frac{\partial E}{\partial w_{ij}} \tag{3}$$

Where in (3) η_{ij} is the learning parameter which is adapted after each correction step according to a suitable Jacob's rule [7, 8].

(4) *Train the network:* Supervised training has been carried out for each parameter. All the weights (w_{ij}) have been initialized randomly between [- 0.5, + 0.5] and adjusted to minimize the error E for 700 configurations. During training of the network, each time step is called an epoch and is defined to be a single sweep through all training patterns. At the end of each epoch, the weights of the network were updated. According to the convergence criteria, training was completed and weights were stored when both normalized errors reached a value of 0.0007 for K_{eff} predictions and 0.003 for P_{max} predictions. This result in a fixed training of about 36 and 24 h, respectively, for both K_{eff} and P_{max} parameters, on a small computational system (300 MHz Pentium II PC).

(5) *Testing the networks*: After training, networks were created for each parameter and tested against the unseen data. One hundred configurations were devoted to this purpose. The prediction results for K_{eff} and P_{max} parameters are shown in Figs 2 and 3, respectively. When the model was recalled, 95% of the training patterns were predicted within ± 0.7%

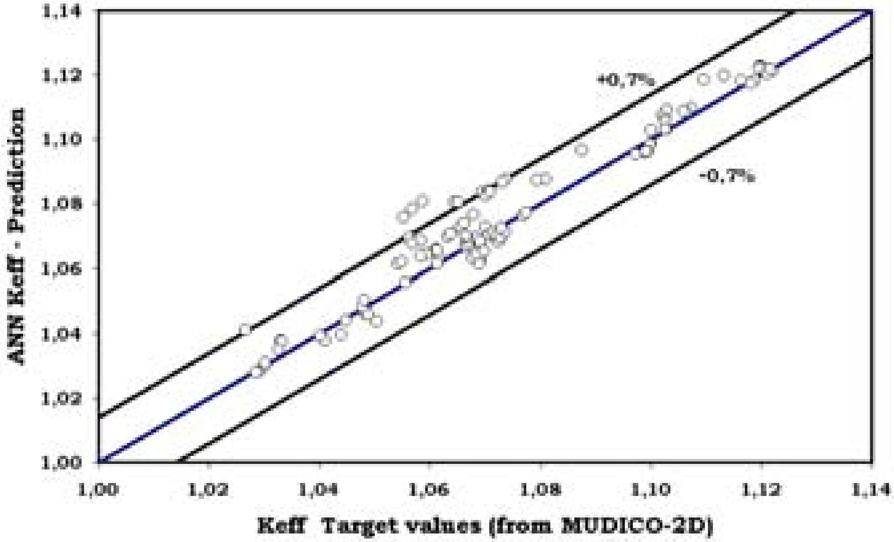


FIG. 2. K_{eff} prediction results.

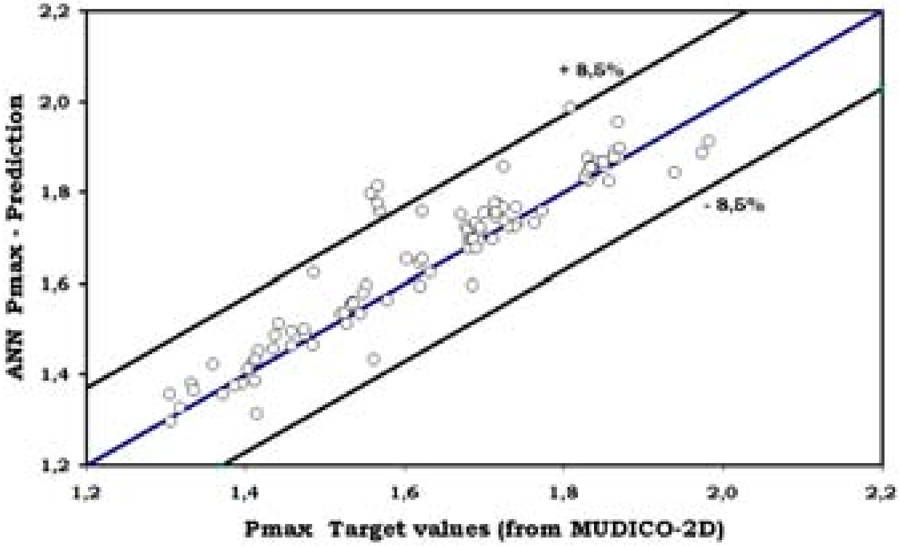


FIG. 3. P_{max} prediction results.

of the reference value for the K_{eff}, with a maximum and average error of 0.84% and 0.54%, respectively. As for P_{max}, the network predicts 95% within ± 8.5% of the reference value, with a maximum and average error of 14.0% and 7.8%, respectively.

3. OPTIMIZATION BY SIMULATING ANNEALING

The optimization by simulating annealing (OSA) algorithm is based on the natural thermodynamic phenomena of slow cooling of a solid. In the physical analogy, individual particles tend to a minimum energy state, while the solid slowly cools.

Metropolis et al. [9], in the earliest days of scientific computing, were the first to introduce a simple algorithm for an efficient simulation of the foregoing phenomena.

Kirkpatrick et al. [10], subsequently clarified 'simulated annealing' and presented it as a general tool for attacking combinatorial optimization problems. Thus, using the objective (cost) function, F, in place of the energy, the probability of accepting the candidate pattern as a new incumbent is expressed as follows (for the maximization problem):

$$P = \exp\left(-\frac{\delta F}{T}\right) \quad \text{for } \delta F \leq 0 \tag{4}$$

$$P = 1 \quad \text{for } \delta F > 0 \tag{5}$$

In equation (4), δF is the difference between the current objective function and the perturbed objective function, and T is simply a control parameter in the same unit as the objective function. The simulated annealing process consist of first 'melting' the system being optimized at a high effective temperature, then lowering the temperature by slow stages until the system 'freezes' and no further changes occur. Analogously, there is a higher probability of accepting a move that increases the objective function at higher temperatures than when the temperature is low.

The extension of this method to new combinatorial optimization problems such as nuclear fuel management was first introduced by Kropaczek and Turinsky [11]. Four ingredients are needed for a successful implementation of the OSA algorithm:

(1) A concise description of a configuration of the system;
(2) A random generator of ‘moves’ or rearrangements of the elements in a configuration;
(3) Defining an appropriate objective function that contains the trade-offs that have to be made;
(4) Determining an optimal annealing schedule of temperatures and length of times for which the system is to be evolved.

3.1. Description of the configuration of the system

In the problem studied, the choice for an initial configuration is not primordial. However, an appropriate initial configuration will let us to save time by giving us the possibility to take an initial lower temperature.

3.2. Random generator of moves

As mentioned previously, every OSA programme conducts a random search. So one prerequisite for tackling a given problem by OSA is a routine which generates appropriate random changes to a given solution. Among the consulted literature, there is an extensive application of algorithms based on the well known travelling salesman problem (TSP), which is why we adopted it in the present study. Each new configuration can be described by a simple permutation between fuel elements present in the core.

However, inappropriate ‘moves’ which may induce large positive reactivity insertion are forbidden. Thus, a list of heuristics rules were established that were to let safe permutations in the reactor core.

3.3. Objective function

For this study, the objective function was set up to optimize on BOC k_{eff} with a penalty function on the peak power constraint:

$$\text{maximize } F(p,k) = w_p P(p) + w_k (k-k_{ref}) \tag{6}$$

The penalty function for normalized power peaking is :

$$\begin{cases} p - p_{ref} \ : \ p > p_{ref} \\ P(p) = 0 \ : \ p \leq p_{ref} \end{cases}$$

Where:

p : normalized power peak
k : BOC k_{eff}
w_p: power peak weighting factor = - 1000
w_k: BOC k_{eff} weighting factor = 100
p_{ref}: power peak limit = 1.90

The high negative penalty on a violation of the power peak constraint causes the algorithm to work to satisfy this criterion before optimizing the k_{eff}.

3.4. Determining an optimal annealing schedule

The annealing schedule is very important to the success of the algorithm. The choice for a cooling strategy is then difficult: if the system is cooled too quickly, the search can become trapped in local minima; conversely, if the system is cooled too slowly, the optimization will be inefficient in approaching the optimum. One clear benefit of OSA is that while convergence on the global optimum is not guaranteed, empirical evidence suggests that it does approach the neighbourhood of the global optimum [12]. This is possible because of the statistical ability of OSA to remove itself from local minima during its search for the global optimum.

Various models have been proposed for particular applications of Metropolis algorithm [13]. Thus, for this study, we adopted the model in which the control parameter is reduced in a series of discrete stages as:

$$g(T) = \mu \times T \text{ with } 0 < \mu < 1 \qquad (7)$$

The cooling parameter μ is significant. Experience has shown that $0.85 \leq \mu \leq 0.95$ is more appropriate for detailed reload optimization because the design space is highly nonlinear. The huge scale of the reload optimization problem requires a truncated search. However, a large search sample (e.g. 10 000 or more candidates) generally yields many good patterns [14].

3.5. Condition 1

Condition 1 determines the total number of transformations of temperature made during the algorithm computational process.

It should be chosen in the way that at the end of the algorithm, the final temperature would be low enough that practically any costly transformation would not be accepted. According to the decreasing scheme adopted for the control parameter, T, as shown in equation (7) and taking a value of μ=0.95 and 60 changes of temperature, the final value of T will be of the amount of 5% of its initial value; that means, if the probability to accept any given rise of amplitude is 0.5 at the beginning of the algorithm, it will be of the order of 3×10^{-7} after 60 changes of temperature T.

3.6. Condition 2

This condition determines the number of global trials of transformations at fixed temperature. The easiest choice consists of giving this iteration number, a value which depends only on the size of the problem treated. For example, an iteration number proportional to n^2 for a TSP, in which n represent the number of positions.

4. SIMULATION RESULTS

A demonstration of the general optimization capabilities of the developed computer program may now be provided. A series of benchmark tests have been performed using a modified core configuration of the 10 MW IAEA LEU research reactor. The core was then modelled with 30 positions in the loading pattern. For this exercise, the optimization process was run using an initial fictitious configuration near the ‘beginning of cycle’ (BOC) state of the research reactor core considered. That is, some fresh fuel elements were present in the core just to prove that the established heuristics were taken into account and the optimization process works well. In addition, some sensitivity studies were also carried out to enhance the quality of the final solution obtained and consequently, to find the best configuration.

Results from using an initial core configuration randomly generated with $(K_{eff})^{ref}$ = 1.0636, and P_{max} = 1.8711 calculated by MUDICO-2D [15], in the optimization process are presented in Table 1 and illustrated in Fig. 4.

In the first test, we have generated randomly a total number of 4000 configurations, considered as our search space. Thus, using the established safety constraints, only 1130 configurations have been accepted. The ‘optimal configuration’ in the sense of best value of the objective function has been obtained after a CPU time of 172 s.

As planned in the optimization algorithm, both safety parameters of interest were improved. The multiplication factor was improved by about 7500 pcm, which is greatly appreciable while the power factor considered as a constraint parameter not exceeding a maximum value of 1.9, was decreased by a factor of 1.47 as reported in Table 1.

TABLE 1. OPTIMIZATION RESULTS ACHIEVED FOR THE PERFORMED TESTS

Parameters	Test 1	Test 2	Test 3	Test 4
Total number of changes in T (condition 1)	80	50	80	80
Configuration changes with T constant (condition 2)	50	80	125	625
Total number of generated configurations	4000	4000	10000	50000
Accepted configurations	1130	1575	2676	7095
Executing CPU time (s)	145	148	362	1515
K_{eff}	1.1184	1.1378	1.1390	1.1390
P_{max}	1.298	1.376	1.383	1.383

In the second test, we tried to see the combined effect of both conditions 1 and 2 on the final solution, maintaining the same total number of explored configurations (4000). The results obtained show that the accepted configurations (1575) are greater than the previous case, but the value of the multiplication factor was depreciated a little more than the one presented in the former case. It means that once we tried to reduce the total number of changes of the control parameter T, the achieved optimal solution has been trapped in a local minima and thus, it may expect some depreciation in the final solution comparatively to the first case studied.

Another test case conducted using bigger search space by modifying condition 2. Thus, carrying the number of generated configuration to 10 000 recommended by Galperin [16], as a minimum number to use in order to achieve a potential *global optimum*. The results show a relative improvement by about 40 pcm in the multiplication factor, while the power factor was increased slightly by a rate of 1.09, compaed with the first case.

In the last case, the explored search space has been modified again to attain a total number of 50 000 configurations generated in order to note whether a best configuration could be obtained or not. The results presented were similar to those of the third case. It can be concluded that the final solution was achieved and may be considered as a global minimum in the explored search space.

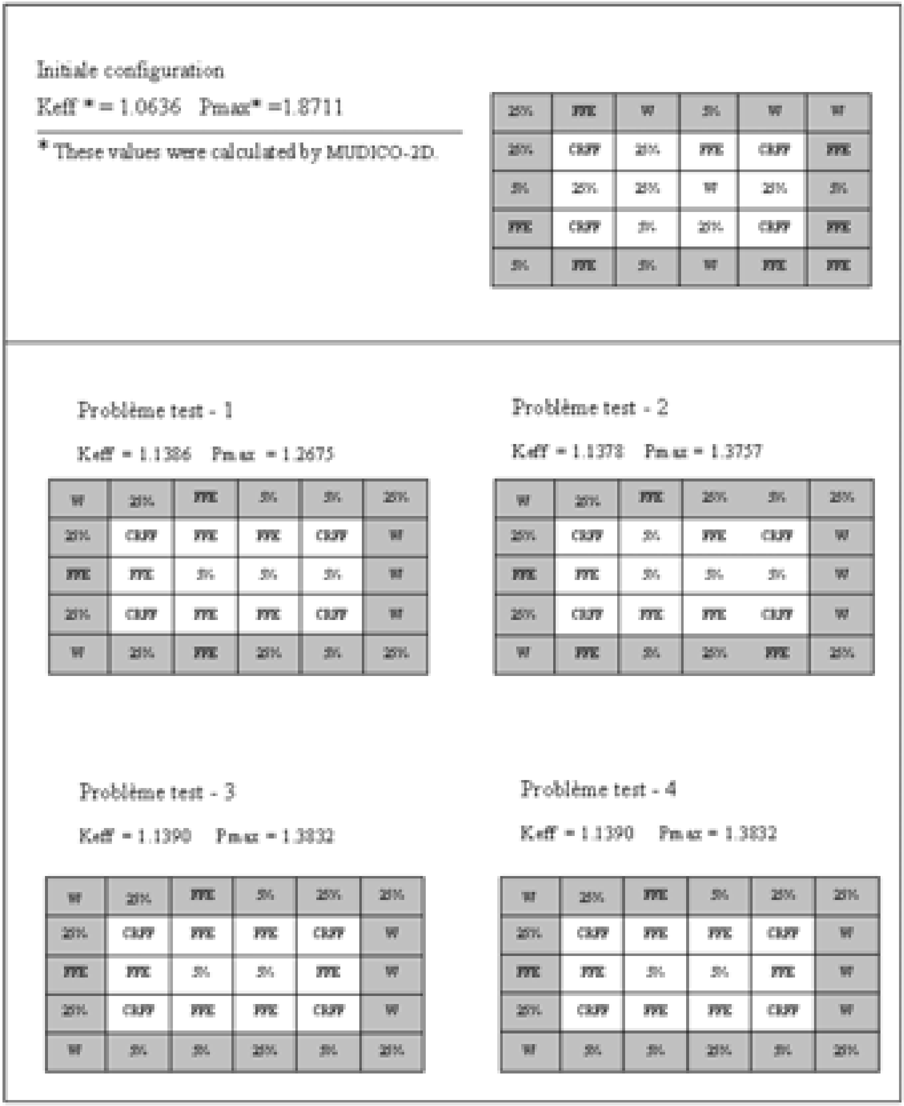


FIG. 4. Optimal configurations obtained during sensitive analysis.

5. CONCLUSION

Throughout the observed results, one can see that the optimization process works well. The large negative penalty introduced on a violation of the peak power constraint causes the algorithm to work in order, first, to satisfy the P_{max} constraint before optimizing the k_{eff}. For a given test case of about 10 000 configurations explored, the optimization process converges nearly in 10 min on a small computational system (300 MHz Pentium II PC) and it seems that an acceptable solution has been achieved in the meantime.

It should be emphasized that the computational system suggests candidate optimum LPs for a unique cycle of a research reactor. Thus, the results achieved should be considered a starting point for a more detailed analysis using improved developed in-core fuel management computer codes. Finally, the program provided with a graphical user interface (GUI), could be easily used as a preliminary and fast estimate results in experiences where core configuration rearrangement is required.

ACKNOWLEDGEMENT

This work was part of a research project performed at the Safety Analyses Laboratory of the Nuclear Research Centre of Algiers (CRNA). H. Mazrou would like to express his appreciation to the IAEA for the grant support awarded to take part at the Chile International Conference on Research Reactors.

REFERENCES

[1] KIM, H.G., CHANG, S.H., LEE, B.H., Optimal fuel loading pattern design using an artificial neural network and a fuzzy rule-based system, Nucl. Sci. Eng. **115** (1993) 152–163.

[2] KIM, H.G., CHANG, S.H., LEE, B.H., Pressurized water reactor core parameter prediction using an artificial neural network, Nucl. Sci. Eng. **113** (1993) 70–76.

[3] MAZROU, H., HAMADOUCHE, M., "Research reactor core parameter prediction using an adaptive back-propagation neural network method", accepted for publication in Transactions of American Nuclear Society, Vol. 88, pp. 312–313, San Diego, CA, (2003).

[4] DECHAINE, M.D., FELTUS, M.A., Nuclear fuel management optimization using genetic algorithms, Nucl. Technol. **111** (1995) 109–115.

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Research Reactor Core Conversion Guidebook, IAEA-TECDOC-643, IAEA, Vienna (1992).

[6] SCHIRRU, R., et al., Intelligent soft computing in nuclear engineering in Brazil, Progress in Nuclear Energy **35**3–4 (1999) 367–391.

[7] JACOBS, R.A., “Increased rates of convergence through learning rate adaptation”, Neural Networks **1** (1988) 295–307.

[8] JODOUIN, J.F., Les réseaux neuromimétiques, modèles et applications, Hermès, Paris (1994).

[9] METROPOLIS, N., ROSENBLUTH, A., ROSENBLUTH, M., TELLER, A., TELLER, E., Equation of state calculations by fast computing machines, J. Chem. Phys. **21** 1087 (1953).

[10] KIRKPATRICK, S., GELLAT, C.D., VECCHI, M.P., Optimization by simulated annealing, Science **220** 271 (1983).

[11] KROPACZEK, D.J., TURINSKY, P.J., In-core nuclear fuel management optimization for pressurized water reactors utilizing simulated annealing, Nucl. Technol. **95** 9 (1991).

[12] MOORE, B.R., TURINSKY, P.J., KARVE, A.A., FORMOSA-B: A BWR in-core fuel management optimization package, Nucl. Technol. **126** 153 (1999).

[13] PARKS, G.T., Advances in optimization and their applicability to problems in the field of nuclear science and technology, Adv. Nucl. Sci. Tech. **21** 195 (1990).

[14] STEVENS, J.G., SMITH, K.S., REMPE, K.R., DOWNAR, T.J., Optimization of pressurized water reactors shuffling by simulated annealing with heuristics, Nucl. Sci. Eng. **121** 67 (1995).

[15] IBRAHIM, K., MAZROU, H., HAMIDOUCHE, T., BENKHARFIA, H., MUDICO-2D: A two-dimensional multigroup diffusion code for perturbation calculation in light water research reactors, New Frontiers of Nuclear Technology: Reactor Physics, Safety and High-Performance Computing—Physor 2002 (Proc. Int. Conf. Seoul, 2002), American Nuclear Society, La Grange Park, IL (2002?).

[16] GALPERIN, A., Exploration of the search space of the in-core fuel management problem by knowledge-based techniques, Nucl. Sci. Eng., **119** (1995) 144–150.

NEUTRONIC DESIGN OF A COLD NEUTRON SOURCE WITH MCNP

D.F. HERGENREDER, C. LECOT, O. LOVOTTI
INVAP S.E.,
Bariloche, Argentina
Email: hergend@invap.com.ar

Abstract

The neutronic design of a cold neutron source (CNS) requires the use of powerful tools to model neutron transport as accurately as possible. For this purpose, nowadays, the increase in hardware calculation power makes it possible to make use of Monte Carlo techniques, even during the design stage. For design purposes, the goal is to find the optimal combination between positioning and geometry of the moderator chamber and composition of the moderator material to produce the maximum cold neutron flux at the experimental location. Close to the optimum balance, the influence of each of these parameters on the cold flux can be expected to be about 1–5%. These small effects must be discriminated from statistical errors without a strong increase of the calculation time. A short description of the calculation line, leading to a fast and reliable method to perform these optimization calculations with low statistical errors and times compatible with a design schedule is presented. Several parametric analyses of the design variables are presented in order to show how this calculation methodology works and how consistent their results are. The analysis was done during the design of the replacement research reactor (RRR) CNS for the Australian Nuclear Science and Technology Organisation (ANSTO). As a conclusion to the paper, we demonstrate the possibility to apply Monte Carlo techniques in a design project framework to obtain an optimized CNS neutronic design.

1. INTRODUCTION

The design of a CNS is affected by several parameters. They are often competitive and their behaviour is not always linear or monotonic. A typical example of this is the introduction of a cavity (or a displacer) in the CNS moderator cell, just in front of the neutron beam entrance, in order to increase the cold neutron current that leaves the source in the beam tube direction. A cavity makes it possible for the neutrons coming from the proximity of the CNS moderator cell centre to enter the beam tube. These neutrons coming from the CNS centre are cooler and the neutron guide will transport them more efficiently. On the other hand, a large cavity volume reduces the volume of the

CNS moderator, i.e. reduces the CNS capability to moderate neutrons and, hence, reduces the average cold neutron flux inside the moderator cell.

The most relevant magnitudes to control during the design of a CNS are:

— Average cold neutron flux inside the CNS moderator cell;
— Neutron flux spectrum that enters into the beam tube;
— Cold neutron flux at an experimental location.

In terms of CNS utilization, it is clear that the cold neutron (CN) flux at the experimental locations is the most significant magnitude. This paper focuses on evaluating the effects that several design parameters have on this flux and the way the optimization process was carried out. To this purpose, Monte Carlo numeric techniques used in a bootstrap scheme appears to be an appropriate solution for this type of system. The MCNP code [1] is the tool used to simulate the neutron production under operating reactor core conditions, to moderate them in the cold moderator and/or reflector, and to transport them to the neutron guide entrance position, located in the beam tube.

From the guide entrance to the experimental location, the neutrons are transported through the neutron guide (typically supermirrors). The transport of neutrons along the neutron guides is simulated using alternative Monte Carlo techniques, which takes into account the reflective properties of the neutron guide as a function of neutron energy, and a comprehensive description of the geometrical conditions, such as dimensions, curvature radii and misalignment effects. A complete description of the CNS calculation line can be found in Ref. [2].

2. DESCRIPTION OF CNS OF THE RRR

The CNS is located in the heavy water reflector tank. The CNS is a liquid deuterium (LD_2) moderator, sub-cooled to ensure maximum moderation efficiency, flowing within a closed natural circulation thermosiphon loop. The LD_2 is located in the CNS moderator cell, a vessel of about 20 l, surrounded by a helium jacket. There is a gap between the moderator cell and the helium jacket to circulate helium that provides appropriate cooling. The CNS moderator cell incorporates into it a helium filled displacer or cavity.

The entire set—the moderator cell plus the helium jacket plus the displacer or cavity—is the so called ‘moderator chamber’, made completely by aluminium alloy. Figure 1 shows a schematic view of the moderator chamber.

The moderator chamber is the lower part of the thermosiphon loop. The upper part of the thermosiphon loop includes a stainless steel heat exchanger. A heavy water filled reflector plug is located between the moderator chamber and the thermosiphon heat exchanger. Its purpose is to minimize the streaming of neutrons up through the void created by the thermosiphon.

The whole thermosiphon is surrounded by a zirconium alloy vacuum containment, that provides thermal insulation and a multiple barriers safety scheme to prevent deuterium from mixing with water or air. Figure 2 shows a schematic view of the CNS thermosiphon loop.

From the calculation viewpoint, the MCNP model includes a detailed description of the core: typical 3D burn-up distribution, burnable poisons, enrichment distribution and critical control rod positions. The comprehensive description extends to the surroundings of the core where different irradiation facilities, neutron sources and neutron beams are located.

Figure 3 shows the MCNP calculation model where it is possible to see a section view of the CNS moderator chamber, displacer and beam tubes.

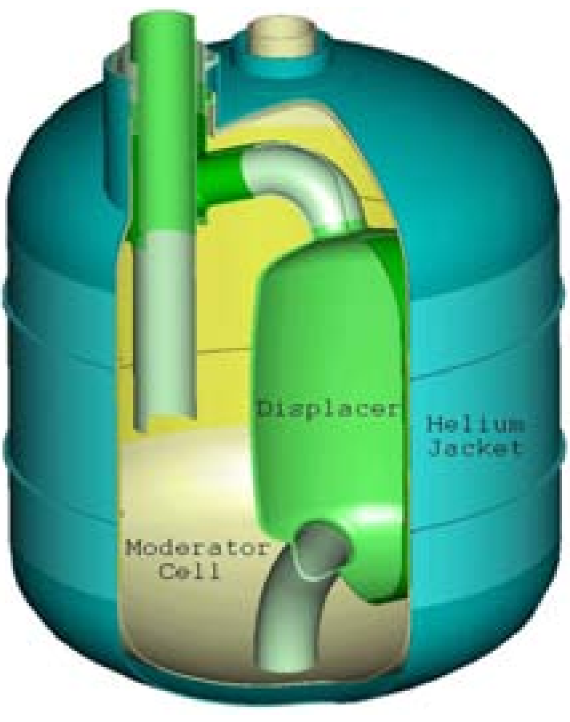


FIG. 1. Schematic view of the moderator chamber.

3. RESULTS

In order to show how the calculation line works and the consistency of the obtained results, a summary of the performed parametric analysis is given.

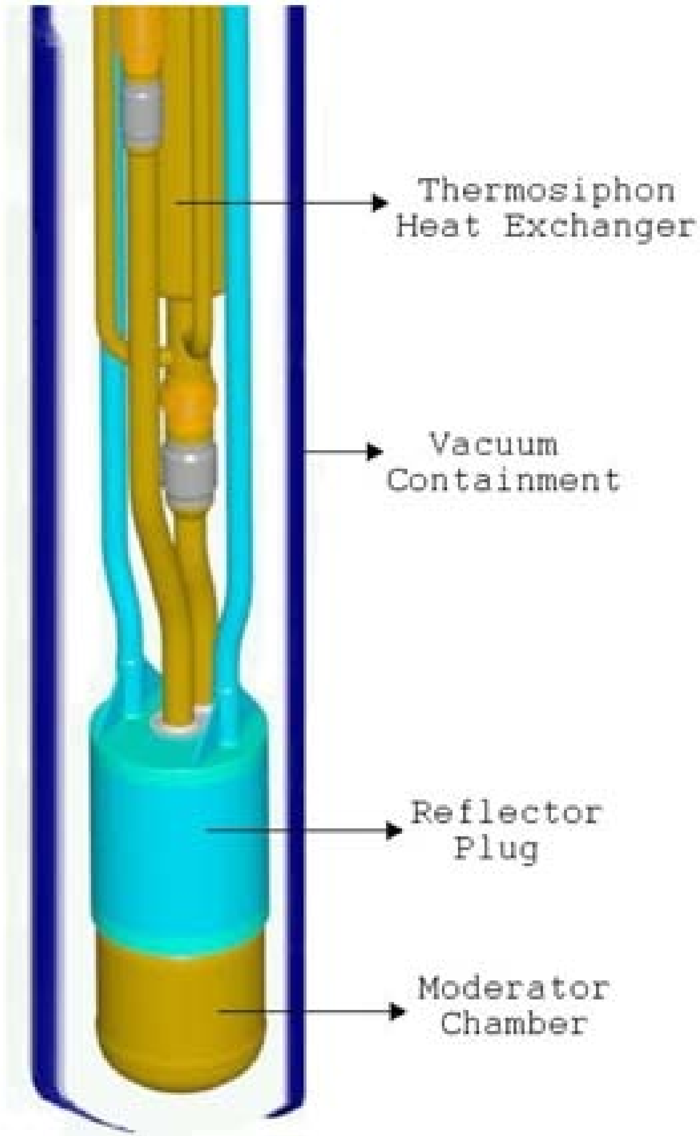


FIG. 2. Schematic view of the CNS thermosiphon loop.

The following design variables were considered:

— CNS position;
— Shape and size of the displacer section;
— Height of the displacer;
— Change of the displacer for a cavity with the same dimensions;
— Liquid deuterium impurity (H_2).

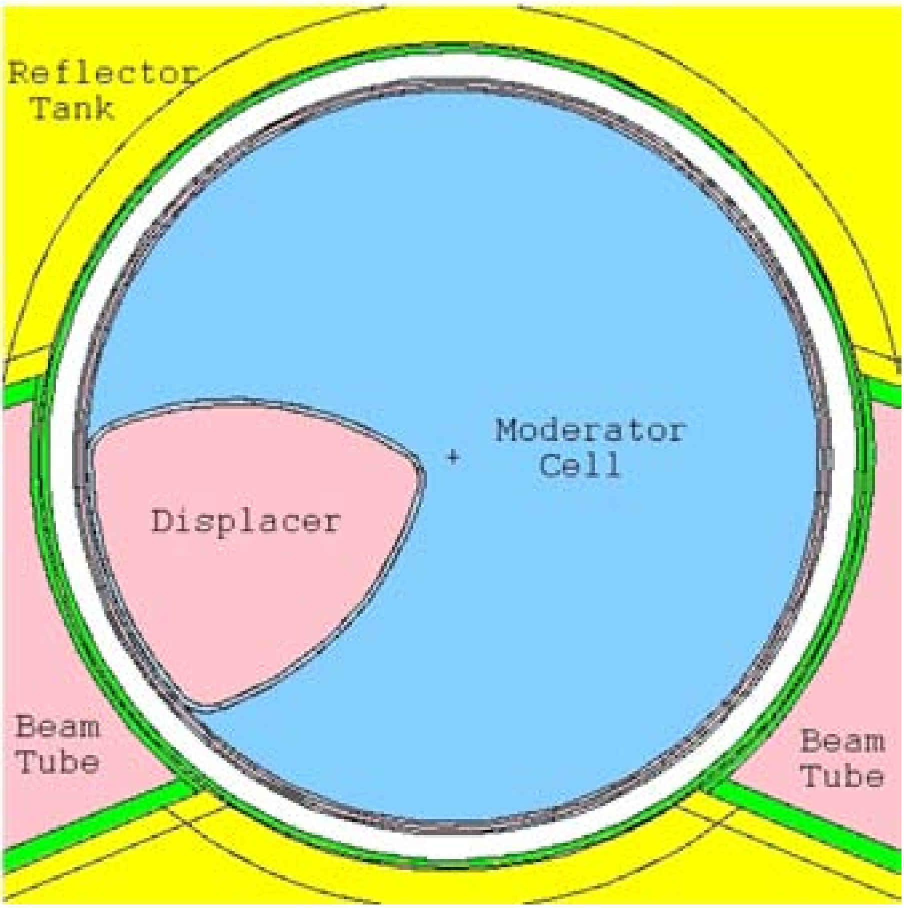


FIG. 3. Section view of the CNS calculation model.

3.1. Cold neutron source position

The CNS positioning shall be analysed in the framework to make compatible the heat load on the system and the available heat removal capability adopted by the project.

In order to determine the optimum ‘neutronic’ position of the CNS, three parameters were taken into account: the average CN flux inside the CNS cell, the CN flux at the neutron guide entrance and the CN flux at the reactor face. Figure 4 shows the behaviour of the CN flux at the guide entrance and at the reactor face, as well as the transmission factor of the neutron guide as a function of the CNS position in the reflector vessel. Additionally, the influence of the moderator’s displacer was also considered through the use of two extremely different designs for it, having 1700 and 3000 cm^3 of volume, respectively.

It was found that the behaviour of the relevant CNS performance parameters was similar for both displacer volumes, but their absolute values are different.

There is a large increment of the guide transmission when a small volume displacer is used. For this reason, with a small displacer, it is possible to obtain a higher CN flux at the reactor face despite a lower CN flux at the guide entrance.

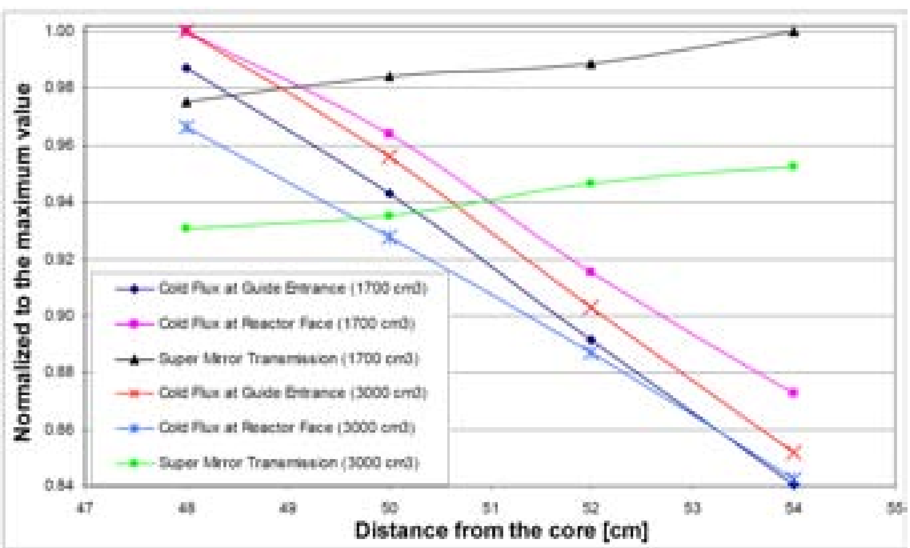


FIG. 4. CNS relevant magnitudes as a function of the core distance.

3.2. Shape and size of the displacer section

Once the CNS position is determined, different displacers were analysed; all them with the same displacer height but different cross-section area. The magnitude used to evaluate the performance of each displacer is the CN flux at the reactor face.

In order to compare different displacers, the following figures show the CN flux at the reactor face as a function of the displacer volume. Figure 5 shows two displacers of large volume. The section of one displacer is constituted by two ellipses and during the optimization process, the small radii of one ellipse was modified. An ellipse and a triangle constitute the section of the second displacer. During the optimization process, the height of the triangle was changed.

Figure 6 shows three variations on a displacer of smaller volume, based on the cylindrical-triangular shape. The first optimization step was carried out, changing the base of the triangle, while the second optimization step was done varying the height of the triangle. In the third step, concave or convex lines substituted the sides of the triangle. The highest CN flux at the reactor face was obtained with a displacer of small volume, based on the cylindrical-triangular shape.

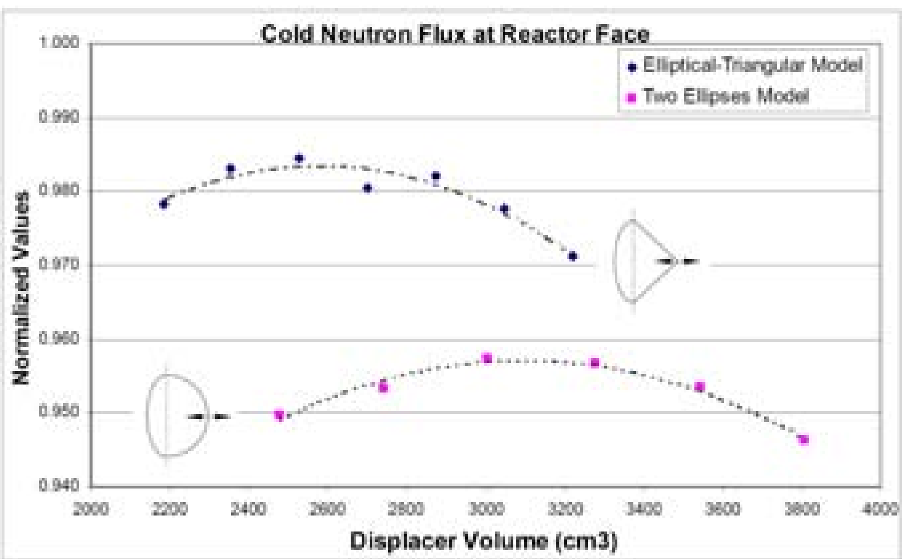


FIG. 5. Cold neutron flux at the reactor face using large volume displacers.

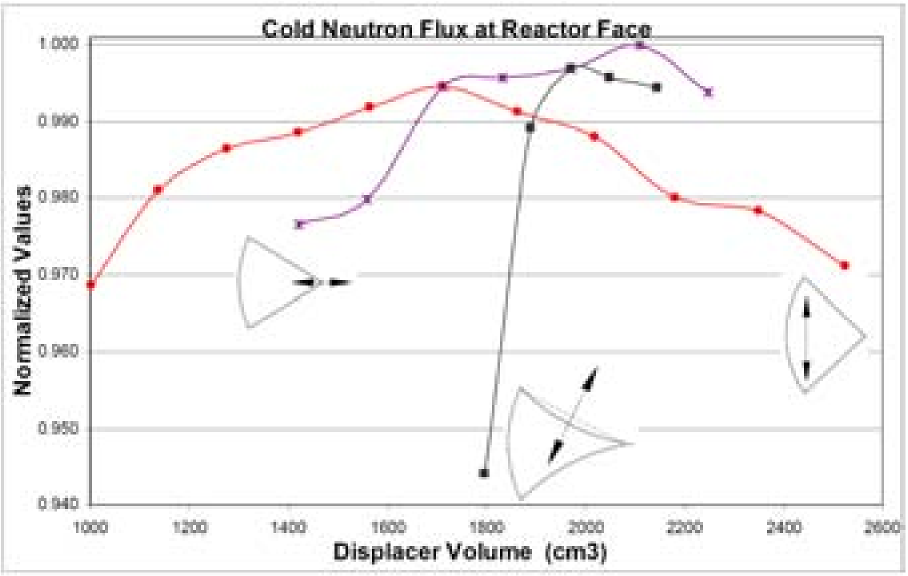


FIG. 6. CN flux at the reactor face, variations on the cylindrical-triangular displacer. Many other displacer shapes were analysed, including elliptical-trapezoidal and cylindrical-scalene triangular models.

3.3. Displacer height

The displacer height was also optimized by the evaluation of the CN flux at the reactor face. The final shapes of the top and bottom faces of the displacer were defined following mechanical and thermal-hydraulic considerations.

3.4. Change of the displacer by a cavity with the same dimensions

A cavity made in the CNS moderator cell could eliminate two extra aluminium walls and a liquid deuterium gap. These extra aluminium walls and the deuterium gap scatter very cold neutrons, producing a reduction of the effective cold neutron flux at the guide entrance position and, hence, a reduction of the neutron guide transmission factor.

A comparison between a cavity and a displacer with the same section was done. Table 1 shows the CN flux at the neutron guide entrance and at the reactor face, and the neutron guide transmission for a cavity, for a displacer and when no device is used. Values are normalized to the no-device case.

TABLE 1. COMPARISON BETWEEN A DISPLACER AND A CAVITY WITH THE SAME SECTION

Device	CN flux at the guide entrance	CN flux at the reactor face	Neutron guide transmission
No device	1.000	1.000	1.000
Displacer	1.013	1.157	1.142
Cavity	1.036	1.207	1.165

Due to the reduction on the dispersion of cold neutrons, the CN flux obtained at the reactor face when a cavity is used is 5% greater than the CN flux obtained with a displacer of the same section.

3.5. Influence of the liquid deuterium impurity (H_2)

The effect of the hydrogen impurity in the deuterium, keeping constant all other CNS geometry parameters, was also analysed. It was found that small levels of hydrogen do not reduce the flux at the reactor face. The maximum in the average CN flux inside the moderator cell is obtained with approximately 7% of hydrogen.

When hydrogen is added to the CNS moderator, the optimum geometry of the system regarding CN flux at the reactor face changes. In other words, the CN flux at the reactor face decreases despite the increment of the average CN flux because the optimum moderator cell dimensions and shapes are not optimized for such a case.

Figure 7 shows the cold flux at the guide entrance, at the reactor face and the neutron guide transmission.

3.6. Neutron temperature distribution inside the CNS cell

A meaningful indicator to visualize the neutron moderation inside the CNS cell is the neutron temperature distribution.

The neutron temperature is defined as the temperature of the Maxwellian distribution that best fits the neutron energy spectrum in the range 0–10 meV.

Figure 8 shows the neutron temperature distribution inside the moderator cell, using a displacer with a cylindrical-triangular section.

Inside the CNS cell, the neutron temperature varies from 27°K to 35°K. The loss of the moderation effect due to the displacer is clearly visualized.

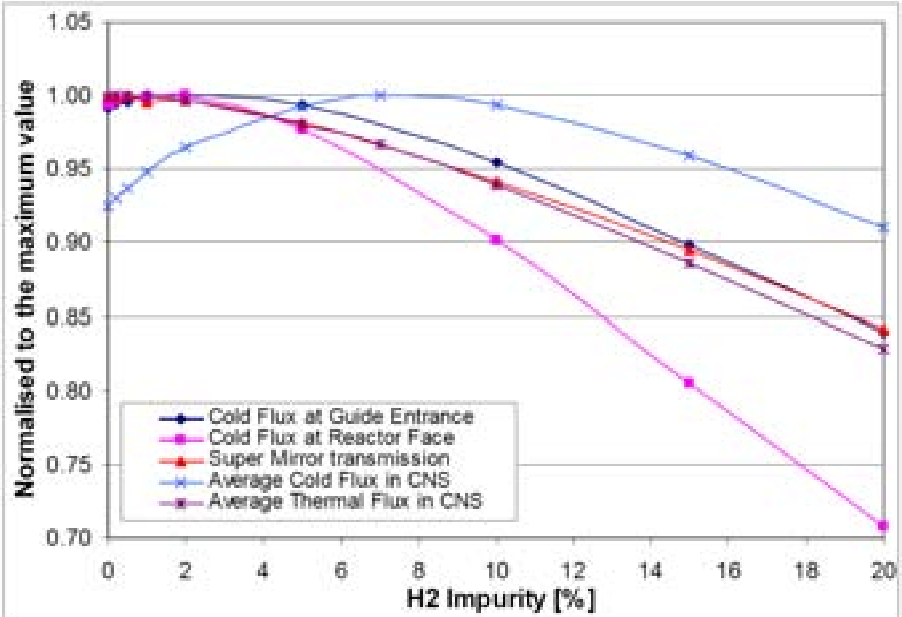


FIG. 7. CNS relevant magnitudes as a function of the H_2 impurity.

4. CONCLUSIONS

The neutronic design of a CNS involves a large number of design variables that influence the cold neutron flux at each experimental location. The design process includes some iteration with other technical areas, such as thermal-hydraulics and mechanical design.

This problem is solved using a calculation line where the MCNP is used as the main tool to transport fission neutrons. MCNP allows a comprehensive description of the operation parameters of the reactor and geometrical characteristics. The further transport through the neutron guides using the Monte Carlo technique allows the continuing of the calculation with a comprehensive description of operation parameters and geometry.

This technique allows us to analyse the influence of each design variable on the cold flux at experimental locations, by having statistical errors adequately low.

The neutron temperature distribution inside the CNS moderator cell is a useful result that allows us to visualize the neutron moderation. Also, it is

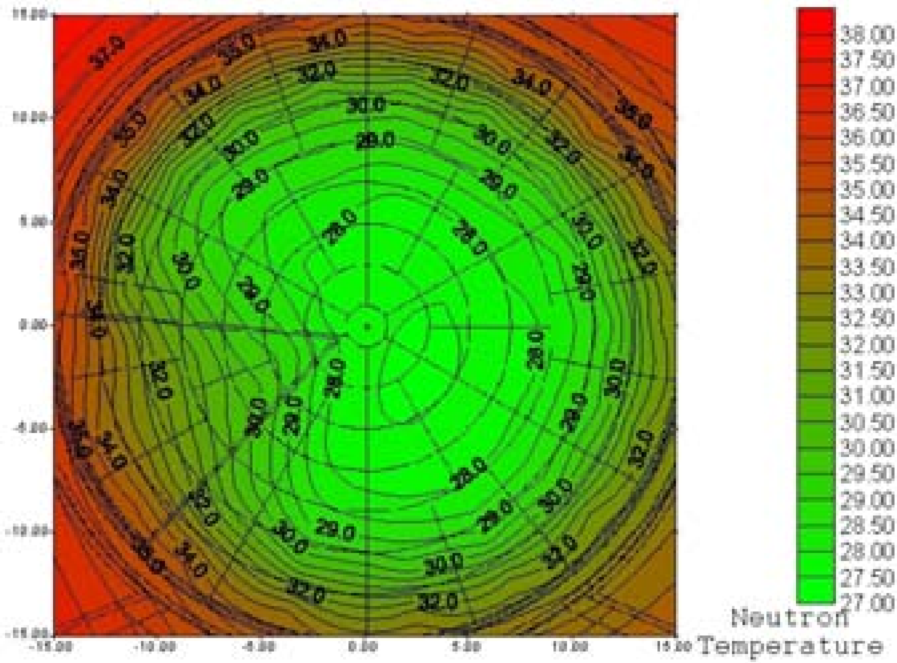


FIG. 8. Neutron temperature distribution inside the moderator cell.

possible to determine the changes on the neutron moderation and neutron temperature distribution due to the change of each design variable.

REFERENCES

[1] BRIESMEISTER, J.F. (Ed.), MCNP—A General Monte Carlo N-Particle Transport Code, Version 4C, LA13709-M, Los Alamos National Laboratory, Los Alamos, NM (2000).

[2] LECOT, C.A., HERGENREDER, D.F., LOVOTTI, O.P., "A CNS calculation line based on a Monte Carlo method", paper presented at Int. Conf. on Research Reactors (Utilization, Safety, Decommissioning, Fuel and Waste Management) Santiago, 2003.

BLANK

STATISTICAL PROCESSING OF CALCULATED REACTION RATES

Z. LAHODOVA, J. ERNEST, S. FLIBOR, M. MAREK, J. RATAJ,
L. VIERERBL
Nuclear Research Institute Řež plc,
Husinec-Rez, Czech Republic
Email: laz@ujv.cz

Abstract

The nuclear research institute (NRI) is part of the research complex situated at Rez near Prague, Czech Republic. The experimental research reactor LVR-15 offers a wide range of high quality services and the reactor is integrated into European research structures. Higher demands on irradiation conditions require—apart from other things—quick, precise and exhaustive processing of experimental data. The goal of the work is to compare the measured reaction rates with the calculated reaction rates by DORT and MCNP4C codes. The data were examined by three statistical tests: validation of the new method (calibration), comparison of two independent selections and paired comparison of selections. The paper builds on the previous comparative experiment undertaken in 2002.

1. INTRODUCTION

Higher demands on quality services and a broad range of work require more precise assignment of the irradiation conditions and more accurate processing of experimental data. Proper determination of neutron fluence is an essential assumption for the accurate evaluation of experiments. We use three computer codes (NODER, DORT, MCNP4C) and the measurement of activation monitors for the calculation of neutron fluence. Our goal is to determine the best process for the assessment of neutron fluence. The target of the experiment was to compare the results of computer codes and the measurement at usual conditions. As the first step, we directly compared neutron fluence. The results are described in Ref. [1]. Because there were some unexpected discrepancies, we decided to accomplish the second step and to compare the measured reaction rates to the calculated reaction rates. In this case, we can apply the results of codes DORT and MCNP4C only. NODER code calculates just the neutron fluence and cannot be used for calculation of the reaction rates.

2. DESCRIPTION OF THE EXPERIMENT

The experiment was designed for the standard reactor core arrangement. Three Al sticks with activation monitors were inserted into channels F6, F8 and F9, respectively. The material of the monitors was Ni (10%), Fe (100%), Ti (100%), Ag (1%) and Co (1%) where values in parentheses are concentrations of the element in the alloy.[1] The form of the monitors was a disc with a diameter of 4 mm and thickness of 0.1 mm, the mass was from 5 mg to 20 mg. The distance between sets of monitors on the stick was 5 cm. The measured range was 80 cm. The irradiation time was 8 h and the mean reactor power was 8.4 MW.

3. COMPUTER CODES AND MEASUREMENT

The measured reaction rates of the reactions $^{54}Fe(n,p)^{54}Mn$, $^{58}Fe(n,\gamma)^{59}Fe$, $^{58}Ni(n,p)^{58}Co$, $^{59}Co(n,\gamma)^{60}Co$ were compared with the calculated reaction rates by DORT and MCNP4C codes. The examples of $^{54}Fe(n,p)^{54}Mn$ and $^{58}Ni(n,p)^{58}Co$ reaction rates in channel F6 are shown in Figs 1(a) and 1(b). The x coordinates correspond to the positions of monitors on the stick and also to the vertical axis at the reactor core. The value 0 represents the centre of the reactor core. The reaction rates (RR) are displayed as y coordinates. Green

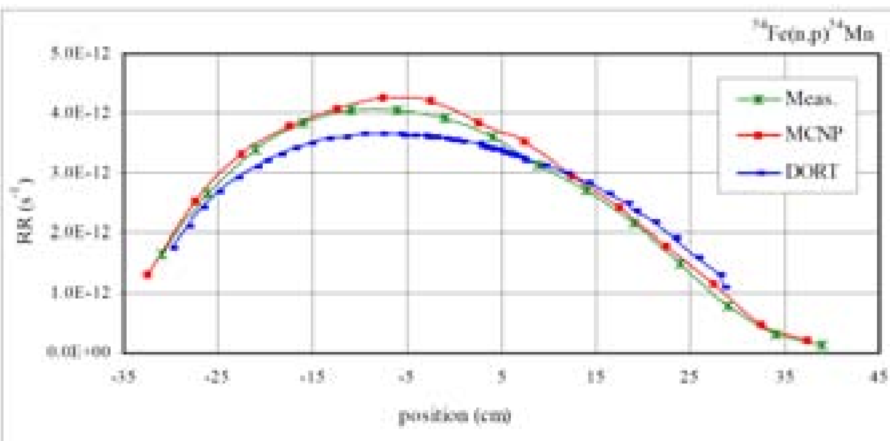


FIG. 1(a). Reaction rates of $^{54}Fe(n,p)^{54}Mn$ reaction for stick in channel F6.

[1] The monitors of Ti and Ag were not used for the reaction rate comparison but only for evaluation of neutron fluence.

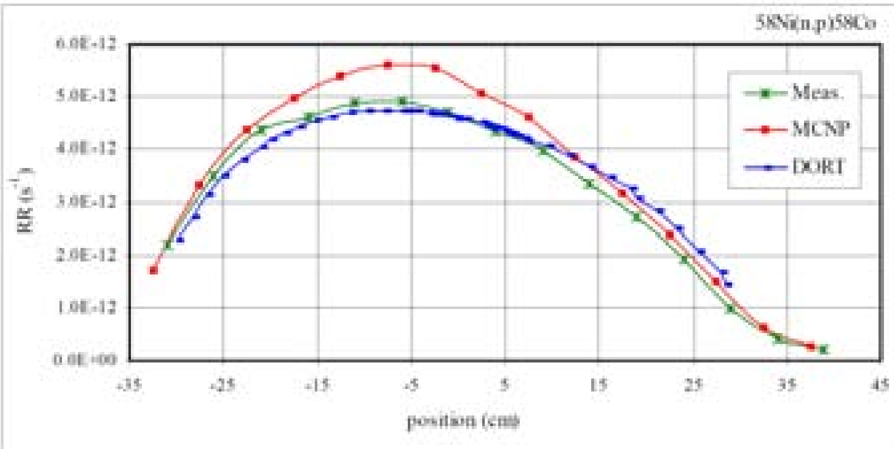


FIG. 1(b). Reaction rates of ^{58}Ni(n,p)^{58}Co reaction for stick in channel F6.

stars correspond to the measured values of the reaction rates, red boxes to the calculations by MCNP code and blue hyphens to the calculations by DORT code.

3.1. Measurement

The weighing of the activation monitors is done with Mettler Toledo balance type AT261 with the resolution of 0.01 mg. The typical mass of the monitor is about 10 mg. The value of the measured mass is automatically stored.

The gamma activities are measured by spectrometric assembly (Canberra) with the HPGe detector. The spectrometric assembly has the relative efficiency of 18% and FWHM=1.8 keV for the energy of 1332 keV. The detector is placed in a shielding box with 5 cm thick lead walls. The calibration of the detector and the method of the measurement are in accordance with ASTM E 181–82 [2]. The detector is equipped with an automatic sample changer.

The reaction rates are evaluated by the custom program Protokol (written in Visual Basic for Applications). All the measured data necessary for calculations: the mass of monitors, the activities and their uncertainties, the time dependence of power during irradiation, are stored and loaded automatically, which minimizes rough human mistakes.

3.2. DORT code

The reaction rates were calculated by the transport programs ANISN and DORT in $S_{10}P_3$ approximation. Both programs solve the Boltzmann transport equation converted to the system of differential equations. Every fuel element in the reactor core has different axial neutron fission density distribution and the DORT code uses the mean values. These mean values are weighted averages of distributions in surrounding fuel elements. For example, channels F5, F7, E6 and G6 were used for the calculation in channel F6.

The three-dimensional neutron fluxes were substituted by the three-dimensional fluxes synthesis according to the following formula:

$$\Phi(X,Y,Z) = \Phi(X,Y)/\ \Phi(X)\times\ \Phi(X,Z)$$

where

$\Phi(X,Y)$ and $\Phi(X,Z)$ are two-dimensional neutron flux distributions in the horizontal and vertical direction, respectively,

$\Phi(X)$ is one-dimensional distribution.

The multigroup data library BUGLE96 based on ENDF/BVI was used as the elementary library for both DORT and ANISN programs.

3.3. MCNP4C code

The MCNP4C with the DLC-200 cross-section library is a radiation transport code based on the general Monte Carlo method. The Monte Carlo code requires more computer time than the other methods to obtain comparable accuracy. The number of histories (nps) used for the calculation changes with the distance of the element from the core centre. Therefore, $4{\cdot}10^8$ histories were used for F6 calculation, $6{\cdot}10^8$ for F8 and $7{\cdot}10^8$ for F9. The average statistical error (1) was less than 5%.

4. STATISTICAL PROCESSING

We assume that the measured reaction rates are independent and true variables with the accuracy given by the accuracy of the spectrometric assembly. Three statistical analyses were used for data processing: calibration, comparison of two independent selections and paired comparison of selections.[2] We also completed an exploration analysis, approximations, finding of outliers, etc. for all mentioned reactions and for all three sticks. These analyses were accomplished by QC.Expert statistical software [3]. For better orientation and illustration, only the results for the $^{54}Fe(n,p)^{54}Mn$ reaction in channel F6 are presented in this article. The other results are discussed in the concluding chapter.

4.1. Calibration

Calibration can be used for the validation of a new method. The 'tried and true' variables (the measured reaction rates) replace the independent variables and the values of the new method (the calculated reaction rates) replace the dependent variables. The number of variables and their sequence has to be maintained. The new method is valid if the resulting calculated model is linear with unitary slope and if the intercept is unbiased. The linear model has N-2 degrees of freedom, where N is the number of variables.

The graph of the calibration linear model is in Fig. 2. The red lines surround the confident bound. The hypothesis of validation can be accepted because the value of the intercept is 0.017 and the confident interval is (–0.125, 0.158). The number zero belongs in this interval. The second condition confirms the unitary slope. The value of the slope is 1.036 and the confident interval (0.986, 1.085).

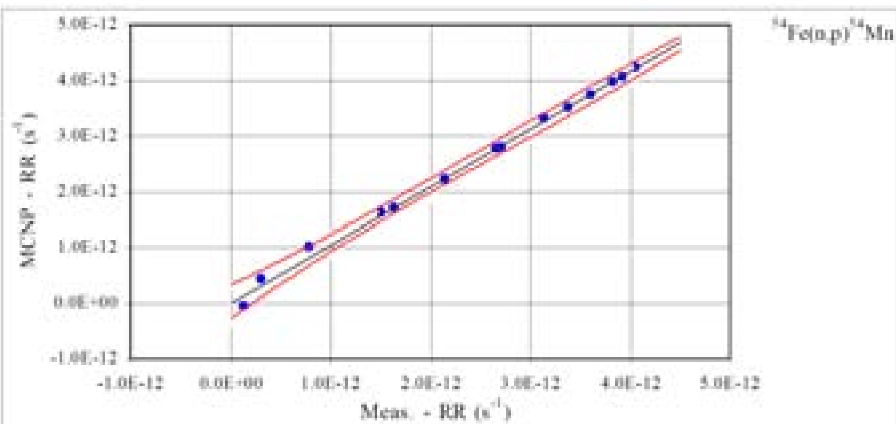


FIG. 2. Calibration—linear model.

[2] The level of significance (alpha) was set to 0.05 for all the hypotheses.

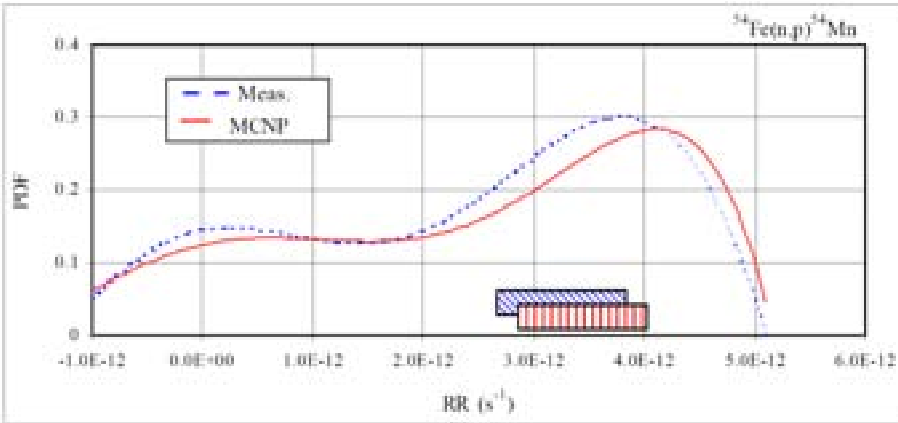


FIG. 3. Density trace.

4.2. Comparison of two independent selections

The two-sample test is used for the evaluation of differences between two samples or distributions. The assumptions for this test are the two independent random samples with the continuous distributions. The selections can have different dimensions. The sequence of elements is arbitrary.

Figure 3 represents the estimation of the density trace. The histogram with the density trace overlay allows the study of the distributional features of the two samples to determine if two-sample tests are appropriate. Density refers to the relative frequency (concentration) of data points along the data range. Mathematically, the density at a value x is defined as the fraction of data values per unit of measurement that lie in an interval centred at x. The interval width increases the smoothness of the chart. The shaded boxes symbolize the confident intervals for averages.

The box graph in Fig. 4 is useful for displaying the mean and spread of a set of data. Several box plots may be displayed side by side to compare the average and spread of several selections. The white, centric rectangle represents 50% of data. The correlative coefficient is 0.998 and the difference rate is 1.076, which means that the hypotheses of equal means, dispersions and distributions can be accepted.

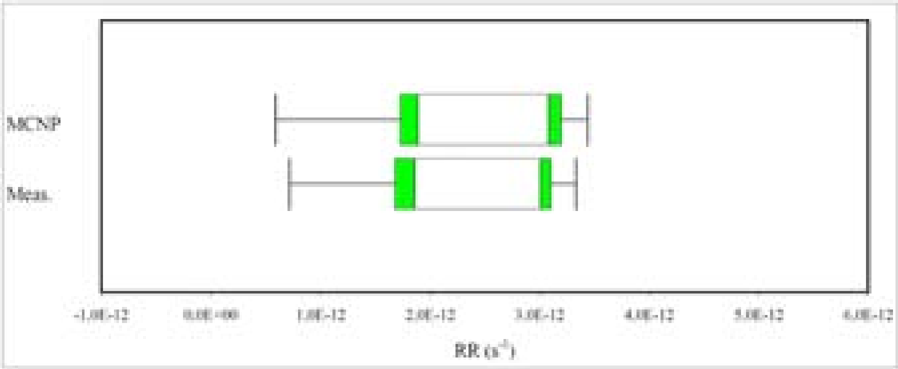


FIG. 4. Box graph.

4.3. Paired comparison of selections

This is a test of a hypothesis that the mean and the standard deviation for difference between two samples are unbiased and a hypothesis that two selections have the same distribution.

The dispersion is shown in Fig. 5. The red dashed line represents the unbiased dispersion. The black line denotes how closely the data fall about these expected values.

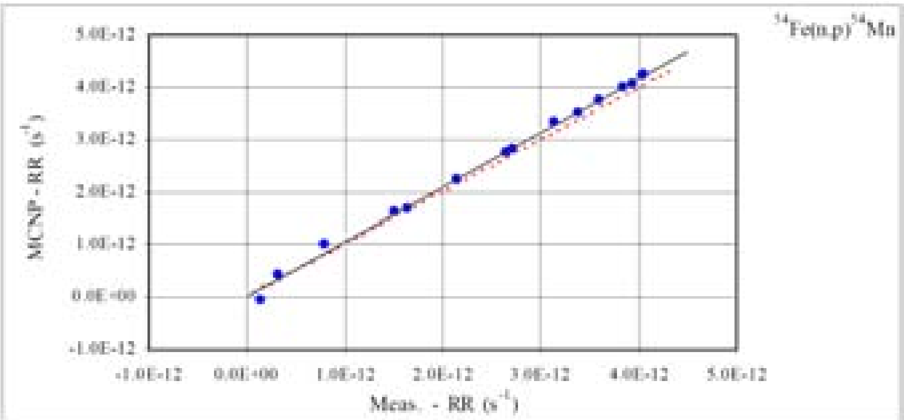


FIG. 5. Graph of dispersion.

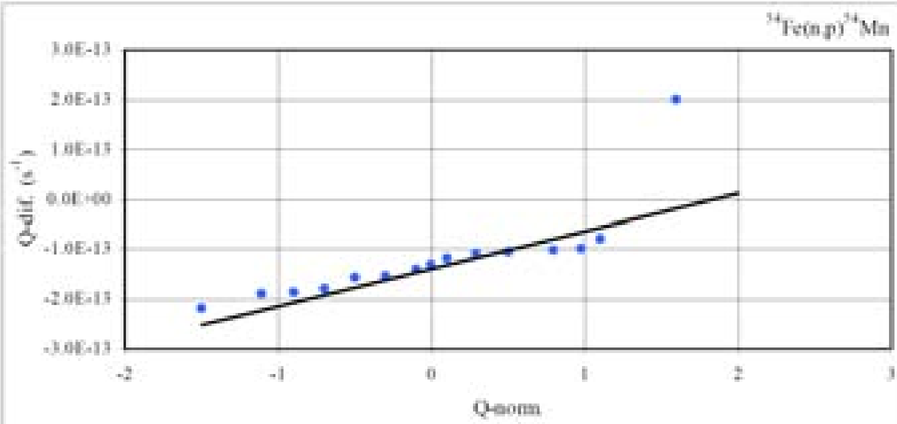


FIG. 6. Q-Q graph.

Q-Q graph in Fig. 6 allows to assess whether the difference between selections has normal distribution. Q-Q graph compares distribution with the theoretical normal distribution.[3] If the points fall along a straight line, we can assume that the distribution is normal. As the average of differences is 0.07, and the confident interval is (–0.09, 0.23), the hypothesis for the paired equality can be accepted. There is an outlier in the upper right corner that relates to the last point in the sequence.

5. CONCLUSIONS

Almost all tests for ^{54}Fe(n,p)^{54}Mn and ^{58}Ni(n,p)^{58}Co reactions for MCNP4C and DORT calculations were successful, so we can say that the differences between measured and calculated reaction rates are statistically unbiased. Therefore, it can be stated that the calculations by MCNP4C and DORT codes give results that correlate with the experimental values. On the other hand, the majority of suggested hypotheses for ^{58}Fe(n,γ)^{59}Fe, ^{59}Co(n,γ)^{60}Co were rejected, thus, there are some significant discrepancies between the measurement and calculations. The differences between the measured and the calculated reaction rates by MCNP4C were lesser than differences between the measurement and calculations by DORT. But generally, where MCNP4C gives correct values, the DORT gives correct values,

[3] The theoretical distribution can be normal, exponential, etc.

too, and vice versa where DORT results were rejected, the results for MCNP4C were rejected too, even though some hypothesis for MCNP4C and for weaker levels of significance 0.01 were accepted.

The results are very similar for all three channels. We assume that the position of the samples in the reactor core is statistically unbiased and does not have an influence on the results. We will continue this experiment because we want to find the sources of discrepancies between the methods described. Our goal is to determine the further advance to assess and process the results of the experiments at the LVR-15.

REFERENCES

[1] LAHODOVÁ, Z., et al., "Comparison of different neutron distribution processing types", XV-ICPRP, Int. Mtg, Alushta (2002).

[2] ASTM Designation E 181–82: Standard General Methods for Detector Calibration and Analysis of Radionuclides.

[3] QC.Expert 2.5, Statistical software, TriloByte Ltd.

[4] ANITA: Analysis of Neutron Induced Transmutation and Activation, EUR 12622, NEA Data Bank, (1998).

[5] BRIESMEISTR, J.F., MCNP-A General Monte Carlo N-Particle Transport Code (Version 4C), Los Alamos National Laboratory, Los Alamos, NM (2000).

[6] ERNEST, J., Calculation of power distribution in active reactor core LVR-15 for 39 campaign, (2002).

[7] MELOUN M., MILITKÝ, J., Statistical Processing of Experimental Data, East Publishing, a.s. (1998).

[8] RHOADES, W.A., CHILDS, R.L., The DORT/PC Two-Dimensional Discrete Ordinate Transport Code System, CCC - 532, RSIC (1991).

[9] VIERERBL, L., MAREK, M., TOMASEK, F., "Neutron fluence and spectrum measurement at the reactor LVR-15", ANS Transactions, Vol. 81, (1999) 285–287.

[10] VIERERBL, L., MAREK, M., ERNEST, J., FLÍBOR, S., RATAJ, J., Thermal neutron distribution measurement at core of the LVR-15 reactor, PHYSOR 2000, ANS Int. Topical Mtg (2000).

[11] DLC - 185/BUGLE96, Broad Group Neutron/Photon Cross Section Library, RSICC - ORNL (1996).

BLANK

THERMAL NEUTRON SELF-SHIELDING FACTOR IN FOILS
A universal curve

I.F. GONÇALVES, E. MARTINHO, J. SALGADO
Instituto Tecnológico e Nuclear,
Sacavém, Portugal
Email: ifg@itn.mces.pt

Abstract

The presence of a sample in the neutron field of a nuclear reactor createsa perturbation of the local neutron fluxes. In general, the interpretation of the sample activation due to thermal and epithermal neutrons requires the knowledge of two parameters: the thermal neutron self-shielding factor, G_{th}, and the resonance neutron self-shielding factor, G_{res}. In recent papers, the authors established a universal curve of G_{res} for isolated resonances and various geometries. The paper deals with the description of G_{th} in foils by means of a universal curve on the basis of a dimensionless variable which includes the physical, nuclear and geometrical properties of the sample. The universal curve is in very good agreement with experimental and calculated values obtained from the literature. The study of other geometries (spheres, wires and cylinders) is in progress.

1. INTRODUCTION

As is well known, the irradiation of a sample in the neutron field of a nuclear reactor is affected by the local perturbation of the neutron fluxes produced by the sample [1]. In general, the interpretation of the sample activation due to thermal and epithermal neutrons requires the knowledge of two parameters: the thermal neutron self-shielding factor, G_{th}, and the resonance neutron self-shielding factor, G_{res}. In recent papers [2–4], the authors established a universal curve of G_{res} for isolated resonances or groups of isolated resonances, and various geometries of the samples (foils, wires, spheres and cylinders).

In relation to G_{th}, some experimental [5–13] and theoretical studies [14] have been carried out to determine this factor for foils and wires of different elements. Their results are presented as tables or graphics of G_{th} for a given element and geometry as a function of the typical dimension. Recently, Copley [15] studied the scattering effect within an absorbing sphere and, according to his results, the self-shielding factor is obtained through a set of curves as a function of the macroscopic absorption and scattering cross-sections of the sample. This paper deals with the description of the thermal neutron self-shielding factor in foils by means of a universal curve. The study of other geometries (spheres, wires and cylinders) is in progress.

2. CALCULATION

For a given element, geometry and sample dimension, G_{th} is calculated as the ratio between the reaction rates per atom in the real sample and in a similar and infinitely diluted sample:

$$G_{th}(x) = \frac{\int_{E_1}^{E_2} M(E)\ \sigma_a(E)\ dE}{\int_{E_1}^{E_2} M_0(E)\ \sigma_a(E)\ dE} \tag{1}$$

where x is the typical sample dimension, $M_0(E)$ is the non-perturbed thermal (maxwellian) neutron flux per unit energy interval (inside the infinitely diluted sample), $M(E)$ is the perturbed thermal neutron flux inside the real sample, $\sigma_a(E)$ designates the (n,γ) cross-section, and E_1 and E_2 are, respectively, the lower and the upper limits of the thermal neutron spectrum at room temperature. The total neutron cross-section is adopted in the calculation of the perturbed neutron flux, thus taking into account the neutron scattering in the sample. In the calculations, the density for infinite dilution is assumed to be $\rho = 10^{-6}\ \rho_0$, ρ_0 representing the density of the real sample. Thermal neutron self-shielding factors in foils of very different elements (see Table 1) have been calculated using the MCNP code [16].

σ_t, σ_s, σ_a, are, respectively, the total, scattering and absorption microscopic cross-sections averaged over the thermal neutron spectrum at room temperature [17].

In order to eliminate practically the effect of the neutrons entering through the foil edge, a ratio of $R/t \geq 100$ (with R ≥ 1 cm) has been adopted, t and R being the thickness and the radius of the foil, respectively.

TABLE 1. PHYSICAL AND NUCLEAR PROPERTIES OF THE STUDIED ELEMENTS [17, 18]

Element	A ($g\ mol^{-1}$)	ρ ($g\ cm^{-3}$)	σ_t (b)	σ_s (b)	σ_a (b)
Al	26.98	2.7	1.617	1.414	0.203
Au	196.97	19.3	95.05	6.86	88.19
Cd	112.41	8.65	2996.5	11.54	2985.0
Co	58.93	8.9	38.98	6.02	32.96
Cu	63.55	8.96	11.23	7.87	3.36
Eu	151.96	5.24	3655.6	6.61	3649.0
Gd	157.25	7.9	36892.3	138.7	36753.6
In	114.82	7.31	176.8	2.55	174.3
Ir	192.22	22.42	384.2	14.7	369.5
Mo	95.94	10.22	7.77	5.48	2.29
Ni	58.69	8.9	22.58	18.69	3.89
Pb	207.20	11.35	11.22	11.07	0.15
Pt	195.08	21.45	21.5	12.4	9.1
Rh	102.91	12.41	136.2	3.25	132.9
Sc	44.96	2.99	46.54	22.48	24.06
Sm	150.36	7.52	8419.3	52.3	8367.0
Ta	180.95	16.65	24.02	5.64	18.38
Fe	55.85	7.86	13.93	11.48	2.45

3. RESULTS AND DISCUSSION

As is shown in Fig. 1, the value of G_{th} depends on the physical and nuclear properties of the material, as well as on the foil thickness.

However, it is possible to introduce a dimensionless variable, z, which converts that dependence into a unique curve. The analysis of the results obtained in the foil (or slab) geometry has shown that this variable is given by:

$$z = t\ \Sigma_t \left(1 - \frac{\Sigma_s}{\Sigma_t}\right)^{0.85} = t\ \Sigma_t \left(\frac{\Sigma_a}{\Sigma_t}\right)^{0.85} \qquad (2)$$

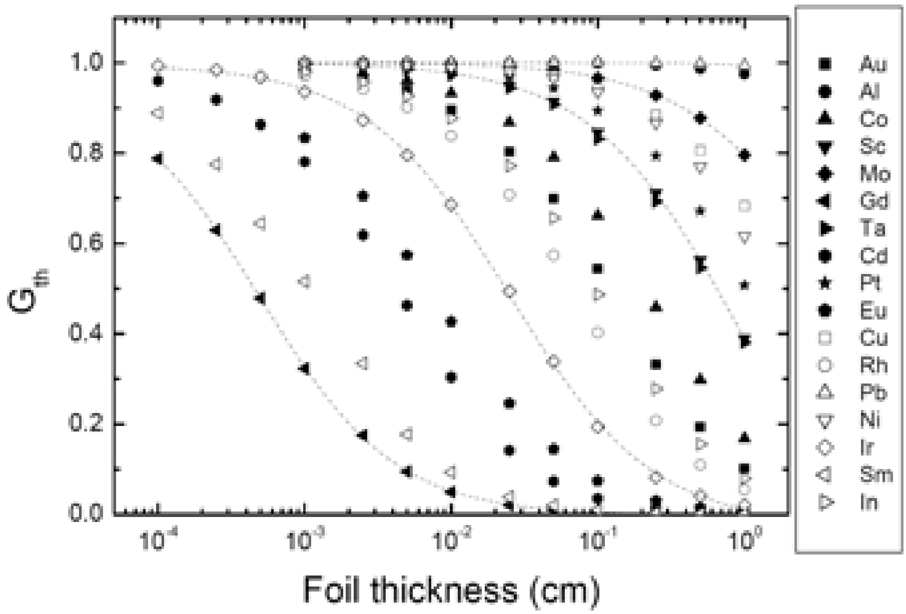


FIG. 1. Calculated G_{th} as a function of the foil thickness.

where Σ_t, Σ_s and Σ_a are, respectively, the total, scattering and absorption macroscopic cross-sections averaged over the thermal neutron spectrum, and t is the thickness of the foil. Note that the variable z takes into account the neutron scattering in the sample.

The analysis of the results also shows that a sigmoid is the best curve to be fitted to the calculated $G_{th}(z)$ values. The expression of this curve is:

$$G_{th}(z) = \frac{A_1 - A_2}{1 + \left(\frac{z}{z_0}\right)^p} + A_2 \tag{3}$$

where A_1, A_2, z_0 and p are the curve parameters, to be adjusted to the calculated values. Note that A_1 is the limit of G_{th} as z tends to zero; A_2 is the limit of G_{th} as z tends to infinity; z_0 is the inflexion point $[G_{th}(z_0) = (A_1+A_2)/2]$; and p is related with the gradient of the curve at $z = z_0$.

Figure 2 shows the calculated values of $G_{th}(z)$ and the sigmoid (universal curve) adjusted to all the points. The values of the curve parameters are:

$$A_1 = 1: \quad A_2 = 0; \quad z_0 = 0.608 \pm 0.006; \quad p = 0.968 \pm 0.009 \tag{4}$$

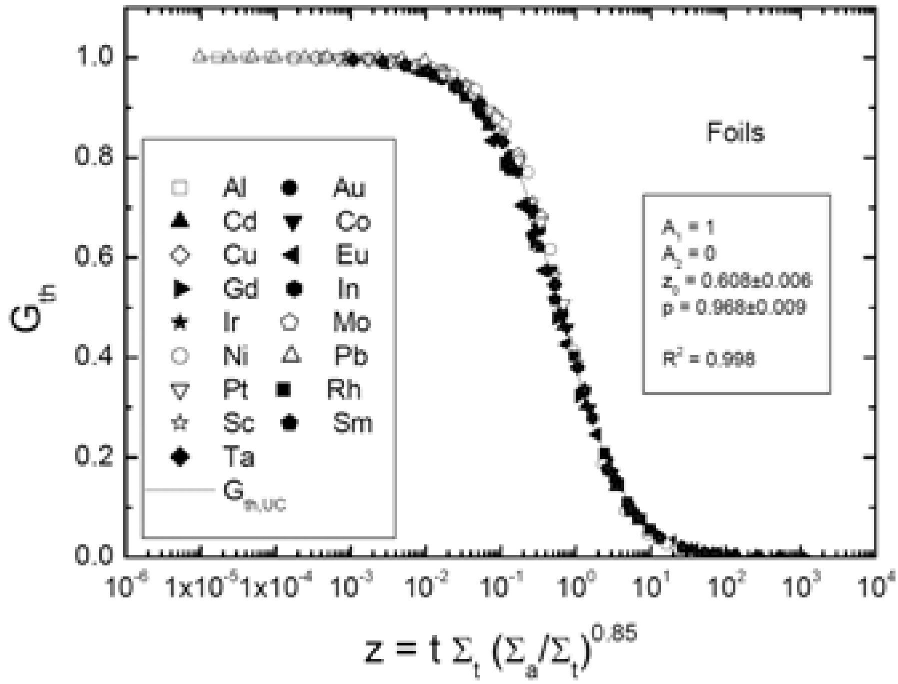


FIG. 2. Calculated G_{th} and the universal curve ($G_{th,UC}$)

For more than 90% of the results, the relative deviation between the universal curve and the calculated values is less than 4% for z<1 (z domain of practical interest).

In Fig. 3, the universal curve is compared with experimental and calculated values of G_{th} from the literature. As can be observed, the overall agreement is very good, thus confirming the validity of the relations (2) to (4) to determine the thermal neutron self-shielding factor in foils. The maximum relative deviation between the universal curve and the experimental values is less than 5%.

4. CONCLUSION

In spite of large differences between the physical and nuclear properties of the studied elements, a universal curve can describe the behaviour of the thermal neutron self-shielding factor for foil samples. In this curve, G_{th} is expressed as a function of a dimensionless variable, which takes into account the physical and nuclear properties of the sample. The universal curve is in very

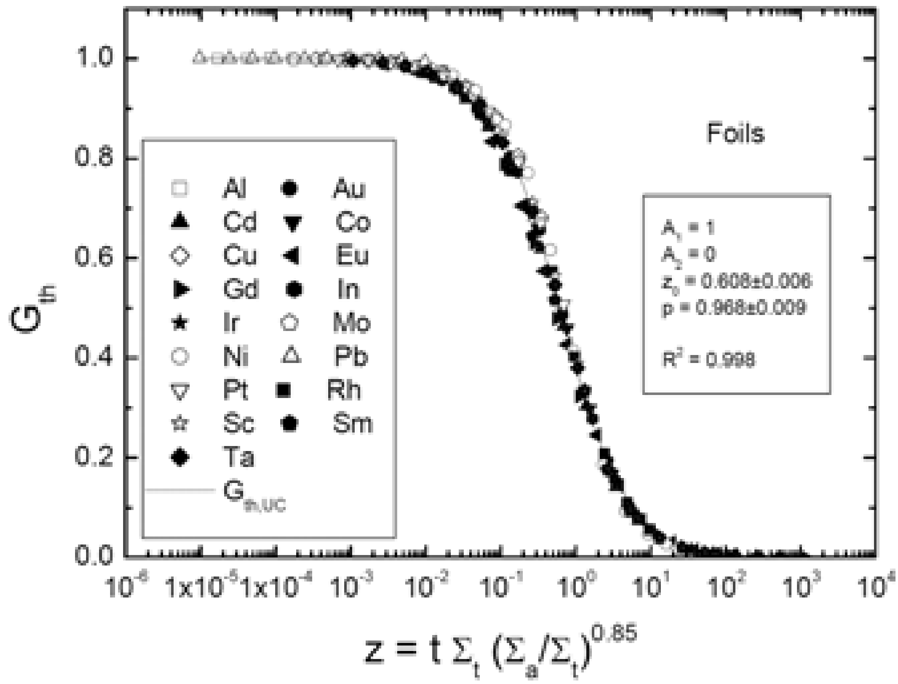


FIG. 3. Comparison of the universal curve ($G_{th,UC}$) with experimental and calculated values from the literature.

good agreement with experimental and calculated values obtained from the literature. The maximum relative deviation between the universal curve and the experimental values is less than 5%.

REFERENCES

[1] BECKURTS, K.H., WIRTS, K., Neutron Physics, Springer-Verlag, London (1964).

[2] MARTINHO, E., GONÇALVES, I.F., SALGADO, J., Universal curve of epithermal neutron resonance self-shielding factors in foils, wires and spheres, Appl. Radiat. Isot. **58** 3 (2003) 371–375.

[3] GONÇALVES, I.F., MARTINHO, E., SALGADO, J., Extension to cylindrical samples of the universal curve of resonance neutron self-shielding factors, Nucl. Instrum. Methods (in press).

[4] SALGADO, J., MARTINHO, E., GONÇALVES, I.F., Methodology for the calculation of neutron self-shielding factor of a group of isolated resonances, J. Radioanal. Nucl. Chem. (in press).

[5] KLEMA, E.D., RITCHIE, R.H., Thermal neutron flux measurements in grafite using gold and indium foils, Phys. Rev. **87** (1952) 167.
[6] SOLA, A., Flux perturbation by detectors foils, Nucleonics **18** (3) (1960) 78.
[7] HASNAIN, S.A., MUSTAFA, T., BLOSSER, T.V., Thermal neutron density perturbations by foils in water, ORNL 3193 (1961).
[8] EASTWOOD, T.A., WERNER, R.D., Resonance and thermal neutron self-shielding in cobalt foils and wires, Nucl. Sci. Eng. **13** (1962) 385.
[9] SPERNOL, A., VANNINBROUK, A., GROSSE, G., Thermal flux perturbation by cobalt detectors, Neutron Dosimetry (Proc. Symp.), Harwell, Vol. 1 (1962) 547.
[10] WALKER, J.V., RANDALL, J.D., STINSON JR, R.C., Thermal neutron flux perturbation due to indium foils in water, Nucl. Sci. Eng. **15** (1963) 309.
[11] CRANE, J.L., DOERNER, R.C., Thermal self-shielding and edge effects in absorbing foils, Nucl. Sci. Eng. **16** (1963) 259–262.
[12] TAYLOR, N.K., LINACRE, J.K., The use of cobalt as an accurate thermal neutron flux monitor, AERE-R-4111 (1964).
[13] CARRÉ, J.C., ROULLIER, F., VIDAL, R., Études des coefficients d'autoabsorption thermique, CEA – Département des Études de Piles, Service des Expériences Critiques, Rapport MIN 76 (1965).
[14] LOPES, M.C., Sensitivity of self-powered neutron detectors to thermal and epithermal neutrons with multiple collision treatment, PhD thesis, University of Coimbra (1991).
[15] COPLEY, J.R.D., Scattering effects within an absorbing sphere immersed in a field of neutrons, Nucl. Instrum. Methods **A307** (1991) 389–397.
[16] BRIESMEISTER, J. (Ed.), MCNP Code, Los Alamos National Laboratory, Report LA-13709, Los Alamos, NM (2000).
[17] http://www2.bnl.gov/ton/
[18] TULI, J.K., "Nuclear wallet cards", National Nuclear Data Center, Brookhaven National Laboratory, Upton, OH (2000).

BLANK

A SET OF PROGRAMS FOR EXPERIMENTAL DATA PROCESSING OF FOCUSING CONFIGURATIONS

I. IONITA
Institute for Nuclear Research,
Pitesti, Romania
Email: ionionita@lycos.com

N.C. POPA
Joint Institute for Nuclear Research,
Dubna, Russian Federation

Abstract

Four numerical programs are presented for experimental data processing of focusing configuration of a high-resolution crystal neutron diffractometer. The corresponding most relevant features of these programs (DAX, Rietveld, Monte Carlo, LSQ1) are given.

1. INTRODUCTION

A focusing high-resolution crystal neutron diffractometer was recently brought into operation. The main features of this configuration are presented in Refs [1–5]. To design this experimental facility, the numerical program DAX has been extensively used. This program computes the resolution function for a given experimental configuration. This information could also be very useful in the Rietveld experimental data processing.

Taking account of the peculiarities of the experimental data processing for the case of the focusing configurations, three numerical programs were realized to solve these specific requirements. These programs are also presented here.

2. DAX PROGRAM

The DAX program [4] allows for the resolution function, line widths and intensities computations for a two-axis neutron spectrometer; the optimum values for some relevant parameters are computed, too.

Configurations with one or two crystals monochromator group, with plane (the conventional case) or curved crystals (the focusing case) with or without Soller collimators, diaphragms or neutron guide are allowed for. To select the desired situation it is necessary only to select the corresponding index according to the 'how to use' indications.

The computational method used is the matrix one [6, 9]. To reduce the involved matrix dimensions, the horizontal and vertical plane computations are separated and put together at the end. This is possible owing to the lack of correlation between the corresponding variables.

The experimental configuration optimization can be made either analytically, using the corresponding optimizations conditions, or numerically, by minimizing the optimization parameter $W^2_{1/2}/I_{max}$ ($W_{1/2}$ is the full line width and I_{max} is the maximum peak intensity); for the numerical optimization, the analytical condition is the '0'-th order approximation. There are 12 parameters for which the numerical optimization is possible to be made, the radius (ii) of curvature, the cutting angle, the mosaic spread, the crystal thickness, the sample position or thickness, the relevant distances between diffractometer components, the angular spreads (for the conventional configurations) and the detector slit width. Though all the 12 (14 for two crystals) parameters can be optimized together, this is not desirable because in this situation the computing time would be too large; for a reasonably long computing time, the number of parameters optimized at the same time would not exceed six.

A library with the relevant data concerning the most used crystals is included in the program. The resolution matrix and its inverse are computed. Also given are the necessary data (the axes lengths) for visualization of the resolution ellipsoid. The sample and the detector intensity are computed, too, together with the effective sample volume and surface. The line widths are computed for rocking, θ-2θ and detector scan (for a monocrystal sample) or for polycristal sample.

3. THE FOCUSING CONFIGURATIONS RIETVELD PROGRAM

A Rietveld refinement program was written to process the pattern measured on this kind of diffractometer. Three peculiarities were taken into account in this program. Firstly, the irradiated sample volume is dependent on the scattering angle. Secondly, like for any focusing geometry, there is a significant Bragg angle dependent shift of the diffraction peaks. At last, the most used models for the sample texture in the existing Rietveld programs are not working in this case. The March model, for example, works when the sample has a cylindrical symmetry along either the diffraction vector or perpendicular to the diffraction plan. This condition is not fulfilled by the present geometry, the diffraction vector changing the direction from one Bragg peak to another.

The diffraction pattern in our Rietveld program is fitted by the following formula:

$$I(2\theta_s) = B(2\theta_s) + sv_c^{-2}\sum_H (VT)_H T_H n_H d_H^3 \tan\theta_H \,|\,F_H\,|^2\, w_H^{-1} \exp\left[-\left(2\theta_s - 2\theta_H - \Delta 2\theta_H\right)^2 / 2w_H^2\right] \quad (1)$$

Here $2\theta_s$ is the current scattering angle, $B(2\theta_s)$ is the background, s is a scale factor, v_c is the unit cell volume, n_H is the multiplicity factor of the diffraction peak corresponding to the reciprocal lattice vector **H**, interplanar distance d_H, Bragg angle θ_H and having the structure factor F_H. By w_H^2 we have denoted the dispersion of the Gaussian peak profile for which we are using the Caglioti formula. The quantity $(VT)_H$ is the product between the fraction of irradiated sample volume and the sample transmission. It is calculated by numerical integration for every peak, by taking into account the intensity variation across the incident beam on the sample. This variation was measured and approximated by an empirical formula. Finally, by T_H and $\Delta 2\theta_H$ we have denoted the texture factor and, respectively, the peak shift.

For the peak shift, we use a parameterized formula derived by taking into account the misalignment errors.

$$\Delta 2\theta_H = d_0 + d_1 \sin 2\theta_H / \sin\alpha_H + d_2 \sin(\alpha_H - 2\theta_H) / \sin\alpha_H \quad (2)$$

where d_i are refinable parameters and $D_i(\theta_H)$ are calculated functions on diffractometer geometry. Our practice has shown that the second order contributions are not relevant for the instruments described above.

The texture factor T_H is just the value of the pole distribution function $A(\mathbf{y,h})$ for $\mathbf{y} = \mathbf{q}$, the unit vector along the wave vector transfer. The unit vectors $\mathbf{y}$ and $\mathbf{h} = \mathbf{H} / H$ are defined in two orthogonal coordinate systems related to the sample and, respectively, to the crystallite, with the **z** axis along the n-fold axis and the **x** axis along a 2-fold axis if this exists. These unit vectors are given by the polar and azimuthal angles (Ψ,γ) and, respectively, (Φ,β). In our Rietveld program, we use the model for the pole distribution function proposed by Popa [7]:

$$A(\mathbf{q},\mathbf{h}) = 1 + \sum_l \sum_{\nu=1}^{N(2l)} \left[\sum_{\mu=1}^{M(2l)} t_{2l}^{\mu\nu} C_{2l}^{\mu}(\Phi,\beta)\right] S_{2l}^{\nu}(\Psi,\gamma) \quad (3)$$

where $t_{2l}^{\mu\nu}$ are refinable parameters and $C_{2l}^{\mu}(\Phi,\beta)$, $S_{2l}^{\nu}(\Psi,\gamma)$ are symmetrized spherical harmonics having crystal and, respectively, sample symmetry.

To test the program, we have used a diffraction pattern measured on a rectangular plate of Al_2O_3 with the dimensions $65 \times 65 \times 1$ mm^3 in the range $2\theta_s$ from 20^0 to 125^0 with the step of 4'. The spatial group of this compound is R$\bar{3}$c. In the rhombohedral setting, the aluminium atoms have the Wyckoff positions 4c with the coordinates (u,u,u) and the oxygen atoms have the position 6e, the coordinates being $(u,1/2-u,1/4)$. For texture, we have supposed a cylindrical sample symmetry with the axis perpendicular to the plate. The pole distribution is then independent on γ and has the expression:

$$\begin{aligned} A(\Psi,\Phi,\beta) = 1 + t_1 P_2^0(\Phi) P_2^0(\Psi) + \left[t_2 P_4^0(\Phi) + t_3 P_4^3(\Phi)\sin 3\beta\right] P_4^0(\Psi) \\ + \left[t_4 P_6^0(\Phi) + t_5 P_6^3(\Phi)\sin 3\beta + t_6 P_6^6(\Phi)\cos 6\beta\right] P_6^0(\Psi) \\ + \left[t_7 P_8^0(\Phi) + t_8 P_8^3(\Phi)\sin 3\beta + t_9 P_8^6(\Phi)\cos 6\beta\right] P_8^0(\Psi) + \ldots. \end{aligned} \tag{4}$$

where P_{2l}^m are Legendre functions. The texture factor in the calculated pattern is: $T_H = A[\alpha_s(2\theta_H) - \theta_H, \Phi_h, \beta_h]$. For the thermal factor, we have taken the isotropic approximation. The result of refinement is given in Table 1.

4. MONTE CARLO PROGRAM TO COMPUTE THE INSTRUMENTAL DIFFRACTION LINE PROFILE FOR A FOCUSING HIGH-RESOLUTION CONFIGURATION

4.1. Introduction

In order to obtain the structure data from the powder diffraction patterns, for any kind of X ray or neutron diffractometer, a complete knowledge concerning the line profile, the dependence on the scan variable (the scattering angle for crystal diffractometry and the wavelength for TOF diffractometry) of the parameters describing the line profile and the line shifts from the corresponding expected positions, are required. Therefore, the proper computation of the instrumental profile is to be given.

While for X ray and conventional neutron diffractometers, this problem is quite simple [10], for a focusing geometry [2, 4], the difficulties involved are significant. This problem is theoretically treated in Refs [11, 12]. The corresponding algorithms are somewhat different but the results should be similar. A significant difference is that in Ref. [11], the exact shape of the pneumatically bent crystal surface is used, while in Ref. [12] is considered only an approximate one, characterized by an equivalent radius of curvature. The algorithm follows the computing procedure described in Ref. [11].

TABLE 1. THE REFINED PARAMETERS FOR AL_2O_3

Lattice parameters	a = 5.1338(2) Å	$\alpha = 55.270(3)^o$
Structural parameters	$u_{Al} = 0.3521(3)$	$B_{Al} = 0.15(5)$ Å^2
	$u_O = 0.5567(3)$	$B_O = 0.19(3)$ Å^2
Scale factor	$s = 072(1)$	
Profile parameters	$T = 0.041(3)$; $U = 078(5)$; $V = 0.067(2)$	
Shift parameters	$d_0 = -0.268(8)$; $d_1 = 0.07(1)$; $d_2 = 0.36(1$	
Texture parameters	$t_1 = 0.52(2)$; $t_2 = 0.14(2)$; $t_3 = 0.08(2)$; $t_4 = 0.$; $t_5 = 0.$;	
	$t_6 = 0.$; $t_7 = -0.13(2)$; $t_8 = 0.$; $t_9 = 0.13(1)$	
Hi-square, reliabilities	$\chi^2 = 1.53$; $R = 0.12$; $R_w = 0.15$	

4.2. The description of the experimental setting

The focusing experimental setting is given in Fig. 1. Such focusing configurations are realized in Pitesti, Bucharest [2] and Missouri [4]; it is possible that other focusing configurations based on the principles given in Refs [2, 4] be realized elsewhere, too.

FIG. 1. The experimental setting.

The distance between the reactor core $Z.A.$ and the curved monochromator M is L_0. Between the reactor core and the monochromator, there are two coarse collimators: C_r with circle shaped window (radius R_r) and length C_r and C_0, with rectangular shaped window, with the corresponding dimensions C_0 (length), $(2v_0,2w_0)$ (the window heights and widths). The disk shaped monochromator thickness and radius are $2v_m$ and R_m respectively. Between the monochromator and sample, there is the coarse collimator C_1 with variable rectangular section of dimensions C_1, $(2v_1',2w_1')$ at entrance, $(2v_1,2w_1)$ at exit. The distance between the monochromator and the collimator entrance is L_1. The distance between the collimator exit and the sample centre is L_{1s}; the distance between sample and detector is L_{2s}, the rectangular detector window dimensions are $(2v_2,2w_2)$ and the sample thickness is $2v_s$. The Bragg angle is θ_m, the reflecting plane (511), and the cutting angle is χ. For a given scattering angle value 2θ the sample inclination angle is $\alpha(2\theta)$ according to the focusing condition [4].

The following alignment errors are taken into consideration:

— taum (min): the real Bragg angle is not $2\theta_m$ but $2\theta_m + \tau_m$.
— (x_{m0}, y_{m0}, z_{m0}) in cm, the errors in the monochromator positioning from the neutron beam centre.
— (φ_m,ψ_m) in min. the monochromator angular alignments errors in horizontal respectively the vertical plane.
— r_0 (cm), the diffractometer rotation axis positioning errors, from the monochromatic beam centre.
— r_s (cm) the sample centre positioning errors, from the diffractometer rotation axis.
— (φ_s,ψ_s) in min, the sample angular positioning errors, from the corresponding expected positions, in horizontal respectively the vertical plane.

4.3. The input and output files description

The Monte Carlo Resolution for Neutron Diffractometer (MCRND) program is written in Fortran; it has the input file MCRND.INP and three output files MCRND.OUT, MCRND.DAT and MCRND.REZ.

The input file MCRND.INP contains the general data concerning the experimental setting, the alignment errors data, six values of the scattering data for which the computations are to be made and general data concerning the MC process: the initial values for the random numbers and the total number of computing cycles. The writing format is a free one.

The output file MCRND.OUT contains: the used input data, the scattering angle values, the ratio between the failed (for which the test was not passed) and the total cycles number, the luminosity (number of impulses), the normalized luminosity, the lines centre and the lines standard width D.

The output file MCRND.DAT contains eight columns of data: in the first two columns are given the $\Delta 2\theta$ in non-centred (the channel width, in min, is given as input data) respectively centred channels and the profiles for the six chosen scattering angles (non-normalized values); these values can be used by any commercially available fitting program, as MICROSOFT ORIGIN is.

The output file MCRND.REZ contains the same data as the above one but the profiles are normalized.

4.4. Test computations

Test computations were made for the focusing configuration existing in Pitesti for a pressure of 0.1315 atm. corresponding to an equivalent radius of curvature of 15.5 m; the chosen scattering angle values are 20°, 40°, 60°, 80°, 100° and 120°.

It seemed for us to be of greatest importance to estimate the presumed line shifts given by the alignment errors. Computations were performed for alignment error parameter values of experimental interest: between 0 and 30 min for taum, between 0 and 1 cm for x_{m0}, $y_{m0,}$ z_{m0}, between 0 and 16 min for Fim, between 0 and 60 min for Psim, between 0 and 1.2 cm for R0, between 0 and 0.5 cm for Rs, between 0 and 6 min for Fis and between 0 and 120 min for Psis.

Computations were also performed for combined alignment errors as follows:

A: taum=10 min, Fim=4 min, Psim = 30 min, all other values zero.
B: taum=10 min, Fim=4 min, Psim = 30 min, x_{m0}= $y_{m0=}$ z_{m0}=0.5 cm, all other values zero.
C: R0 = Rs =0.4 cm, all other values zero.
D: Fis = 2 min, Psis = 30 min, all other values zero.
E: R0 = Rs =0.4 cm, Fis = 2 min, Psis = 30 min, all other values zero.
F: taum=10 min, Fim=4 min, Psim = 30 min, x_{m0}= $y_{m0=}$ z_{m0}=0.5 cm, R0 = Rs =0.4 cm, Fis = 2 min, Psis = 30 min, all other values zero.

A graphical presentation of these computations is given. These examples prove the significant effect of the alignment errors, experimentally observed both at Pitesti NAD Bucharest focusing configurations.

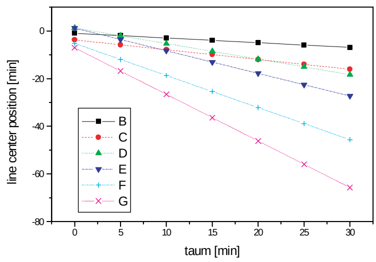


FIG. 2. Case A.
B: 2q = 20 deg., C: 2q = 40 deg.,
D: 2q = 60 deg., E: 2q = 80 deg.,
F: 2q = 100 deg., G: 2q = 120 deg.

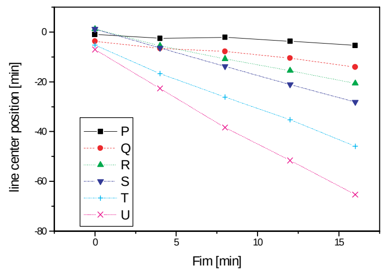


FIG. 3. Case B.
P: 2q = 20 deg., Q: 2q = 40 deg.,
R: 2q = 60 deg., S: 2q = 80 deg.,
T: 2q = 100 deg., U: 2q = 120 deg.

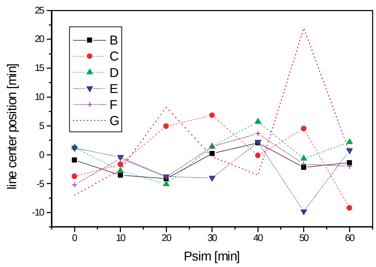


FIG. 4. Case C.
B: 2θ = 20 deg., C: 2θ = 40 deg.,
D: 2θ = 60 deg., E: 2θ = 80 deg.,
F: 2θ = 00 deg., G: 2θ = 120 deg.

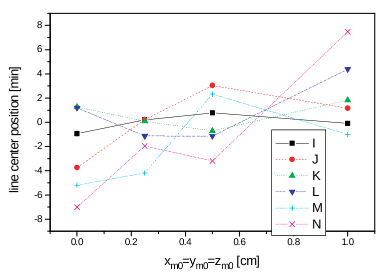


FIG. 5. Case D.
I: 2θ = 20 deg., J: 2θ = 40 deg.,
K: 2θ = 60 deg., L: 2θ = 80 deg.,
M: 2θ = 100 deg., N: 2θ = 120 deg.

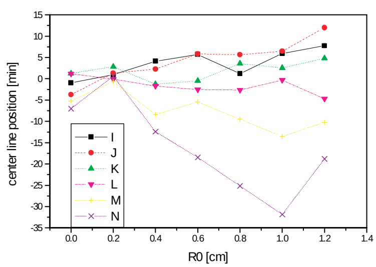


FIG. 5. Case E.
I: 2θ = 20 deg., J: 2θ = 40 deg.,
K: 2θ = 60 deg., L: 2θ = 80 deg.,
M: 2θ = 100 deg., N: 2θ = 120 deg.

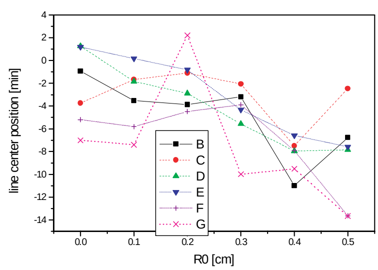


FIG. 7. Case F.
B: 2θ = 20 deg., C: 2θ = 40 deg.,
D: 2θ = 60 deg., E: 2θ = 80 deg.,
F: 2θ = 100 deg., G: 2θ = 120 deg.

4.5. Conclusions

The computations presented above prove that the line shifts related to the alignment errors depend rather randomly on the scattering angle and cannot be described using a certain function, for example, a polynomial. As the proper line positions are necessary in a Rietveld processing data, the program described is a first step in developing a Rietveld program, suited for a focusing configuration as presented in Refs [2, 4].

5. LSQ1 PROGRAM FOR THE GRAIN SIZE AND THE SECOND ORDER STRAIN DETERMINATION

The diffraction line profile is given by the instrumental contribution, the grain size contribution and the strain contribution. The last two contributions are wavelength dependent. The interval strain are either first order (real) ones given by mechanical processing or thermal treatments, for which the average value is not zero, and the second order ones given by the existence of a composition variation and described by the square root of the standard deviation square (in this case, the average value is zero). The second ones determine the line profile.

The de-convolution of these contributions, size broadening and strain broadening, can be made either by the Warren-Averbach method or using the Rietveld method. In the case of the second, one of the problems related to the line overlapping or the correct background description is overcome. In the corresponding minimization process, the relevant parameters to be optimized are the atom's positions in the unit cell, its dimensions, the thermal factors and the line profile parameters related to the grain size and the second order strain.

The measured intensity for the '$2\theta_s$' value of the scattering angle is:

$$I_s(2\theta_s) = B(2\theta_s) + \frac{s}{v_c^2}\sum_H \frac{n_H p_H v_H a_H t_H F_H^2}{\sin\theta_H \sin(2\theta_H)} V_H(2\theta_s - 2\theta_H - \Delta 2\theta_H) \tag{5}$$

where $B(2\theta_s)$ is the background, s is a scale factor, v_c is the unit cell volume, θ_H is the Bragg angle for the line corresponding to the 'H' inverse lattice vector, n_H, p_H, v_H, a_H, t_H are the multiplicity a polarization, the irradiated volume, the sample transmission and the texture factors, respectively; F_H is the structure factor and $\Delta 2\theta$ is the maximum line shift given by the instrumental aberrations and the first order strain. $V_H(2\theta\text{-}2\theta_H\text{-}\Delta 2\theta_H)$ is the line profile, given by:

$$\mathrm{VH}(\Delta 2\theta)=\int d(\Delta 2\theta)\mathrm{RH}(\Delta 2\theta)\int d(\Delta 2\theta)\int \mathrm{LH}(\Delta 2\theta+\Delta 2\theta+(\Delta 2\theta')\mathrm{DH}(\Delta 2\theta') \tag{5'}$$

where R_H is the instrumental profile, L_H is the grain size profile and D_H is the strain profile. The L_H and D_H function are well approximated by a Lorenz and a Gauss function, respectively. The R_H function could be either computed (using the DAX program) or experimentally determined and is also basically a Gauss one. Therefore, the convolution $G_H=R_H*D_H$ is also a Gauss function and the convolution $L_H*G_H=V_H$ is a Voigt function. The Voigt function components can be written as:

$$L_{(x)} = \frac{1}{\beta_L}\frac{1}{1+\eta^2 X^2/\beta_L^2} \qquad G_{(x)} = \frac{1}{\beta_G}\exp(-\pi x^2/\beta_G^2) \tag{6}$$

where β_L, β_g are the corresponding full widths given by:

$$\beta_g^2 = T\tan 2\theta_H - U\tan\theta_H + V + Z\frac{\sin 2\theta_H}{\sin\alpha_H} + 8\pi\tan 2\theta_H < \sum{}_{44}^{2} > \tag{7a}$$

$$\beta_L = \frac{2\lambda}{3 < R_H > \cos\theta_H} \tag{7b}$$

In (7a), the first four terms represent the instrumental contribution [2], α_H is the sample orientation angle measured from the incident beam direction, $<\Sigma^2_{44}>$ is the microstrain dispersion in the $h = \frac{\bar{H}}{H}$ direction and $<R_4>$ is the coherence region radius in the h direction [8]. The method used is not a genuine Rietveld one because this method involves too many optimization parameters and a great number of experimental measured diffraction peaks. For the proposed method, only a parameterized formula for the measured intensity (instead of 5) is used:

$$I(2\theta_s) = B(2\theta_s) + \sum_H A_H V_H(2\theta_i - 2\theta'_H)/V_H(0) \tag{5"}$$

And, therefore, only the amplitudes A_H and the line position $2\theta'_H$ are taken as fitting parameters.

The background is described by a 4th degree polynom:

$$B(2\theta I)=Bo+B1(2\theta I)+B2(2\theta I)2+B3(2\theta I)2+B4(2\theta I)3 \tag{8}$$

The total fitting parameters are

B_o, B_1, B_2, B_3, B_4, T, U, V, Z, STR=$8\pi<\Sigma^2{}_{44}>$, $DIM = \dfrac{2\lambda}{3 < l_n >}$, $A_1(2T)$ …

$A_{NM}(2T)$, where T=$20^o{}_K$ and NM is the total diffraction peaks considered. Therefore, their number is NP=11+2*NM

It is to be stressed, according to the formula (7), that only the sum of the V and STR parameters can be determined. Therefore, to determine the microstrain parameter STR is necessary to determine first the instrumental contribution, using a standard sample with large size dimensions. Such a sample can be obtained by using the thermal treatments.

Program description

There are in fact two programs: INPFSS and the main program LSQ1. The name and the extension of the input and output data fills are optional. The program INPFSS prepares one of the effective input files using the experimental data. The exit file OPT. DAT contains the following columns: the current number, the scattering angle, the measured intensity, the Poisson weight W_I (0 if the corresponding data are not to be used).

This file is one of the input files for the main program LSQ1. The second input file OPT. PAR contains the input values for the fitting parameters. The LSQ1 program produces two output files. The first one OPT1. PAR has the same columns number as OPT. PAR, containing the final values of the fitted parameters. An added column contains the corresponding standard deviations, the confidence factors and the correlation matrix values.

The second output file OPT1.DAT contains five columns, the current number, the scattering angle $2\theta_s$, the measured intensity, the computed intensity, their difference; this file can be used for a graphic representation in connection with a standard program commercially available (e.g. GRAPHER, ORIGINAL).

REFERENCES

[1] IONITA, I., et al., TRIGA Conference, Mainz (1996).

[2] IONITA, I., STOICA, A.D., POPOVICI, M., POPA, N.C., Nucl. Instrum. Method., **A 431** (1999) 509-520; IAEA Technical Committee Meeting on Neutron Beam Research, Lisbon, 1997.

[3] POPOVICI, M., STOICA, A.D., IONITA, I., J. Appl. Crystallogr. **20** (1987) 90–101.
[4] POPOVICI, M., et al., Nucl. Instrum. Methods **A 338**, (1994) 99–110.
[5] IONITA, I., CIOCANESCU, M., STOICA, A.D., POPOVICI, M., 12th European TRIGA Users Conference, Bucharest (1992).
[6] STOICA, A.D., Acta Crystallogr. **A31** (1975) 189–192.
[7] POPA, N.C., J. Appl. Crystallogr. **25** (1992) 611–616.
[8] POPA, N.C., J. Appl. Crystallogr. **31** (1998) 176–180.
[9] IONITA, I., STOICA, A.D., 13th European TRIGA Users (Proc. Conf. Ljubljana, 1994) J. Appl. Crystallogr. **33** (2000) 1067–1074.
[10] Mathematical theory of X-ray powder diffractometer, A.J.C Wilson, Netherlands Philips Technical Library, (1963).
[11] POPA, N.C., IONITA, I., The instrumental profile for a neutron powder diffractometer in focusing geometry, Physica B 276–278 (2000) 192–194.
[12] STOICA, A.D., POPOVICI, M., YELON, W.B., Instrumental line shape in neutron powder diffraction with focusing monochromators, J. Appl. Crystallogr. **33** (2000), 137–146.
[13] IONITA, I., STOICA, A.D., POPA, N.C., "Focusing configurations and optimization conditions in crystal neutron diffractometry", Yugoslav Nuclear Society 2nd European Conference on Neutron Scattering (Proc. Conf. Belgrade, 1998), Budapest (1999).
[14] DORNER, B., Acta Crystallogr. **A28** (1972) 319–327.

CORE MANAGEMENT AND CALCULATION TOOLS FOR THE WWR-M RESEARCH REACTOR IN UKRAINE

Y.P. MAHLERS
Institute for Nuclear Research,
Kiev, Ukraine
Email: mahlers@gluk.org

Abstract

Location of fuel assemblies in the core satisfying all the safety constraints and fuel requirements, fuel types used, number of fuel assemblies of each type and their discharged burn-ups, as well as the number of beryllium blocks at the periphery of the core, are optimized for the WWR-M research reactor in Ukraine using the code PORT. To determine the best core layout providing high neutron flux and low fuel expenditures under the safety and fuel constraints, two optimization algorithms using successive integer linear programming and simulated annealing are applied. For neutronics calculation, the iterational hybrid method combining the diffusion model with higher approximations of the neutron transport equation is used. The thermal-hydraulics model is based on an empirical formula for the heat transfer coefficient and measurement data for hydraulic parameters, including relative coolant velocities and pressure drops. It has been shown that a mixed core consisting of WWR-M2 and WWR-M5 fuel provides higher neutron flux and even less fuel expenditures in comparison with the core consisting of only WWR-M2 fuel. Utilization of mixed fuel is especially effective for high 'neutron price'. The WWR-M reactor in Ukraine is being studied for conversion from HEU to LEU fuel. It is feasible to convert the reactor to LEU fuel using qualified LEU WWR-M2 fuel assemblies, which are currently available for this purpose.

1. INTRODUCTION

The WWR-M research reactor in Ukraine is used for various purposes, including neutron physics and materials research, radioisotope production and neutron transmutation doping of silicon. Thus, its core management should be flexible to correspond to various objectives with different priorities and to provide the best possible result.

Core loading patterns satisfying all the safety constraints and fuel requirements, fuel types used, number of fuel assemblies of each type and their discharged burn-ups, as well as the number of beryllium blocks at the periphery of the core should be determined by the optimization procedure using computer codes based on special optimization techniques. To provide high reliability and safety of the reactor, adequate neutronics and a thermal-hydraulics models should be applied.

2. NEUTRONICS MODEL

Diffusion approximation is invalid for neutronics modelling of the WWR-M reactor because of neutron streaming phenomenon in beam tubes. The Monte Carlo method and various high-order approximations of the neutron transport equation can be used for such calculations but they are computationally very intensive. Moreover, the Monte Carlo method can not provide statistically reliable results for local characteristics of real heterogeneous reactors, for example, for detailed power distribution and, hence, for power peaking factor. As concerns discrete ordinate methods, they are not quite accurate for calculation of the reactors containing voids, even for high-order angular approximations, because of ray effects.

The iterational hybrid method combining the diffusion model with higher approximations of the neutron transport equation has been developed for the neutronics calculation of the WWR-M reactor [1]. This technique has been verified by comparing its results for 1-D and 2-D test problems to solutions obtained for the same problems using the Monte Carlo and high-order discrete ordinate methods. The code VICA based on this method has been developed for the 3-D neutronics calculation of the WWR-M reactor. The results of the calculation were shown to be consistent with the results of measurement [2]. The total number of measurement data was about 100. The root-mean-square deviation of the calculated and measured values of the thermal neutron flux was about 7%. Errors in calculation of the effective multiplication factor for various core loading patterns were less than 0.5%.

The code was examined also by the analysis of a research reactor accident such as damage to a large neutron beam tube located near the core with accompanying release of positive reactivity due to the penetration of water. The penetration of water into the void tube was simulated by loading an ampoule filled up by water. The measured value of reactivity was 0.09 β_{eff} while the calculated value was 0.12 β_{eff}.

3. THERMAL-HYDRAULICS MODEL

Thermal-hydraulics calculation of the WWR-M reactor core is performed using the approach developed in the Peterburg Nuclear Physics Institute [3].

The model is based on an empirical formula for the heat transfer coefficient and measurement data for hydraulic parameters, including relative coolant velocities and pressure drops [4]. The model was validated by the measurement of fuel assembly surface temperatures and by special experiments with the heating of dummy fuel elements until their destruction [5].

4. CORE LOADING PATTERN OPTIMIZATION

An effective way to improve performance of the reactor is to optimize its core layout, increasing neutron flux in the reactor channels and decreasing its fuel expenditures. Optimal locations of all fuel assemblies in the core, fuel types used, number of fuel assemblies of each type and their discharged burn-ups, as well as the number of beryllium blocks at the periphery of the core should satisfy all the safety constraints and fuel requirements including the constraint on maximum temperature of fuel assembly surface, which is dependent on detailed power distribution in the core.

The problem stated can be characterized as a large combinatorial problem with non-linear compute-intensive objectives and constraints. Although various heuristic approaches exist and help to generate improved core loading patterns, they can not provide high quality of the solutions obtained. To solve such optimization problems and improve core management for the WWR-M reactor, the code PORT has been developed [6].

To determine the best placement of fuel assemblies in the core, PORT can use two algorithms. In the first one [7], the objective function and constraints are successively linearized in terms of independent binary variables specifying locations of fuel assemblies and the problem is cast into integer linear programming. To decrease the computer time necessary to calculate sensitivity coefficients, backward diffusion calculation is applied. By application of this algorithm, one optimization problem can be solved for two hours using an Intel 80486–based personal computer.

Successive integer linear programming is valid for core reload optimization of the WWR-M reactor because of little fuel depletion during the cycle and, hence, the possibility to restrict alterations in core characteristics arising during fuel shuffling to apply the linearity approximation. However, as a more powerful computer is available, the second algorithm based on simulated annealing can be used [8]. It requires much more CPU time but provides better solutions, especially for mixed-fuel cores. By application of this algorithm, one optimization problem can be solved for three hours using a Pentium III/733–based personal computer.

PORT has a graphical interface for automation and visualization of core design to specify and control input data and review the calculation results in a convenient form. Moreover, information for all core reloads is recorded in a database, thus the whole history of each fuel assembly can be reviewed and analysed [6].

5. CORE MANAGEMENT

Since the WWR-M reactor is used for various purposes, its efficiency can be characterized by the following parameter:

$$\sum_n p_n \Phi_n - C_F$$

where p_n and Φ_n are the 'neutron price' and neutron flux in the n-th irradiation channel, respectively, and C_F is the feed fuel cost. The feed fuel cost can be written as follows:

$$C_F = \sum_{i=1}^{M} \frac{c_i P_i}{\gamma_i m_i B_i^D}$$

where γ_i, m_i and c_i are the thermal energy released per g of the ^{235}U burned, initial mass of ^{235}U in g and cost for fuel assembly of type i, respectively, M is the number of fuel types, and P_i and B_i^D are the total power of all fuel assemblies of type i in the core and their average discharge burn-up, respectively.

The core of the WWR-M reactor can be built using two available types of fuel assemblies: WWR-M2 with 36% enrichment and WWR-M5 with 90% enrichment, containing 37 and 66 g of ^{235}U, respectively. Their parameters and designs are shown in Table 1 and Fig. 1. WWR-M5 fuel assemblies provide more intensive heat transfer than WWR-M2. However, WWR-M5 discharge burn-up has to be less than 50% of the initial mass of ^{235}U, while WWR-M2 fuel assemblies operate successfully up to 80% burn-up. The total power of the reactor is restricted by 10 MW. The number of channels for experiments, isotope production and neutron transmutation doping of silicon is not too large. Moreover, the current design and requirements to the control rod system do not allow the channels to be near the centre of the core.

It has been shown (see Ref. [9]) that a mixed core consisting of WWR-M2 and WWR-M5 fuel provides higher neutron flux and even less fuel expenditures in comparison with the core consisting of only WWR-M2 fuel. Utilization of mixed fuel is especially effective for high 'neutron price'.

TABLE 1. FUEL ASSEMBLY PARAMETERS

	WWR-M5	WWR-M2	LEU WWR-M2
Enrichment, %	90	36	19.75
Number of fuel elements	6	3	3
Mass of ^{235}U, g	66	37	41.7
Fuel meat composition	UO_2-Al 1.2 gU/cm^3	UO_2-Al 1.1 gU/cm^3	UO_2-Al 2.5 gU/cm^3
Pitch/flat-to-flat, mm	35/33.5	35/32	35/32
Element/clad/meat, mm	1.25/0.43/0.39	2.5/0.76/0.98	2.5/0.78/0.94
Specific heat transfer surface, cm^2/cm^3	6.6	3.67	3.67
Pressure drop	6.5	4.35	4.35
Relative coolant velocities between the fuel elements (starting from the centre)	0.90; 1.01; 1.08; 0.98; 1.06; 0.88	1.18; 0.89; 1.05; 0.86	1.18; 0.89; 1.05; 0.86

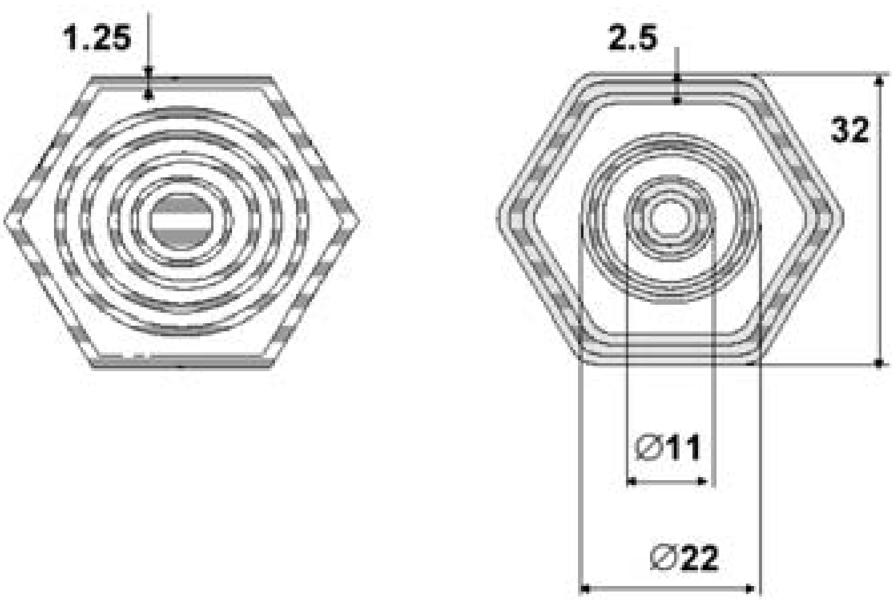


FIG. 1. Fuel assembly designs.

Optimum location and the number of fuel assemblies of different types in the core is strongly dependent on the 'neutron price'. With growth of the 'neutron price', the total number of fuel assemblies in the core should be decreased, while the number of WWR-M5 fuel assemblies should be increased. For a not too low 'neutron price', WWR-M5 fuel assemblies should be located near the irradiation channels to make power density in these areas, limited by the maximum allowed temperature of fuel surface, as high as possible, thus increasing neutron flux [9].

The WWR-M reactor is being jointly studied with the Argonne National Laboratory for conversion from highly enriched uranium (HEU) to low enriched uranium (LEU) fuel [10]. Candidate LEU replacement fuel assemblies are LEU WWR-M2 (19.75%), which have been tested in the WWR-M reactor in Gatchina by irradiation to over 75% burn-up [11]. To study the reactor performance with the HEU (36%) and LEU (19.75%) WWR-M2 fuel assemblies, the MCNP, DIF3D, REBUS, WIMS-ANL and NJOY codes using ENDF/B-VI data were used [12–18]. It is feasible [10] to convert the WWR-M reactor in Ukraine to LEU fuel using qualified LEU fuel that is currently available for this purpose.

REFERENCES

[1] MAHLERS, Y.P, SIROTKIN, A.M., An iterational approach to treat local heterogeneous effects, Fizika Yaderno-Energ. Ustanovok, pp. 38–41, Energoatomizdat, Moscow (1988) (in Russian).

[2] IEVLEV, S.M., et al., Calculation and Measurement of Neutron Distributions in the WWR-M Reactor, Rep., preprint no. 89-2, Kiev Institute for Nuclear Research, Kiev (1989) (in Russian).

[3] ISAKAS, I.E., KIRSANOV, G.A., KONOPLEV, K.A., Thermal calculation of the WWR-M reactor core, Rep., preprint no. 859, Peterburg Nuclear Physics Institute, Gatchina (1983) (in Russian).

[4] KIRSANOV, G.A., KONOPLEV, K.A., PIKULIK, R.G., SHISHKINA, Z.A., Hydraulics of the WWR-M reactor core, Atomn. Energ. **39** 5 (1975) 320 (in Russian).

[5] KIRSANOV, G.A., KONOPLEV, K.A., SYASIN, A.N., SHISHKINA, Z.A., Determination of the maximum allowed power density for the WWR-M reactor fuel assemblies, Rep., preprint no. 285, Peterburg Nuclear Physics Institute, Gatchina (1976) (in Russian).

[6] MAHLERS, Y.P., PORT: "Core design optimization code for WWR-M research reactors", Mathematical Methods and Supercomputing for Nuclear Applications, Joint Int. Conf. Saratoga, USA, 1997, Vol.1, pp. 217–219.

[7] MAHLERS, Y.P., Core loading pattern optimization for research reactors, Ann. Nucl. Energy **24** 7 (1997) 509.
[8] MAHLERS, Y.P., Core reload optimization for equilibrium cycles using simulated annealing and successive linear programming, Ann. Nucl. Energy **29** 11 (2002) 1327.
[9] MAHLERS, Y.P., "Performance of the WWR-M research reactor in Ukraine with a mixed-fuel core", 7th Int. Topical Mtg on Research Reactor Fuel Management (RRFM 2003), Aix-en-Provence, France, 2003.
[10] POND, R.B., HANAN, N.A., MATOS, J.E., MAHLERS, Y.P., DYAKOV A.G., "Neutronic performance of the WWR-M research reactor in Ukraine", 24th Int. Mtg on the Reduced Enrichment for Research and Test Reactors (RERTR 2002), Bariloche, Argentina, 2002.
[11] KONOPLEV, K.A., PIKULIK, R.G., ZAKHAROV, A.S., PANEVA, G.V., TSCHMSHKYAN, D.V., "LEU WWR-M2 fuel qualification", 24th Int. Mtg on the Reduced Enrichment for Research and Test Reactors (RERTR 2002), Bariloche, Argentina, 2002.
[12] BRIESMEISTER, J. F. (Ed.), MCNP™ – A General Monte Carlo N-Particle Transport Code, Version 4C, LA-13709-M, Los Alamos National Laboratory, Los Alamos, NM (2000).
[13] DERSTINE, K.L., DIF3D: A Code To Solve One-, Two-, and Three-Dimensional Finite-Difference Diffusion Theory Problems, ANL-82-64, Argonne National Laboratory (1984).
[14] LAWRENCE, R.D., The DIF3D Nodal Neutronics Option for Two- and Three-Dimensional Diffusion Theory Calculations in Hexagonal Geometry, ANL-83-1, Argonne National Laboratory (1983).
[15] TOPPEL, B.J., A User's Guide for the REBUS-3 Fuel Cycle Analysis Capability, ANL-83-2, Argonne National Laboratory (1983).
[16] OLSON, A.P., A User's Guide for the REBUS-PC Code, Version 1.4, ANL/RERTR/TM02-32, Argonne National Laboratory (2001).
[17] DEEN, J.R., WOODRUFF, W.L., COSTESCU, C.I., LEOPANDO, L.S., WIMS-ANL User Manual, Rev. 4, ANL/RERTR/TM-23, Argonne National Laboratory (2001).
[18] MACFARLANE, R.E., MUIR, D.W., BOICOURT, R.M., The NJOY Nuclear Data Processing System, Volume I: User's Manual, LA-9303-M (ENDF-324), Los Alamos National Laboratory, Los Alamos, NM (1982).

BLANK

NEUTRONIC DESIGN OF THE FIRST CORE OF THE REPLACEMENT RESEARCH REACTOR

E. VILLARINO
Nuclear Engineering Division,
INVAP S.E.
Bariloche
Email: men@invap.com.ar

S. GOMEZ, V. ISHIDA
NuEnCon,
Buenos Aires

Argentina

Abstract

The paper describes the general neutronic characteristics of the first core of the replacement research reactor (RRR) for the Australian Nuclear Science and Technology Organisation (ANSTO). A compact core with 16 FA has been designed to fulfil all the very demanding neutronic requirements of the RRR facility. The contractual performance parameters must be verified for the equilibrium core; a very important design effort was carried out in the initial fresh core to have a similar performance. The description covers different aspects of the neutronic design: a detailed nuclear design of U_3Si_2 first core, the design calculation tools, together with a comparison of the first core performance against the core design criteria and the equilibrium core performance.

1. RRR DESCRIPTION

The replacement research reactor (RRR) facility is a multi-purpose open-pool type reactor. The nominal fission power of the reactor is 20 MW. The core is located inside a chimney, surrounded by heavy water contained in the reflector vessel. The whole assembly is at the bottom of the reactor pool, which is full of de-mineralized light water acting as coolant and moderator and biological shielding. The core is an array of 16 plate-type fuel assemblies (FAs) and five absorber plates, which are called control plates (CP). The FAs are square shaped, each of them containing 21 fuel plates and using Cd wires as burnable poison. The coolant is light water, which flows upwards.

Reactor shutdown can be achieved by two independent means, which are the insertion of five CRs into the core, or the partial drainage of the heavy water from the reflector vessel.

The fuel plates consist of meat and cladding. Two types of meat are considered: uranium silicide powder or uranium molybdenum powder, both dispersed in an aluminium matrix with enrichment lower than 20%.

The reactor will be provided with advanced cold and thermal neutron sources to produce neutrons in specific spectral ranges. Specially designed neutron beam guides featuring high technology super-mirrors will extract these neutrons from their sources.

One of the main purposes of the reactor includes the large-scale production and processing of radionuclides. The following facilities are specified:

— Bulk production irradiation facilities, to irradiate targets contained in rigs that are placed inside irradiation tubes provided in the reflector tank.
— Long residence time general purpose irradiation facilities, to irradiate targets contained in sealed cans. The cans are sent to irradiation rigs at the reflector tank by means of a pneumatic transport system.
— Short residence time facility, to carry out neutron activation analysis.
— Large volume irradiation facilities, for neutron transmutation doping of single-crystal silicon ingots and for bulk irradiation of ore samples for neutron activation analysis.

The following sections will describe general aspects of the neutronic design of the first core of the RRR, its neutronic nuclear safety characteristic and its neutronic requirements.

2. NEUTRONIC NUCLEAR SAFETY DESIGN CRITERIA

The neutronic design of the RRR is guided by a set of neutronic design criteria. These criteria include the contractual requirements and the nuclear safety requirements. This section gives a summary of the nuclear safety neutronic design criteria, as follows:

— *Reactivity design criteria*: Different criteria were settled for the core design, such as negative feedback coefficients, enough shutdown margins for both shutdown systems, including single failure criteria. Actuation time of the shutdown systems (measure as shutdown margin versus time), reactivity worth of the irradiation facilities, reactivity rate of the irradiation facilities, reactivity rate of the control system, etc.

— *Thermal-hydraulic related criteria:* For 20 MW(th), the number of FAs shall be 16 and the power peaking factor lower than 3.
— *Operating condition design bases*: A minimum end of cycle reactivity to allow the reactor to return to full power operation 30 min after a reactor trip occurs and the cycle length must be at least 28 days, with two days for refuelling and maintenance.

3. NEUTRONIC REQUIREMENTS

The contract specifies several neutronic aspects to be fulfilled by the core design and to be verified during the commissioning tests. The following list gives a summary of such warranted values.

Flux level

Several irradiation facilities require a minimum value for the neutron flux; but there are other irradiation facilities requiring a minimum and maximum neutron flux value.

Spectra

There are requirements on the flux value for different energy of neutrons:

— Cold neutron flux: neutrons with energy lower than 10 meV.
— Thermal neutron flux: there are two different definitions.
— Neutron sources or beams: neutrons with energy lower than 100 meV and greater than 10 meV.
— Radioisotope production: neutrons with energy lower than 0.6 eV.
— Fast neutron flux: neutrons with energy greater than 1 MeV.

There is an additional requirement to keep room for a future hot neutron source. The hot neutron flux is defined for neutrons with energy lower than 1 eV and greater than 100 meV. In the case of the neutron beams (cold and thermal), there is also a requirement on energy of the spectrum peak.

Homogeneity

There are requirements on the flux homogeneity on different irradiation facilities, for example, the axial homogeneity on NTD facilities, inside an irradiation can, and within several targets of the same flux level.

Perturbation

There are requirements on the flux perturbation due to the movement of the irradiation samples. It means the flux in the other facilities must not change a given value when an irradiation device is moved.

Burn-up

A minimum discharge burn-up is required.

Restart capability

The reactor must have the capability to return to full power operation after a trip, within 30 min. This condition is not only a neutronic requirement, it also requires an important excess of reactivity at EOC to have enough time to return the reactor at full power.

4. FIRST CORE DESIGN GOALS

Additional to the nuclear safety design criteria, the following design goals were deemed:

(a) The reactor must operate at maximum power.
(b) All irradiation facilities must be available with a high performance.
(c) The operating cycle should meet the requirements of the equilibrium core.
(d) Fresh FAs are used.
(e) Minimize the number of different FA.
(f) The differences between these fuels are only uranium load and the BP distribution.

5. DESIGN CALCULATION CODES

The calculation of the RRR is done in several steps and using different validated codes. These steps and codes are summarized in a calculation line. The calculation line for RRR is divided into three different methodologies:

Calculation using macroscopic cross-sections

This methodology is used for almost all the neutronic parameters. The equilibrium core burn-up distribution is the most important calculated parameter.

Calculation using microscopic cross-sections

This methodology is used for the calculation of the kinetic parameters and time dependent calculation.

Monte Carlo code

This calculation methodology is used for the verification of several neutronic parameters.

The first two methodologies are divided into three steps:

— Library generation;
— Cell calculation;
— Core calculation.

The last methodology is divided into two steps:

— Library generation;
— Monte Carlo calculation.

These methodologies and their interfaces are shown in Fig. 1. Several codes and a short description of the most relevant items are given.

Nuclear data library

Two different primary data are used to generate ESIN type libraries.

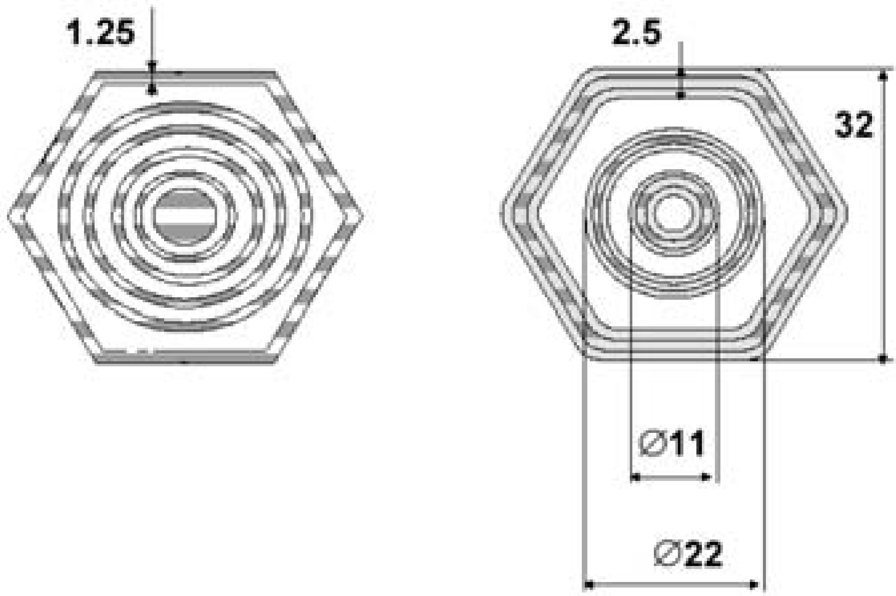


FIG. 1. Calculation line scheme.

- *WIMS-ESIN:* This results from the 69 groups WIMS library, which has good thermal detail as well as resonant parameters. Moreover, it has isotopes added from the ENDF/B-IV library for control absorber material definition and a new set of isotopes was added from the ENDF/B-VI: Ir and Te, using NJOY system.
- *HELIOS-ESIN:* Primary data of the HELIOS are from the ENDF/B-VI library. The library has three different group structures: 190, 89 and 34 groups.
- *CONDOR:* The CONDOR Code for neutron calculations is used to calculate fuel cells, fuel-rod clusters, as well as fuel plates with slab geometry or 2D geometry. Flux distribution within the region to be calculated is obtained through the collision probability method or the Heterogeneous Response Method in a multi-group scheme with various types of boundary conditions.
- *HXS:* The HXS program (cross-section handler) represents a major utility. It handles macroscopic cross-sections (identified by a name) in library form.
- *CITVAP:* The CITVAP reactor calculation code is a new version of the CITATION-II code, developed by INVAP's Nuclear Engineering Division. The code was developed to improve CITATION-II

performance. In addition, programming modifications were performed for its implementation on personal computers. The code solves 1, 2 or 3-dimensional multi-group diffusion equations in rectangular or cylindrical geometry. Spatial discretization can also be achieved with triangular or hexagonal meshes. Nuclear data can be provided as microscopic or macroscopic cross-section libraries.

MCNP Monte Carlo code

This well-known Monte Carlo transport code for neutron and gamma calculations uses ENDF/B-VI cross-sections in any order and performs 3-D calculations. It is used to verify some neutron parameters through an independent calculation method.

6. FIRST CORE DESIGN

6.1. General aspect

A compact core with 16 FA has been designed to fulfil all the neutronic design criteria. Figure 2 shows a scheme of the core layout with the FA and the control rods.

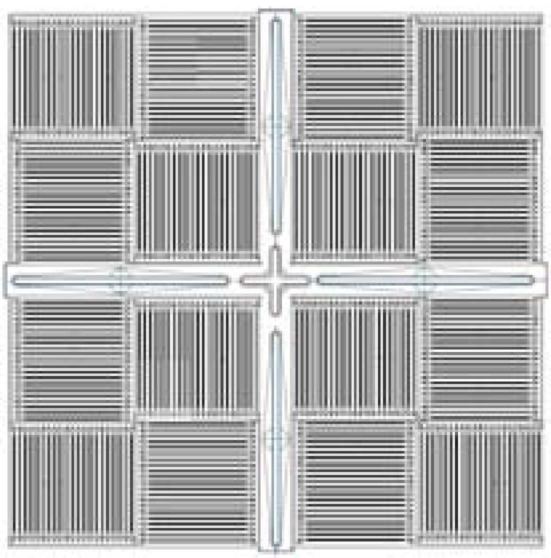


FIG. 2. Core layout scheme.

The equilibrium core uses U_3Si_2 with an uranium density of 4.8 g-U/cm^3. The fuel management strategy has an operating cycle of 33 FPD and 2 days for maintenance, and the number of FA per cycle is 3.

Five control plates are needed to control and shut down the reactor. The regulating plate layout (central rod) minimizes the flux perturbation on the irradiation facilities and on the power peaking factor. The usage of burnable poison and the size of the regulating rod fulfil a design basis to control the operating cycle with only the regulating rod.

The layout of the irradiation facilities is shown in Fig. 3. Their position was optimized, maximizing the margins to fulfil the flux requirement and the flux perturbation between them.

The first core will be assembled with fresh FA and a very important design effort was carried out in the initial fresh core to have a similar performance. For this reason, different FA are used in the first core configuration. The only difference between FA is the U loading mass and the burnable poison. Only two additional types of FA are needed, and the fresh fuel assemblies loaded in the following refuellings are of the standard type.

Three types of fuel assemblies are used:

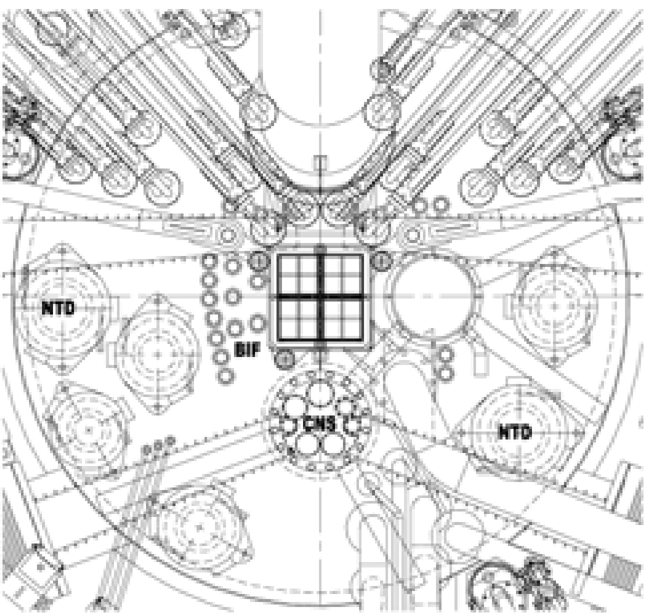


FIG. 3. Upper view of the reflector tank and main irradiation facilities.

TABLE 1. CORE REACTIVITIES

	First core		Equilibrium core	
State	BOC	EOC	BOC	EOC
Full power	4640	1270	4270	1830
Hot. No xenon	8150	4860	7960	5590
Cold. No xenon	8460	5190	8250	5890

— 212 Gr U^{235}/FA without burnable poison.
— 383 Gr U^{235}/FA with burnable poison.
— 484 Gr U^{235}/FA with burnable poison (standard design).

Figure 5 shows the FA distribution.

Table 1 summarizes the main equilibrium core reactivities (in pcm) for a simplified fuel management strategy, for both types of FA.

Table 2 shows a conceptual verification of the most important design criteria verification.

6.1.1. Flux irradiation facilities

This subsection gives neutronic fluxes per irradiation facility type. The values presented are for the BOC state of the U_3Si_2 FA type.

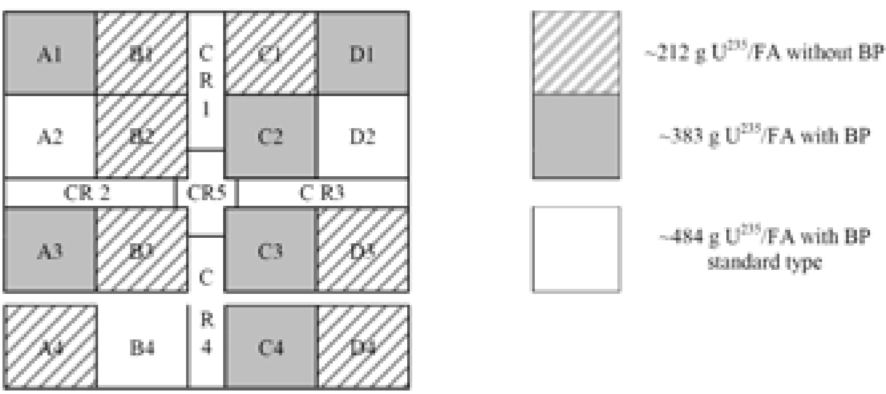


FIG. 4. FA distribution.

TABLE 2. SUMMARY DESIGN CRITERIA VERIFICATION

Design criteria	Limit	1st core	Eq. core
First Shutdown System (FSS): shutdown margin (SM)	≥ 3000 pcm	11730	9240
First Shutdown System: SM with single failure	≥ 1000 pcm	6070	4580
Second Shutdown System (SSS): shutdown margin	≥ 1000 pcm	6140	6270
Power peaking factor	= 3.0	2.42	2.30
End of cycle reactivity	≥ 1000 pcm	1270	1830
Actuation of FSS at 0.5 s	≥ 2000 pcm	11200	8860
Actuation of SSS At 15 s	≥ 3000 pcm	5290	5340

— *Bulk production irradiation facilities*: These facilities have 17 irradiation tubes with five targets each. Table 3 shows the maximum and minimum thermal flux per target and the average value for all the targets.

6.2. Pneumatic conveyor flux facilities

These facilities have 19 irradiation rigs with a different number of targets per facility (it ranges from 1 to 5 targets per rig). Table 4 shows the thermal flux (except FF: fast flux facility) for each level of flux requirement. It shows the maximum and minimum per rig.

The other requirements such as homogeneity are fulfilled.

TABLE 3. BULK PRODUCTION IRRADIATION FACILITY FLUXES

Value	1st core	Eq. core
	Very high flux facilities (2 tubes)	
Minimum	1.6E+14	1.6E+14
Maximum	3.2E+14	3.1E+14
Average	2.4E+14	2.3E+14
	High flux facilities (3 tubes)	
Minimum	1.0E+14	1.1E+14
Maximum	2.1E+14	2.1E+14
Average	1.5E+14	1.5E+14
	Medium flux facilities (12 tubes)	
Minimum	6.1E+13	6.2E+13
Maximum	1.4E+14	1.3E+14
Average	9.0E+13	8.8E+13

TABLE 4. PNEUMATIC IRRADIATION FACILITY FLUXES

Value	1st core	Eq. core	Value	1st core	Eq. core
LVL 7: 2 Rigs 10 Targets			LVL 6: 2 Rigs 6 Targets		
Minimum	1.1E+14	1.1E+14	Minimum	7.0E+13	6.9E+13
Maximum	1.4E+14	1.4E+14	Maximum	7.9E+13	7.6E+13
LVL 5: 2 Rigs 6 Targets			LVL 4: 2 Rigs 6 Targets		
Minimum	5.0E+13	5.0E+13	Minimum	3.1E+13	3.0E+13
Maximum	5.7E+13	5.5E+13	Maximum	3.5E+13	3.4E+13
LVL 3: 4 Rigs 12 Targets			LVL 2: 2 Rigs 6 Targets		
Minimum	1.5E+13	1.5E+13	Minimum	7.8E+12	8.0E+12
Maximum	1.7E+13	1.6E+13	Maximum	9.4E+12	8.9E+12
LVL 1: 1 Rigs 3 Targets			FF: 2 Rigs 6 Targets		
Average	3.4E+12	3.4E+12	Minimum	6.7E+12	7.1E+12
			Maximum	8.0E+12	8.2E+12
NAA: 1 Rigs 1 Targets			DNAA: 1 Rigs 1 Targets		
Average	2.7E+13	2.7E+13	Average	6.3E+12	6.3E+12

6.3. Large volume irradiation facilities

These facilities have six irradiation rigs and they are dedicated to the neutron transmutation doping. Table 5 shows the minimum and maximum thermal flux for all the facilities.

The other requirements, such as axial homogeneity and thermal to fast ratio, are fulfilled.

TABLE 5. NTD IRRADIATION FACILITY THERMAL FLUX

NTD: 6 Rigs		
Value	1st Core	Eq. Core
Minimum	2.9E+12	2.9E+12
Maximum	1.6E+13	1.6E+13

7. CONCLUSIONS

The first core of the replacement research reactor (RRR) has a safe and high performance core design, which fulfils all the irradiation fluxes and the operational requirements. The neutronic parameters of the first core are quite similar to the equilibrium core.

OPTIMIZATION CALCULATIONS AT TR-2

M.H. TURGUT, A. AYTEKIN
Turkish Atomic Energy Authority,
Çekmece Nuclear Research and Training Centre,
Istanbul, Turkey
Email: turgutm@nukleer.gov.tr

Abstract

The main objective of the optimization calculations at TR-2 is to increase radioisotope production. Reactor operation regimes and irradiation locations were investigated separately. A second objective of this study is to obtain similar activities in the irradiated samples irrespective of the irradiation positions. The effects of possible geometrical modifications in the irradiation devices have been investigated and some suitable new designs were proposed. This study also includes the maximization of the discharge burn-up levels of the HEU elements in a mixed HEU–LEU core, so both safe and economical utilization of the reactor is attained.

1. INTRODUCTION

The TR-2 is a swimming pool-type research reactor with maximum operational power of 5 MW. Unfortunately, it is operating at 300 kW for the moment, due to regulatory restrictions. The initial fuel have 23 plates per standard element of 93% enrichment. New silicide LEU fuel elements have been imported based on the core conversion studies made at ANL-USA [1, 2]. They have the same geometry as the HEU fuel elements. The initial core has been modified several times according to the irradiation and operational needs [3, 4, 5, 6].

Tc-99m and I-131 radioisotopes, which are widely used in hospitals and laboratories for diagnosis, and Ir-192, which is used in industry for non-destructive testing purposes, are produced by the irradiation of Mo, Te and Ir isotopes at TR-2. The activities of these radioisotopes depend on the neutron flux levels and irradiation regimes of the reactor. So, a series of investigations have been conducted on TR-2 to improve the production efficiencies of these radioisotopes. Time [7] and space [8, 9, 10] dependent analyses have been carried out independently.

2. MULTI-GROUP CROSS-SECTIONS AND COMPUTATIONAL TOOLS

A five group structure is used for the burn-up dependent cross-section libraries that are generated by the EPRI-CELL [11] code. The RABANL integral transport option of the MC^2-2 [12] code was used to accurately account for the resonance self-shielding of U-238. Transport corrected effective cross-sections were used for the control rod regions [13]. The data for Mo, Tc and Te isotopes were not available in this library, so new data were generated using GGC-4 [14] and ANISN [15] codes. In order to have a better understanding of the neutronic interactions, especially in the epithermal energy range, five and in addition nine group cross-section libraries of all the isotopes in the core have been generated with the aforementioned [14, 15] codes. The FUKMOD [16] code is used for the time dependent analyses. The 2D diffusion-depletion code GEREBUS [17] is used for the reactivity and burn-up calculations. Control rods CR-1 and/or CR-2 were assumed to be partially inserted in the burn-up calculations during each cycle in order to reflect the criticality adjustments made throughout the operations.

3. OPTIMIZATION CALCULATIONS

The irradiation of the samples has been made in the irradiation elements inside the core and in the water boxes placed around the core up to now. Since the flux at the periphery is much lower, the relevant activities are quite low and they differ considerably from one box to another. The radioisotope production department had difficulties in extracting the required amount of activity from the irradiated samples from time to time. Replacement of these samples from one position to another during the irradiation causes operational complications. So, two of the water boxes were moved inside the reactor core to overcome this kind of problem. In this way, eight new irradiation positions were obtained inside the core. The suitable positions were chosen according to the optimization calculations. In addition, new core designs and the effect of different operational regimes of the reactor were investigated in order to improve the radioisotope production.

3.1. Time dependent investigations

The TR-2 reactor is generally operated at 5 MW for 6 h/d, a few days per week. This is surely a very inconvenient way of operation for the purpose of radioisotope production. So, a series of calculations has been performed for different operating regimes in order to see the influences and efficiencies of each on the production rates. Fuel consumption and the savings in the target materials were also compared. Zero dimensional FUKMOD code has been used to calculate the time dependent activity changes for the aforementioned isotopes. Neutron flux is assumed to be constant ($= 1.0 \times 10^{13}$) during the irradiation. Self-shielding corrections were made to account for the flux depressions inside the samples. Calculation results agreed quite well with the measured values.

The considered operating regimes:

(1) 20 d continuous operation
(2) 2dO+1dS+2dO+1dS+2dO+1dS+.....+2dO+1dS+2dO
(3) 4dO+3dS+4dO+3dS+4dO+3dS+4dO+3dS+4dO
(4) 5dO+2dS+5dO+2dS+5dO+2dS+5dO
(5) (6hO+42hS+6hO+18hS+6hO+18hS+6hO+66hS) ×19+(6hO+42hS+6hO+18hS+6hO+18hS+6hO)
(6) 6hO+18hS+6hO+18hS+6hO+18hS+.....+6hO+18hS+6hO
(7) 42hO+6hS+42hO+6hS+42hO+6hS+.....+42hO+6hS+42hO
(8) 12hO+12hS+12hO+12hS+12hO+12hS+.....+12hO+12hS+12hO

where

dO : days of operation	hO : hours of operation
dS : days of shutdown	hS : hours of shutdown

The activities of 40 g TeO_2, 50 g MoO_3 and 1.813 g Ir have been calculated for a total of 20 d of irradiation. The results of different operation regimes are presented in Table 1. The time dependent changes of the isotope activities mentioned are given in Figs 1–3. The decrease in the activities during shutdown is not shown in the graphics. The activity levels are decreased considerably, especially for short-lived isotopes, at longer shutdowns, as can be seen from the figures. Saturation activities are also very much regime dependent, which means no matter how long one operates the reactor, the values of continuous operation can never be reached by intermittent regimes. Ir-192 production is less regime dependent due to a longer half-life (74.2d).

TABLE 1. ACTIVITY VALUES OF DIFFERENT ISOTOPES AFTER 20 DAYS OF OPERATION

Operation regime	Te-127	Te-129	Te-131	Activity Te-131m	[mCi] I-131	Mo-99	Tc-99m	Ir-192
1	6979	2065	3191	558	2958	1601	1600	96771
2	6815	2060	3170	464	2308	1202	1162	92951
3	6969	2058	3181	511	2186	1235	1197	91755
4	6976	2061	3186	536	2523	1392	1370	94218
5	3044	1980	3100	151	575	326	273	60062
6	3072	1991	3104	171	925	442	400	74972
7*	6862	2061	3182	518	2704	1446	1430	92462
8	4983	2053	3138	319	1782	856	813	88726

* Total operation time = 19 d + 6 h

3.2. Space dependent optimization

In this part, the most suitable irradiation places and core designs have been investigated independent of the operation regimes. Five main goals have been considered in these optimization calculations:

— Safe operation of the reactor;

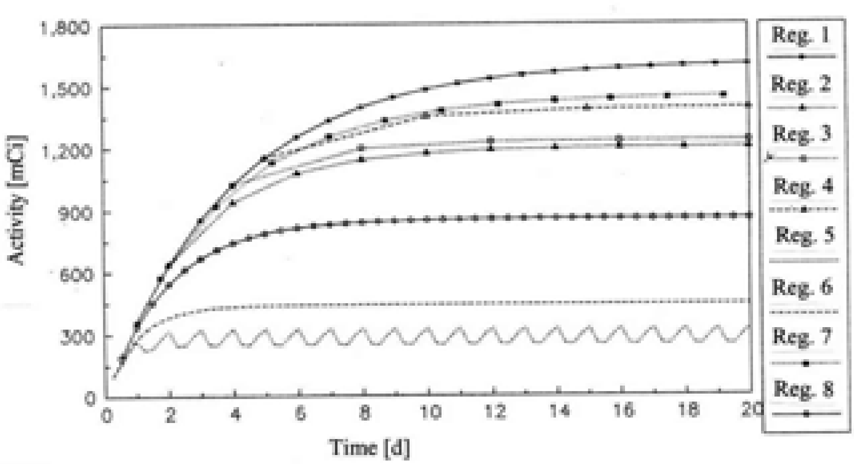


FIG. 1. Mo-99 activity changes for different operating regimes.

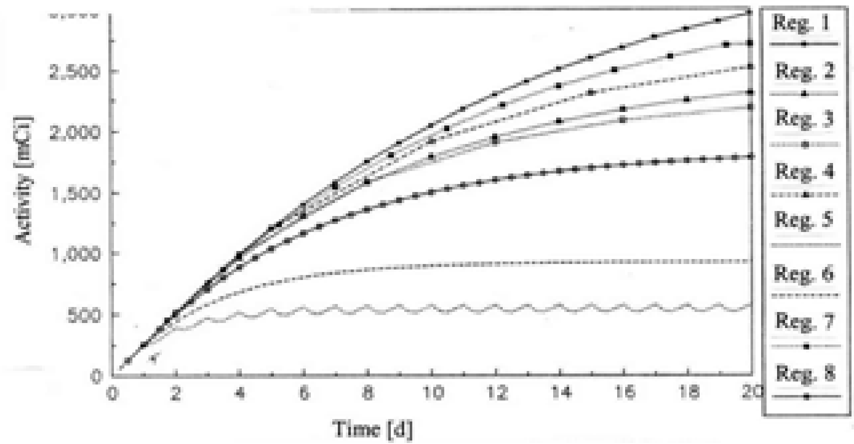


FIG. 2. I-131 activity changes for different operating regimes.

— Maximum yields in the activity levels;
— Similar activity values in all irradiation positions;
— Economical usage of the fuel;
— Facilitation of the reactor operation.

First, the best irradiation positions have been identified on the existing core (C13AD). Many new core designs and various irradiation positions have

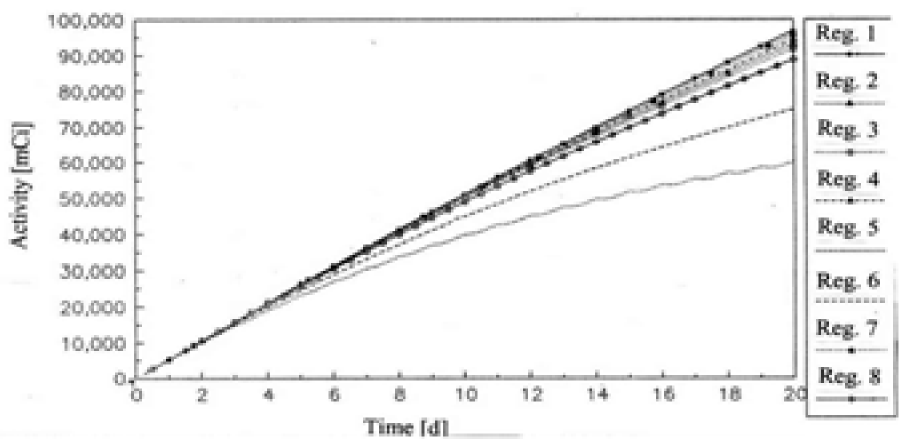
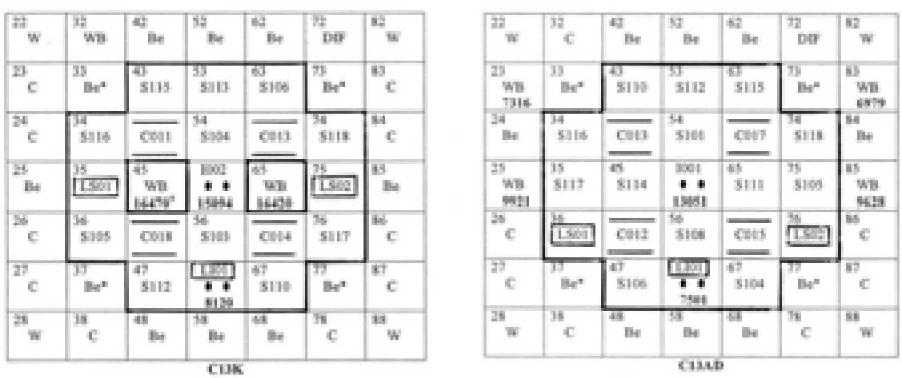


FIG. 3. Ir-192 activity changes for different operating regimes.

been investigated for the purposes mentioned. The reactor core was designed to be as compact as possible, in order to have higher flux values for the irradiation samples. The water boxes which were used for irradiation purposes up to now have been moved from the periphery to the most suitable positions inside the reactor core. This resulted in a decrease in the excess reactivity due to the reduction in the number of fuel elements. New graphite and Be reflectors have been added around the core to enhance the excess reactivity and the discharge burn-up levels. These modifications have yielded higher production rates and a uniform distribution in the activity levels of the irradiation samples.

The excess reactivities and related cycle lengths, control rod anti-reactivities, the values and positions of the power peaking factors (PPFs), and Mo-99 activities were calculated for every loading. The optimum positions found for the Tc-99m isotope turned out to be, again, the most suitable places for I-131 production. The nine group calculations gave higher activity values than the five group results, but the relative variations between different core positions remained the same, as could be expected. Using HEU and LEU fuel elements in a mixed core also introduced additional power peaking problems to be eliminated. This problem was considerably reduced by using an out-in loading philosophy for the fresh LEU elements.

The Mo activities at the water boxes and irradiation elements are also included in Fig. 4. The calculated and measured excess reactivities and the control rod worths for these loadings are given in Table 2. The absolute and relative thermal flux distribution are given in Fig. 5. The thermal flux profiles



*	Mo activity	W: Pool water	DIF: dry irra. fac.	I002: HEU irr. ele.
	C: Graphite block	WB: Water box	C011: HEU Control ele.	LS01: LEU stan. ele.
	Be: Be block	Be*: Be bl. with hole	S115: HEU Standard ele.	LI01: LEU Irr. el.

FIG. 4. TR-2 core loadings (C13K and C13AD) and calculated Mo activities.

TABLE 2. CALCULATED AND MEASURED REACTIVITIES FOR THE TWO CORE LOADINGS

	C13K		C13AD	
	Calculations	Measurements	Calculations	Measurements
Core excess reactivity [pcm]	3744	3720	6868	6867
SR-1 worth [pcm]	5822	4503	6219	4511
SR-2 worth [pcm]	5755	5231	5936	4294
CR-1 worth [pcm]	5847	3991	6051	4402
CR-2 worth [pcm]	5748	4146	5734	4394

along the axis passed over the water boxes are given for the two cases in Fig. 6. A large increase in the fluxes can be observed for the case C13K which contains two water boxes inside the reactor core. Relative Mo activities given in Table 3 also reflects this increase (66–130%). As can be seen from Tables 2 and 3, and from Fig. 5, the agreement between the calculations and the experiments is quite satisfactory. The deviations mainly come from:

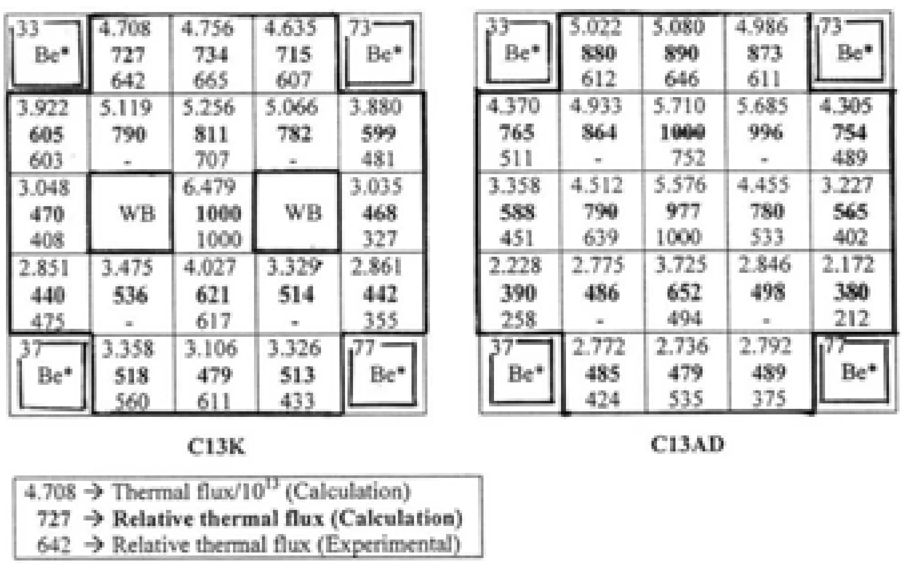


FIG. 5. Absolute and relative thermal flux distributions for the two core loadings.

TABLE 3. RELATIVE MO ACTIVITIES FOR THE TWO CORE LOADINGS

Calculation name			% increase			
C13K		C13AD	Calculations	Average	Measurements	Average
WB45	/	WB25	66.0	68.2	68.2	66.1
WB65	/	WB85	70.5		64.0	
WB45	/	WB23	125.1	130.2	—	—
WB65	/	WB83	135.3		—	
IE55	/	IE55	15.7	12.0	–8.0	3.4
IE57	/	IE57	8.3		14.8	

— The impossibility to reflect the actual daily movements of the control rods in the calculations;
— Calculation errors due to:
- Homogenization;
- Region averaged fluxes;
- Other assumptions;

— Experimental uncertainties and time dependent changes:
- Measurements in 3D geometry are control rod position dependent;
- Fluxes are measured only at one point for each element.

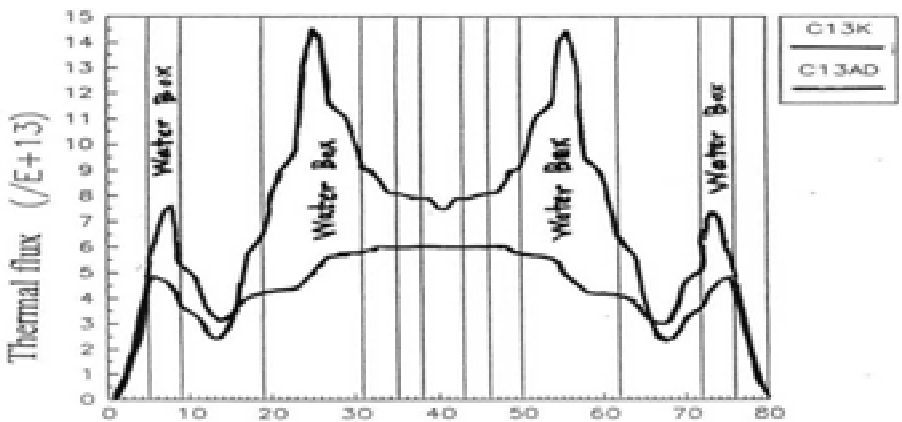


FIG. 6. Thermal flux profiles for C13K and C13AD loadings.

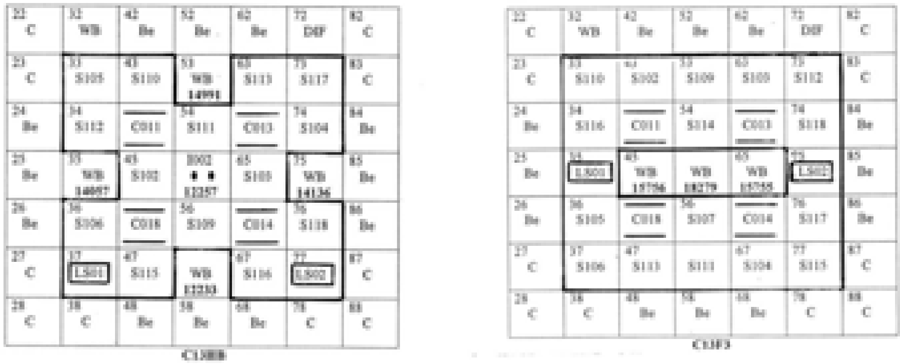


FIG. 7. TR-2 core loadings (C13HB and C13F3) and calculated Mo activities.

There are several other possibilities for the TR-2 core loading and the optimum is based on the utilization of the reactor. For the moment, C13K seems to be satisfactory for today's irradiation needs. C13HB (Fig. 7) is also a possibility for increased demands due to a large number of irradiation positions. There is one irradiation element in the middle, and the four water boxes are partially inserted into the core from four sides. C13F3 is another possibility for obtaining higher activities. There is no irradiation element in this loading, and the three water boxes are inserted into the middle of the core. This can be a long term solution after the usage of all irradiation elements. Decisions can be made according to the needs and conditions on a particular day.

4. NEW IRRADIATION ARRANGEMENTS

In addition to all the studies mentioned, the effect of possible geometrical modifications in the irradiation devices has been investigated. The thickness of the irradiation tubes were varied in order to see the self-shielding effects. The water thickness around the tubes were also changed to have better moderation. By reducing the tube thickness and increasing the water around it, higher activation values were obtained. An appropriate target material thickness lies around 1.0 cm, and this can bring an increase of nearly 25% in the production rates. It seems reasonable to make some modifications in the target tube and irradiation device geometries in order to bring these theoretical results into practice. Two different, suitable new designs were proposed for this purpose. In the first one (Fig. 8), the original tube design has not changed, only 1 cm diameter Al block or vacuum is placed in the middle of it. In this way, the target material thickness is reduced to 1.3 cm and it is placed between the two rings.

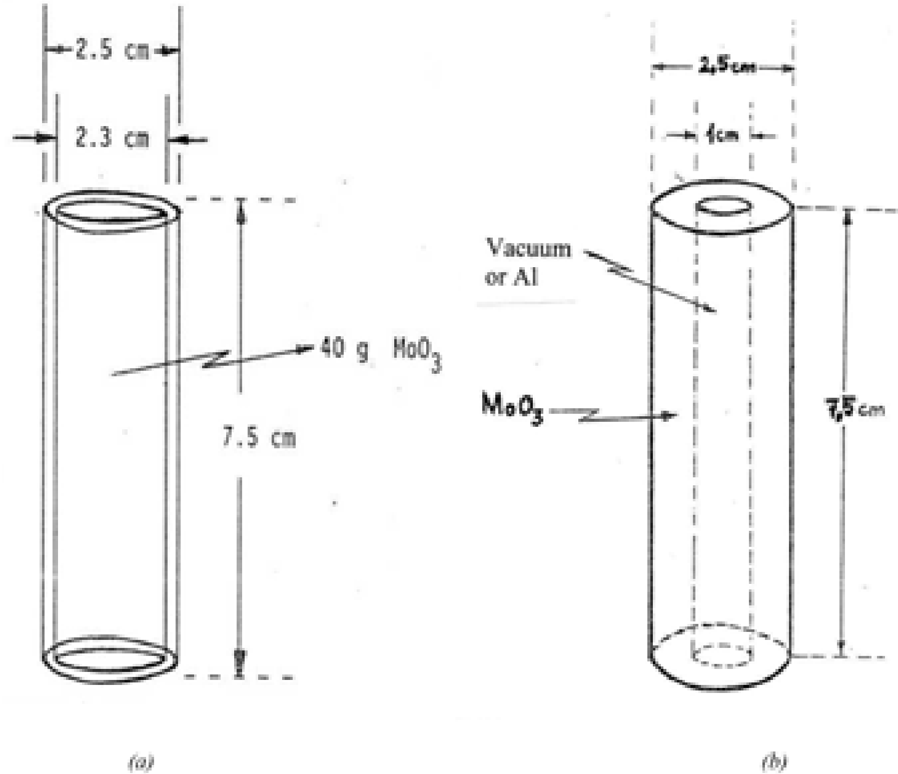


FIG. 8. The available and proposed tube geometries: (a) available irradiation tube; and (b) proposed irradiation tube.

Two tubes can be put on top of each other and higher efficiencies can be obtained for the same amount of target material. In the second proposal (Fig. 9), the outer diameter of the tube is reduced to 1.2 cm, and in this way, eight tubes can be placed inside the irradiation device instead of four. Again, two tubes can be put on top of each other and 16 tubes can be used in one irradiation device. More refined calculations can be made in order to find the optimum geometry for the irradiation devices.

5. CONCLUDING REMARKS

The results of these optimization calculations can be summarized as follows:

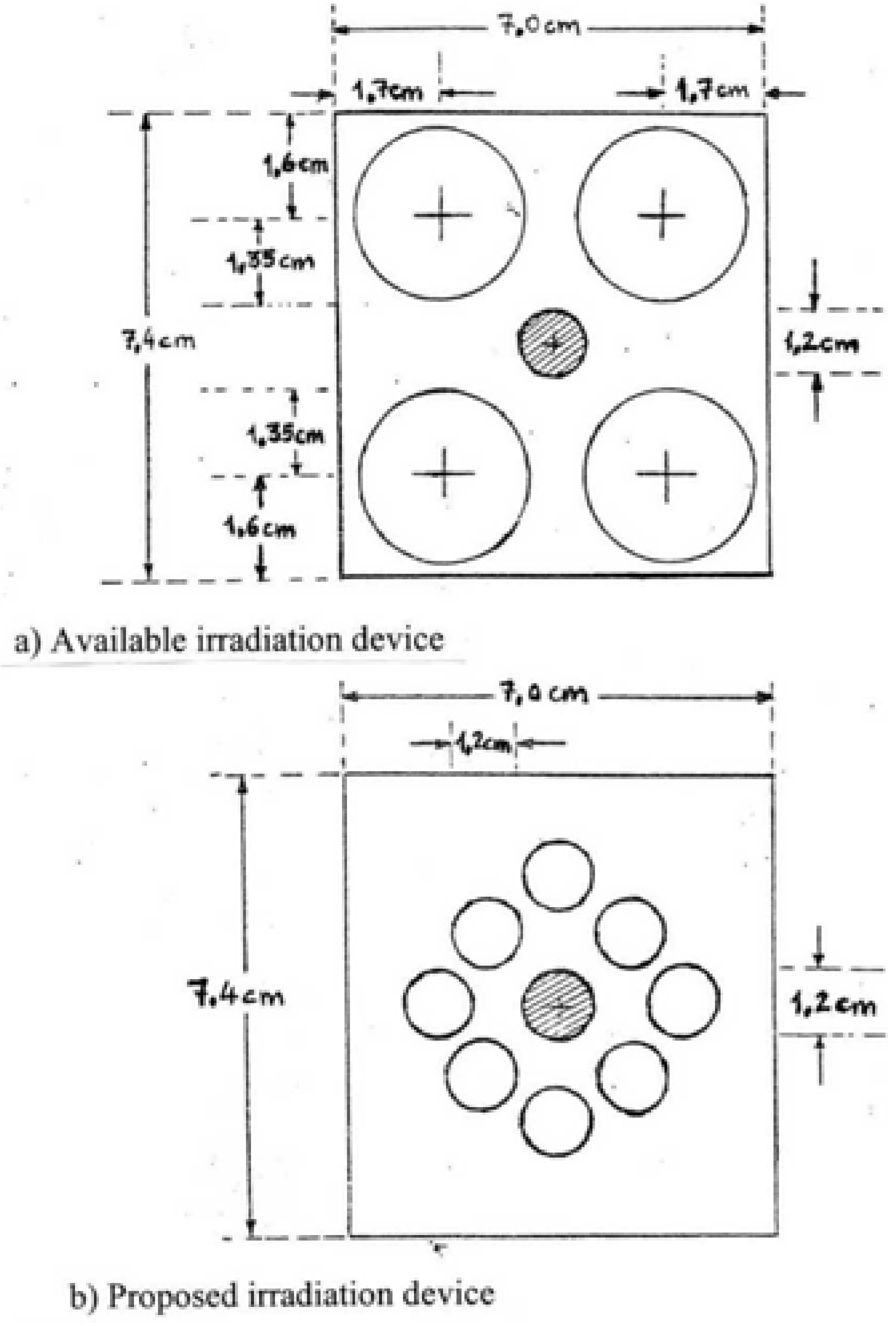


FIG. 9. The available and proposed irradiation device geometries: (a) available irradiation device; and (b) proposed irradiation device.

— The activity levels are decreased considerably, especially for short-lived isotopes, at longer shutdowns as can be expected.
— Saturation activity of I-131 is reached in 20 d, and of Mo-99 is reached in 10 d at continuous operation.
— Saturation activities of both isotopes are reached between 2 to 20 d, depending on which intermittent regime is used.
— Saturation activities are different for every regime.
— No matter how long one operates the reactor, the saturation activities of continuous operation can never be reached by intermittent regimes.
— Ir-192 production is less regime dependent due to a longer half-life.
— Fuel consumption for the same activation levels are very much regime dependent. Fuel savings are up to 79% for short-lived, and up to 30% for long-lived isotopes at continuous operation.
— The same activities can be obtained by less target materials at higher fluxes. The savings at target materials were calculated to be up to 60%.
— Space dependent optimization studies have yielded higher production rates and, in addition, a uniform distribution in the activity levels of the irradiation samples.
— Radioisotope production can still be improved by new core designs. Especially insertion of water boxes inside the reactor will increase the yields up to 66–130%. This is also verified by the experiments.
— The nine group calculations give higher activity values than the five group results, but the relative variations between different core positions remains the same, as could be expected.
— The optimum positions found for the Tc-99m isotope turned out to be, again, the most suitable places for the I-131 production.
— The activity levels of both isotopes are higher at inner water boxes then irradiation elements. So, there is no need to buy new LEU irradiation elements for the production of these isotopes.
— Replacing Be and graphite reflectors around the core increases the production rates and also the discharge burn-up levels.
— The discharge burn-up levels of the HEU elements can further be increased only by the mixed core loading.
— The PPFs become more important in mixed core analysis. Loading patterns should be carefully adjusted for each step for the minimization of the peaks until the full conversion to LEU equilibrium core is reached.
— The agreement between calculations and the experiments seems to be reasonable.
— No operational or safety related problems were observed during the initial phase of the mixed core during 371.9 MWDs of operation.

— Irradiation efficiency can further be improved (up to 25%) by changing the geometry of the irradiation devices. Two proposals are made, but more refined calculations are suggested for a final optimum design.
— There are many loading possibilities for the TR-2, and the decision can be made according to irradiation needs and/or other utilization factors.

REFERENCES

[1] ALDEMIR, T., TURGUT, M.H., BRETSCHER, M.M., SNELGROVE, J.L., A Feasibility Study Concerning the Conversion of the TR-2 Reactor from Using Highly Enriched Uranium to Light Enriched Uranium, Rep. NAEM-R-217, NAEM, Istanbul (1982).

[2] TURGUT, M.H., TRKER, T., Core Conversion Calculations for the TR-2 Reactor; Part A: Neutronics Calculations, Rep. NAEM-AR-308, NAEM, Istanbul (Feb. 1993).

[3] TURGUT, M.H., Neutronic Calculations of the TR-2 Reactor Present Core, NAEM Technical Report No. 30, NAEM, Istanbul (Sept. 1986).

[4] TURGUT, M.H., Density Search Calculations for the Silicide Fuels, NAEM Technical Report No. 37, NAEM, Istanbul (Dec. 1986).

[5] TURGUT, M.H., Optimization of the Usage of the Remaining HEU Fuel, NAEM Technical Report No. 42, NAEM, Istanbul (Sept. 1987).

[6] TURGUT, M.H., STN, G., Calibration Calculations for the TR-2 Reactor, Rep. NAEM-AR-245, NAEM, Istanbul (Apr. 1988).

[7] STN, G., ADALIOLU, U., Isotope Irradiation Regimes for the TR-2 Reactor, Rep. NAEM-AR-297, NAEM, Istanbul (Jan. 1992) (in Turkish).

[8] TURGUT, M.H., AYTEKIN, A., Space Dependent Analysis to Enhance Radioisotope Production, Rep. NAEM-AR-309, NAEM, Istanbul (Apr. 1993) (in Turkish).

[9] AYTEKIN, A., TURGUT, M.H., ADALIOLU, U., Tc-99m Production at TR-2, Rep. NAEM-AR-318, NAEM, Istanbul (July 1994) (in Turkish).

[10] AYTEKIN, A., TURGUT, M.H., ADALIOLU, U., I-131 Production at TR-2, Rep. NAEM-AR-319, NAEM, Istanbul (Sept. 1994) (in Turkish).

[11] ZOLOTAR, B.A., et al., EPRI-CELL code description, Advanced Recycle Methodology Program System Document, Part II, Chapter 5, Electric Power Research Institute (Sept. 1977).

[12] HENRYSON III, H., TOPPEL, B.J., AND STENBERG, C.G., MC^2-2: A Code to Calculate Fast Neutron Spectra and Multi-group Cross-Sections, Argonne National Laboratory Report, ANL-8144 (June 1976).

[13] BRETSCHER, M.M., Blackness Coefficients, Effective Diffusion Parameters, and Control Rod Worths for Thermal Reactors, Argonne National Laboratory Report, ANL/RERTR /TM-5 (Sept., 1984).
[14] ADIR, J., AND LATHROP, D., Theory of Methods Used in GGC-4 Multi-group Cross Section Code, Gulf General Atomic, GA-9021 (Oct. 1, 1968).
[15] ENGLE, W.W., JR., A Users Manual for ANISN: A One Dimensional Discrete Ordinates Code With Anisotropic Scattering, Union Carbide Corporation, Nuclear Division, K-1693 (March 30, 1967).
[16] STN, G., TURGUT, M.H., ADALIOLU, U., FUKMOD: A Multi-purpose Code for the Calculation of Radioactive Isotope Chains, (in Turkish) NAEM-AR-272 (Dec., 1989).
[17] CONSOLE, M., DANERI, A., AND SALINA, E., EREBUS: A Multi-group Diffusion Program in two Dimension, FN-E-88 (FIAT, 1967) (GEREBUS is GKSS version of EREBUS code).

CORE MANAGEMENT OF THE DALAT NUCLEAR RESEARCH REACTOR

PHAM VAN LAM, LE VINH VINH, HUYNH TON NGHIEM,
NGUYEN KIEN CUONG, LUONG BA VIEN
Nuclear Research Institute,
Dalat, Vietnam
Email: phvlam@hcm.vnn.vn

Abstract

The Dalat nuclear research reactor (DNRR) is a pool-type research reactor which was reconstructed from the 250 kW TRIGA-MARK II reactor. The reconstructed reactor reached its initial criticality in November 1983 and attained its nominal power of 500 kW in February 1984. The DNRR uses Russian fuel assemblies, type WWR-SM. The first fuel reloading was executed in April 1994 after more than ten years of operation with 89 fuel assemblies. Research on core management of DNRR with the purpose of maintaining safe operation and effective utilization of reserve fuel assemblies is being carried out at the Nuclear Research Institute. Calculations of fuel burn-up for the Dalat nuclear research reactor are carried out based on the cell calculation program WIMS and two diffusion calculation programs HEXAGA and HEXNOD. Experimental measurement of fuel burn-up for the Dalat nuclear research reactor was realized by a measurement method of long-life isotopes from fission products. Optimum second fuel reloading and future refuelling for DNRR have been gained. A second fuel reloading for the Dalat nuclear research reactor was realized in March 2002. After reloading the working configuration of the reactor, the core consisted of 104 fuel assemblies. Research results for future refuelling for DNRR show that with 36 reserve fuel assemblies, the reactor will be operated for at least 17 851 h at nominal power since the second fuel reloading.

1. INTRODUCTION

The DNRR is a pool-type reactor, moderated and cooled by light water. It was upgraded from the TRIGA Mark-II reactor built in the early 1960s. During 1982–1983, the reactor was reconstructed. Some structures of the former reactor, such as the reactor aluminium tank, the graphite reflector, the thermal column, the horizontal beam tubes and the radiation concrete shielding were retained [1]. The reactor core, the control and instrumentation system, the primary and secondary cooling systems, as well as other associated systems, were newly designed and installed. The natural convection mechanism

TABLE 1. REACTOR SPECIFICATIONS

Reactor type	Swimming pool
Nominal thermal power	500 kW
Neutron flux (thermal, max.)	2.2×10^{13} neutrons/cm^2.s
Coolant and moderator	Light water
Reflector	Graphite, beryllium and water
Fuel type	WWR-SM, U-Al alloy, 36% enrichment
Number of control rods	7 (2 safety rods, 4 shim rods, 1 regulating rod)
Control rod material	B_4C for safety and shim rods, stainless steel for automatic regulating rod
Neutron measuring channels	9 (6 CFC, 3 CIC)
Vertical irradiation channels	4 (neutron trap, 1 wet channel, 2 dry channels)
Horizontal beam-ports	4 (1 tangential, 3 radial)
Thermal column	1

of light water for reactor core cooling was kept unchanged. The reactor core, positioned inside the graphite reflector, is suspended from above by an inner cylindrical extracting well in order to increase the cooling efficiency for coping with the higher thermal power of the reactor. The renovated reactor achieved first criticality in November 1983 with 69 fuel assemblies and attained its planned nominal power of 500 kW in February 1984. Main reactor specifications of the DNRR are shown in Table 1. The vertical section of the reactor is shown in Fig. 1 and the cross-section view of the reactor core is shown in Fig. 2.

The DNRR is operated mainly in continuous runs of 100 h, once every four weeks, for radioisotope production, neutron activation analyses, training and research purposes. The remaining time between two consecutive runs is devoted to maintenance activities and also to short runs.

Reactor control and protection are effected by six control rods composed of boron carbide (two of which are safety rods and the other four are shim rods), and an automatic regulating rod composed of stainless steel. The core of the reactor utilizes fuel of aluminium-uranium alloy of Soviet-designed standard type WWR-SM, enriched to 36%, clad in aluminium. The fuel assembly is shown in Fig. 3. Each fuel assembly contains about 40.2 g of U-235 distributed on three coaxial fuel tubes (fuel elements), of which the outermost one is hexagonal shaped and the two inner ones are circular. The fuel layer with a thickness of 0.7 mm is wrapped between two aluminium alloy cladding layers of 0.9 mm thickness.

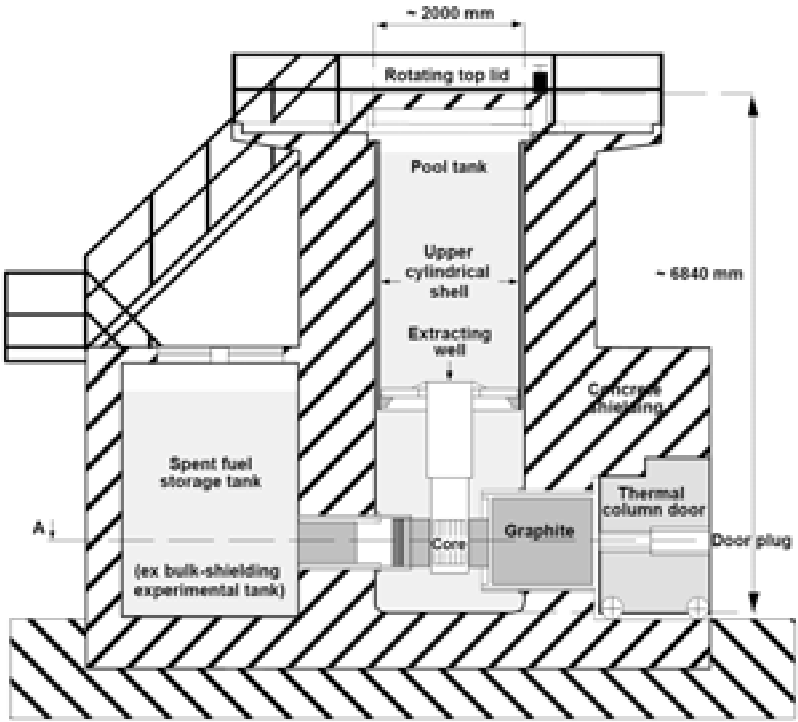


FIG. 1. Vertical section view of the reactor.

2. REACTOR CORE EVOLUTION

In April 1994, after more than 10 years of operation with 89 fuel assemblies, the first fuel reloading was executed. The 11 new fuel assemblies were added in the core periphery, at previous beryllium element locations. After reloading, the working configuration of the reactor core consisted of 100 fuel assemblies. The reloading operation increased the reactor excess reactivity from \$3.8 to \$6.5. Research on core management of DNRR is being carried out in the framework of a research theme in the years 2000–2001 [2]. Calculations of fuel burn-up for the Dalat nuclear reactor are carried out based on cell calculation program WIMS and two diffusion calculation programs HEXAGA and HEXNOD. Fuel burn-up in November 2000 is shown in Fig. 4. Experimental measurement of fuel burn-up for the Dalat nuclear reactor was realized by a measurement method of long-lived isotopes from fission products [3]. The

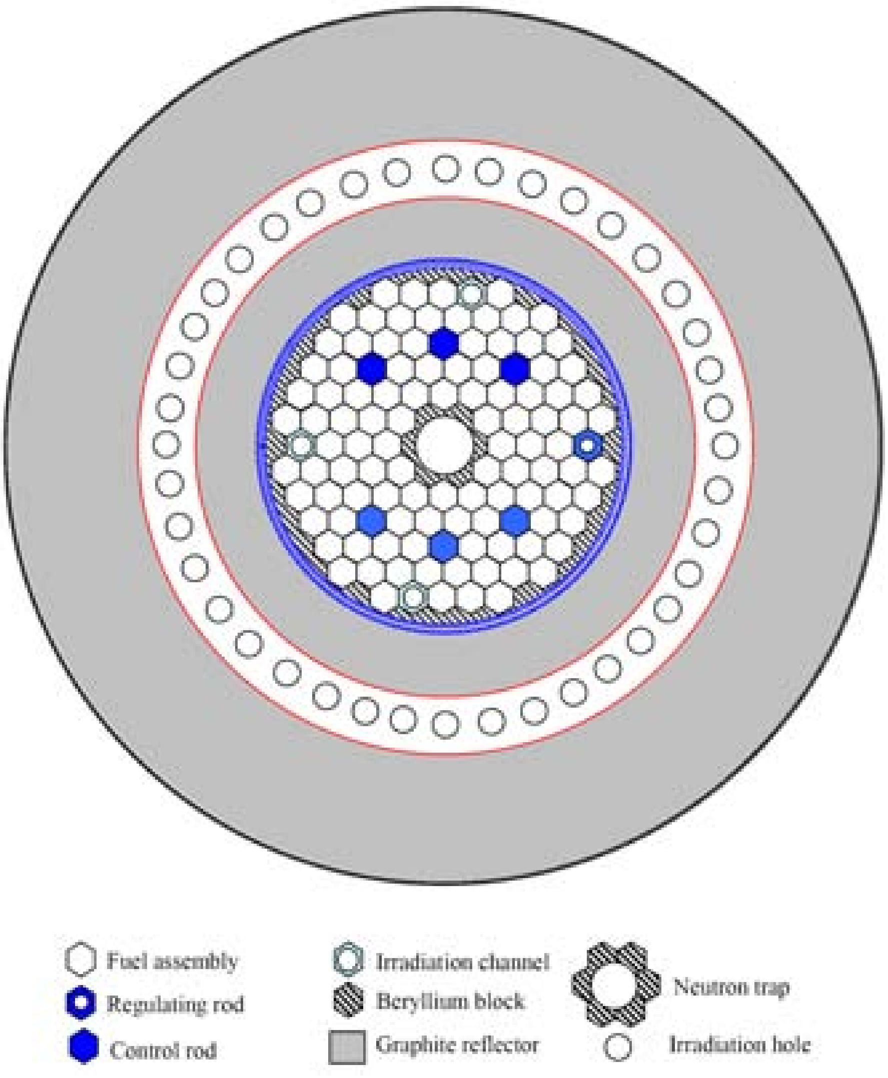


FIG. 2. Cross-section view of the reactor core.

measuring system is shown in Fig. 5; Fig. 6 presents a gamma spectrum of a fuel assembly.

The burn-up measured for 68 fuel assemblies in the reactor in November 2000 based on ^{137}Cs activity distribution is shown in Fig. 7 [3].

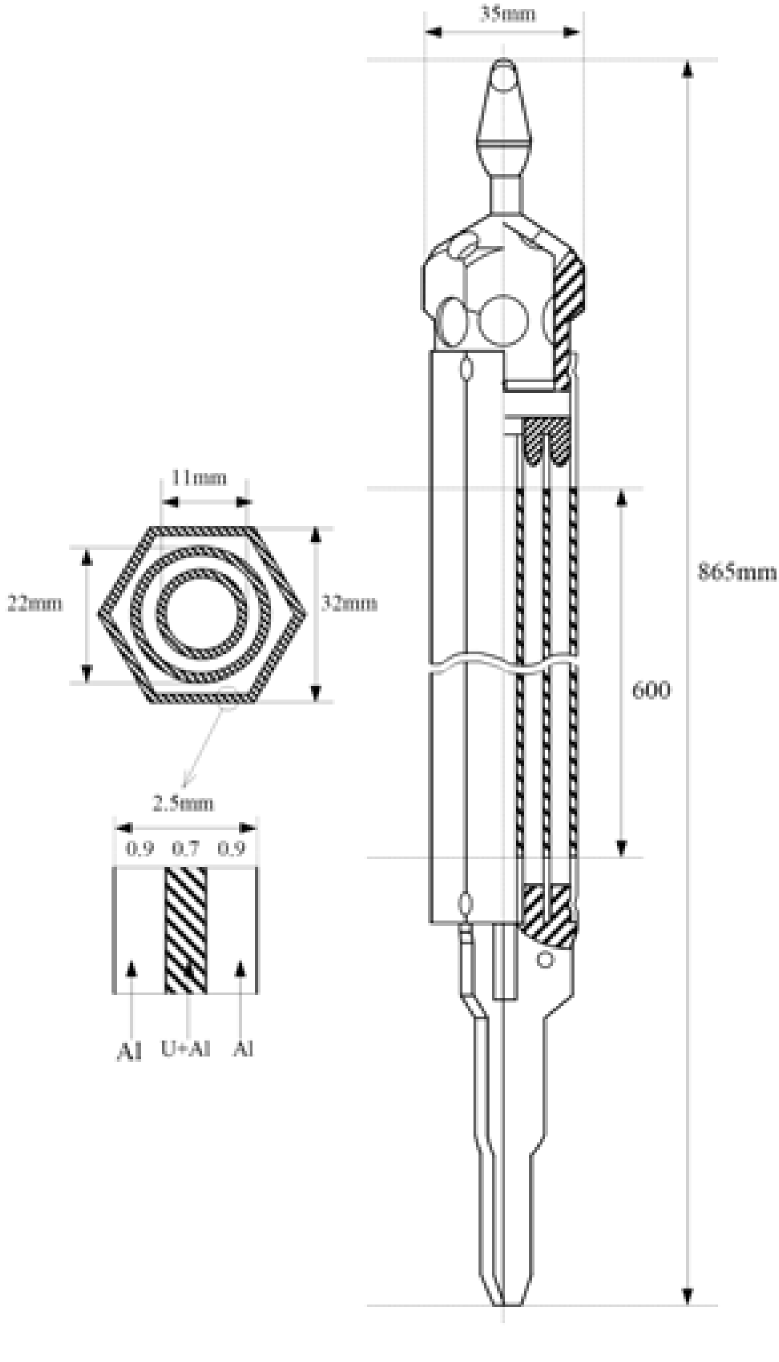


FIG. 3. Fuel assembly type WWR-SM of the DNRR.

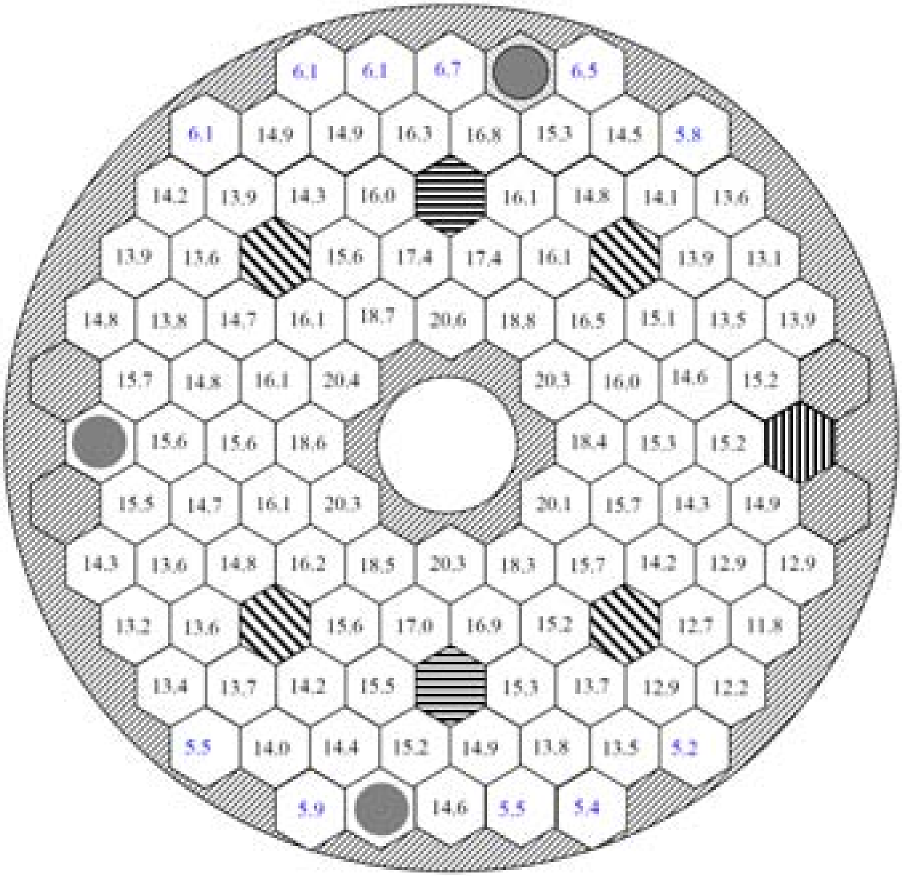


FIG. 4. Fuel burn-up distribution in March, 2002.

Optimum second fuel reloading for DNRR and strategy utilization of fuel assemblies based on fuel burn-up distribution were gained from calculation and experimental measurement.

The second fuel reloading was executed in March 2002 [2]. The four new fuel assemblies were added in the core periphery, at previous beryllium element locations. After reloading, the working configuration of the reactor core consisted of 104 fuel assemblies as shown in Fig. 2. The reloading operation increased the reactor excess reactivity from \$2.7 to \$3.8. This fuel reloading will ensure efficient exploitation of the DNRR for three years with 1200–1300 h/a at nominal power. By realizing this pattern, fuel burn-up of spent fuel assemblies taken out from the reactor in the next refuelling will be

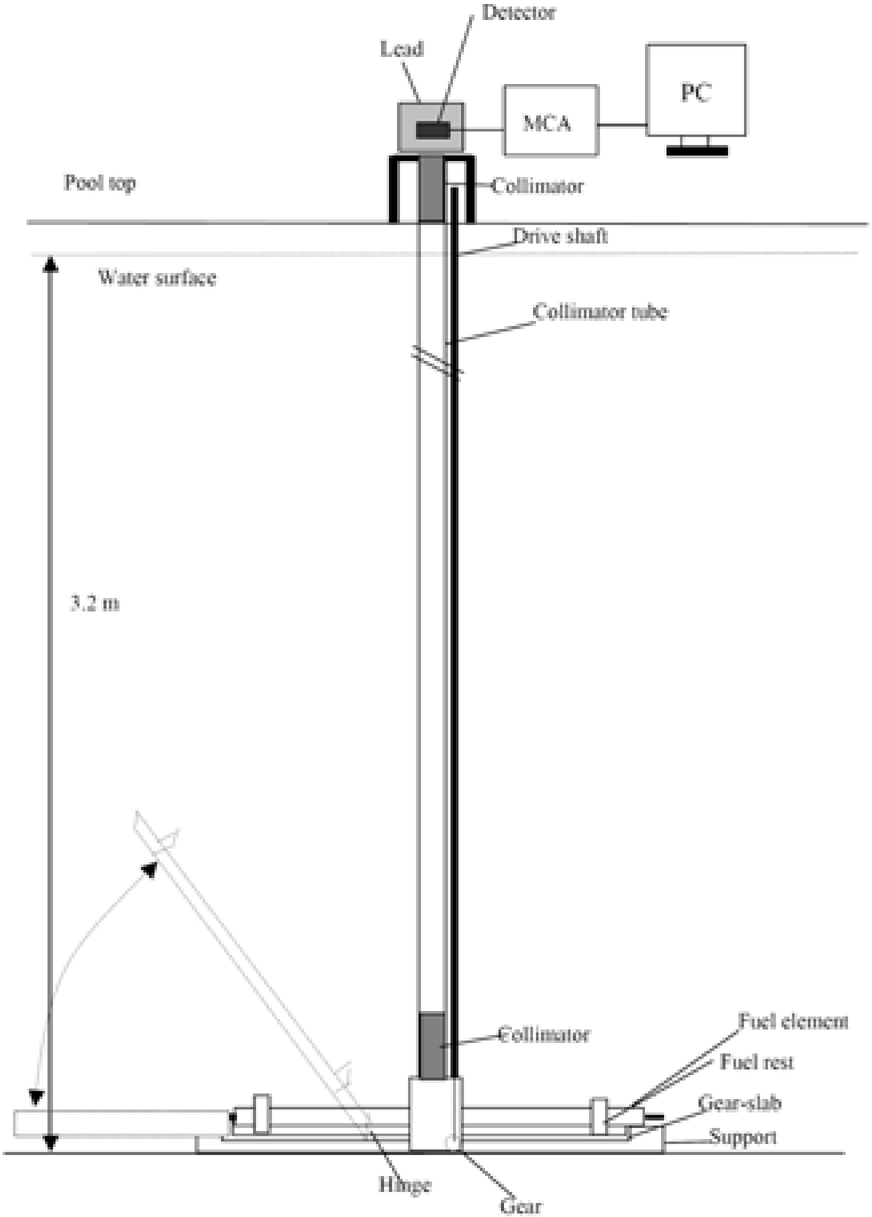


FIG. 5. Measuring system.

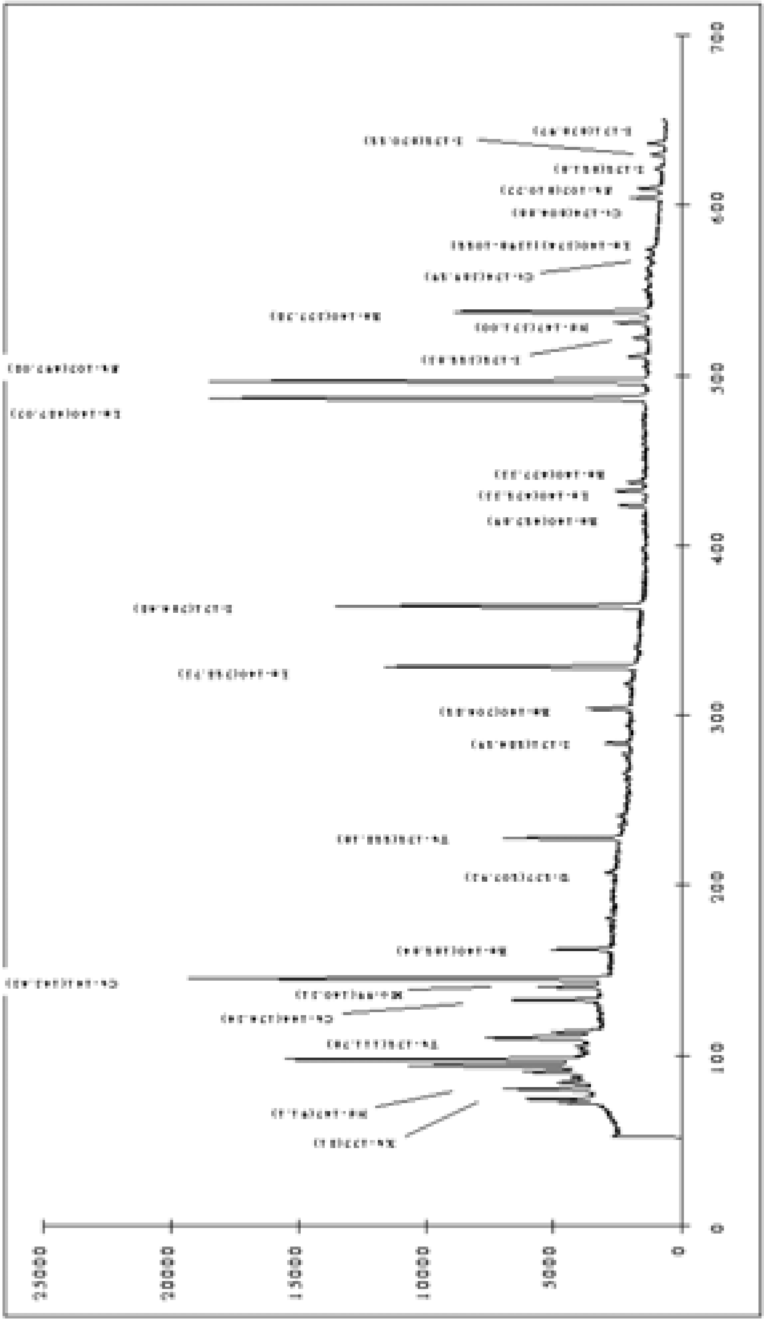


FIG. 6(a). *Dalat reactor fuel gamma spectrum measured after 17 days cooling (up to 650keV).*

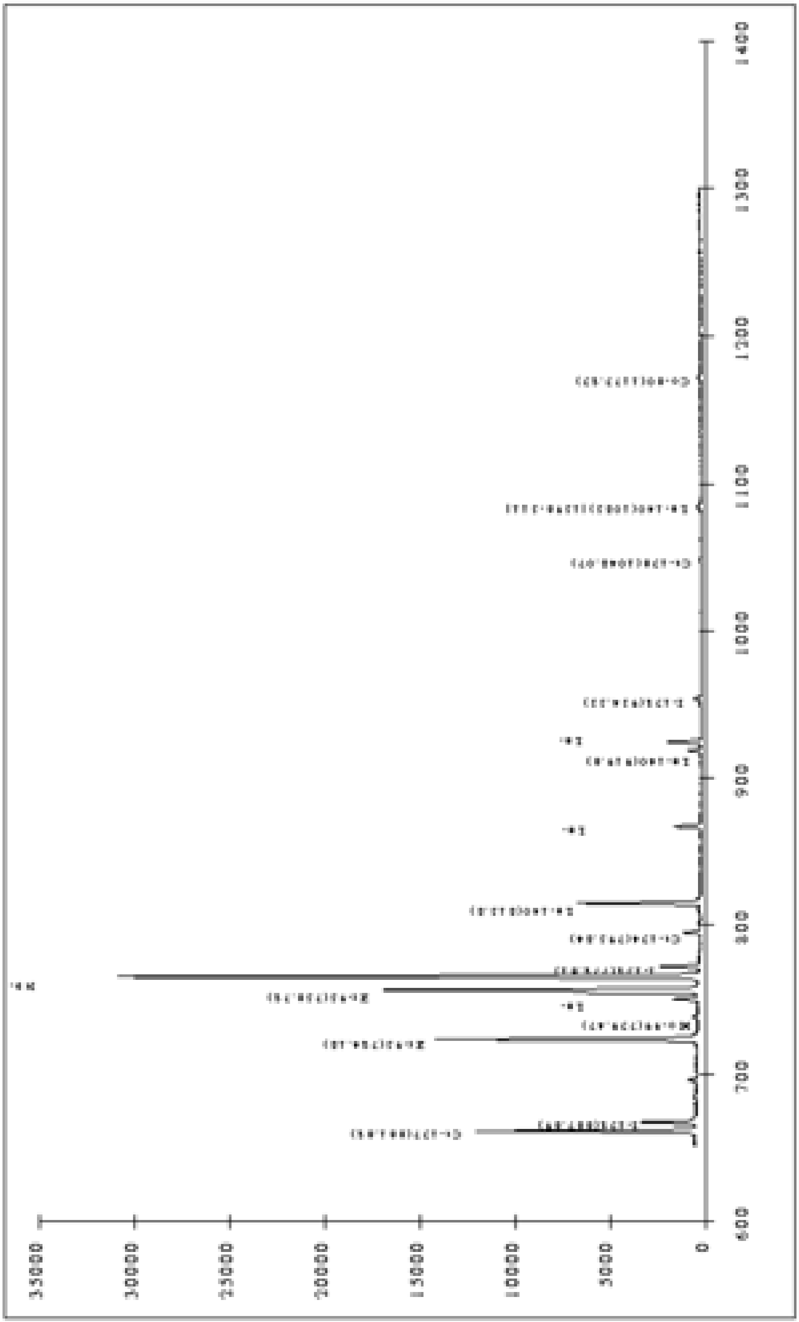


FIG. 6(b). Dalat reactor fuel gamma spectrum measured after 17 days cooling (from 650 to1300keV).

increased. Measured parameters of reactor core after refuelling show that the reactor will be operated safely from the nuclear and thermal safety points of view.

Calculation results for future refuelling for the DNRR show that with 36 reserve fuel assemblies, the reactor will be operated for at least 17 851 h at nominal power since the second refuelling in March 2002 [2]. This will ensure exploitation of the DNRR for about 15 years with 1200 h/a at nominal power.

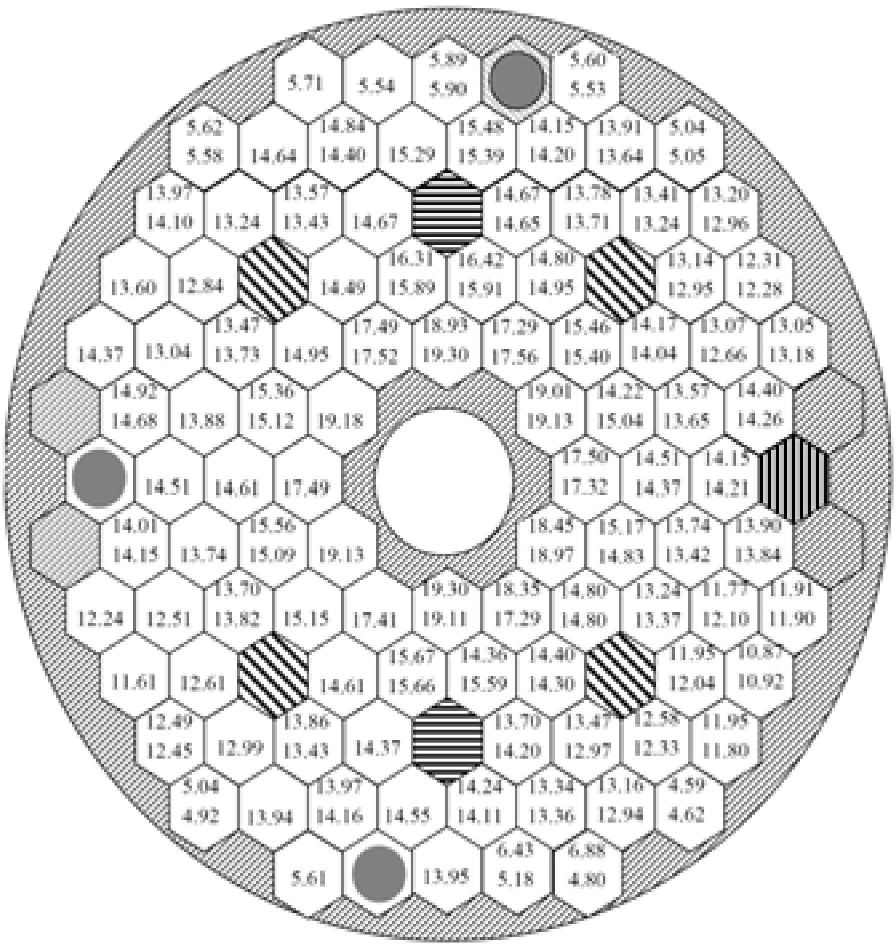


FIG. 7. Fuel burn-up distribution in November 2000 (upper values are experimental data and lower values are calculated data).

3. CONCLUSION

Research on the core management of DNRR with the purpose of maintaining safety operation and effective utilization of reserve fuel assemblies is being carried out at the Nuclear Research Institute. Calculations of fuel burn-up and burn-up distribution for DNRR are carried out based on the cell calculation program WIMS and two diffusion calculation programs, HEXAGA and HEXNOD. Experimental measurement of fuel burn-up for DNRR was realized by the measurement of long-lived isotopes from fission products. Optimum second refuelling pattern and future refuelling for DNRR have been gained from calculations and experimental research.

Research results for future refuelling for DNRR shows that with 36 reserve fuel assemblies, the reactor will be operated for at least 17 851 h at nominal power since the second refuelling in March 2002. This will ensure exploitation of the DNRR for about 15 years with 1200 h/a at nominal power.

In research on the core management of DNRR, other codes such as MCNP, SRAC and RELAP are being used.

REFERENCES

[1] TRAN HA ANH, et al., "Main experiences in renovation of the Dalat nuclear research reactor", Int. Group on Research Reactors (Proc. 5th Mtg, Aix-en-Provence, November 1996).

[2] Ministerial Research Theme on Core Management of Dalat Nuclear Research Reactor, Final Rep. No. BO/00/01-01, Dalat (April 2002) (in Vietnamese).

[3] LE VINH VINH, et al., "Application of gamma-scanning method for fuel burn-up determination for the Dalat nuclear reactor", Proc. 4th Symp. on Nuclear Physics and Technology, Hanoi, 2001 (in Vietnamese).

BLANK

REACTOR SIMULATOR DEVELOPMENT FACILITY FOR OPERATING PERSONNEL TRAINING

A. ETCHEPAREBORDA, C.A. FLURY,
F. MACIEL, D. ALEGRECHI, G. IBARRA, M. MUGUIRO
Grupo Control de Procesos, Centro Atómico Bariloche,
Comisión Nacional de Energía Atómica (CNEA),
Bariloche
Email: etche@cab.cnea.gov.ar

F. LEMA
Instituto Balseiro,
Bariloche

M. GIMÉNEZ, M. SCHLAMP, A. VERTULLO
Grupo Seguridad, Centro Atómico Bariloche,
Comisión Nacional de Energía Atómica (CNEA),
Bariloche

Argentina

Abstract

A reactor simulator development facility (FARSim) for operator training is presented. This facility is a software product that can be divided into four main modules: the model manager (MM), the simulator human machine interface (SHMI), the instructor station (IS) and the simulation manager (SM). It is designed as a distributed system where each module takes charge of a specific simulator task that could run in the same computer or distributed in a computer network. The main module is the SM which is responsible for routing the messages between the other modules and managing the simulation. The MM interfaces to the plant mathematical model (PMM). The IS is the process where the instructor commands the simulation, performing tasks such as start, pause and stop the simulation. The SHMI is the interface with the simulator SCADA (which can be identical to the plant SCADA) and is used by the trainee to observe the simulated plant output and to act upon it. The PMM is built using Matlab-Simulink simulation engine and graphical design user interface, for which specific libraries have been developed with a comprehensive set of nuclear and thermohydraulic plant components. This simulator development facility is being used to develop the ANSTO replacement research reactor full-scope and partial replica reactor training simulator (RTS).

1. INTRODUCTION

Current trends in simulator technology are leading to a reduction in the simulator costs. The design of adequate user interfaces that relieves the simulation engineer of many software engineering problems, and the introduction of simulation objects that can be reused and grouped to form complex systems lead to a reduction in the simulator development time, and ease all the simulator development process, from the definition of the data input to the maintenance and modification process [1]. Several nuclear plant simulator development environments have been designed following these objectives [2, 3, 4], with large investments in development, maintenance and actualization of these tools. The main features of these are:

(a) Documentation facilities are included for an adequate maintenance. Information about links between the data input (plant design data) and the simulator data, including parameter tuning information.
(b) Facilities for the creation of objects that represent components and for the integration of the components into plant subsystems, with facilities for parameterization based on plant design data.
(c) Early testing of components and subsystems before its integration into complex systems.
(d) Documentation facilities, versions management, users' privileges.

These features are usual in general purpose low cost simulation development environments such as Matlab-Simulink [5] or Dymola [6]. The facilities for development, testing, documentation and version management are similar, and they provide tools for the implementation of several modelling techniques, normally through general purpose basic components libraries.

The approach selected in the reactor simulator development facility (FARSim) model manager (MM) is to use Matlab-Simulink (MS) and Real-Time Workshop (RTW) [5] as the plant mathematical model (PMM) development environment, and to develop, based on its basic components libraries, specialized libraries with several water-air-vapour, nuclear and electrical components. In this way, all the investments in development, maintenance and actualization of the simulation development environment are reduced to the maintenance and actualization of the specialized library of components. The models are developed on graphical windows interfaces based on component libraries, where they can be easily constructed, debugged, tested and compiled to C source code that is embedded into the MM. In this way, the simulation engineer has all the graphical model development facilities provided by MS, such as a CAD-like user interface, flowsheet P&I-like diagrams, components menu dialogs, division of a large process into several flowsheet diagrams, browsing through the process, HTML-like helps on components, proprietary models embedding, snapshots saving, several variables display facilities, parameterization of components with automatic link to a plant database. All these features ease the model maintenance and modification process. The simulation engineer does not need to be involved with the flow network solution algorithms. The simulator end user does not need the MS, and RTW user licences, which lowers the costs of the simulator. The components libraries developer is only involved with the solution algorithms of the flow networks and with the verification and validation of these components. There is no need to maintain or update the simulation engine and user interface, lowering the costs of maintenance of this software product. The general features of FARSim are presented in the following sections.

2. DESCRIPTION

The product can be divided into four main modules: the model manager (MM), the simulator human machine interface (SHMI), the instructor station (IS) and the simulation manager (SM). It is designed as a distributed system where each module takes charge of a specific simulator task that could run in the same computer or distributed in a computer network.

The main module is the simulation manager (SM) which is responsible for routing the messages between the other modules and managing the simulation. The model manager (MM) interfaces to the plant mathematical model (PMM). The database manager (DBM) handles the database, in order to access and save necessary information during the simulation. The timing of the simulation is accomplished by the clock manager (CM) and the error and system messages are handled by the logger. The instructor console (IC) is the process where the instructor commands the simulation, performing tasks such as start, pause and stop the simulation. The SHMI is a process interface with the simulator SCADA (which can be identical to the plant SCADA) and is used by the trainee to observe the simulated plant output and to act upon it.

The PMM source code, together with the necessary libraries is encapsulated into the MM to implement the model initialization and the one-step simulation. The MM provides the initial conditions, and the model input, the PMM provides the plant output. Depending on the complexity of the model, it can be divided and the calculations can be distributed among different computers.

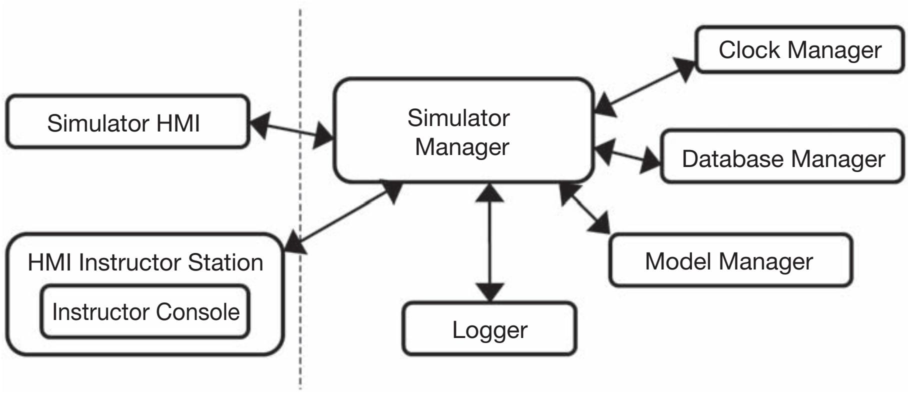


FIG. 1. Main software modules architecture.

The SHMI encapsulates the simulator SCADA, performing the communication of all the inputs/outputs with the SM and controlling the timing and initialization of the different SCADA components (trend charts, control logics, mimics) to implement the different simulation velocities, the backtrack and the snapshot.

The main functions that the instructor can perform using the IS are load initial conditions (including different core life cycles and plant states); start the simulation; 'pause' the simulation; set the simulation mode on slow, fast or real time; restart the simulation; store snapshots (to be used as ICs); backtrack the simulation; replay the simulation; edit, define, save, load and activate malfunction sequences; and execute local actions upon trainee request, such as local variables inspection or equipment startup. The access to the local variables and equipment is through navigation on system hierarchies using plant tags. It also prints out-of-scope messages when the simulation runs out of the model limits. All these actions performed during a simulation session in the IS can be stored and further reloaded.

3. MODEL MANAGER

The MM interfaces to the plant mathematical model (PMM). Proprietary models can be embedded directly into the MM or into MS.

The different plant process models are developed on graphical windows interfaces based on component libraries. In Fig. 2, the characteristics of the interface with the model of the reactor containment ventilation system of a research reactor is presented. The model browser in the left window indicates

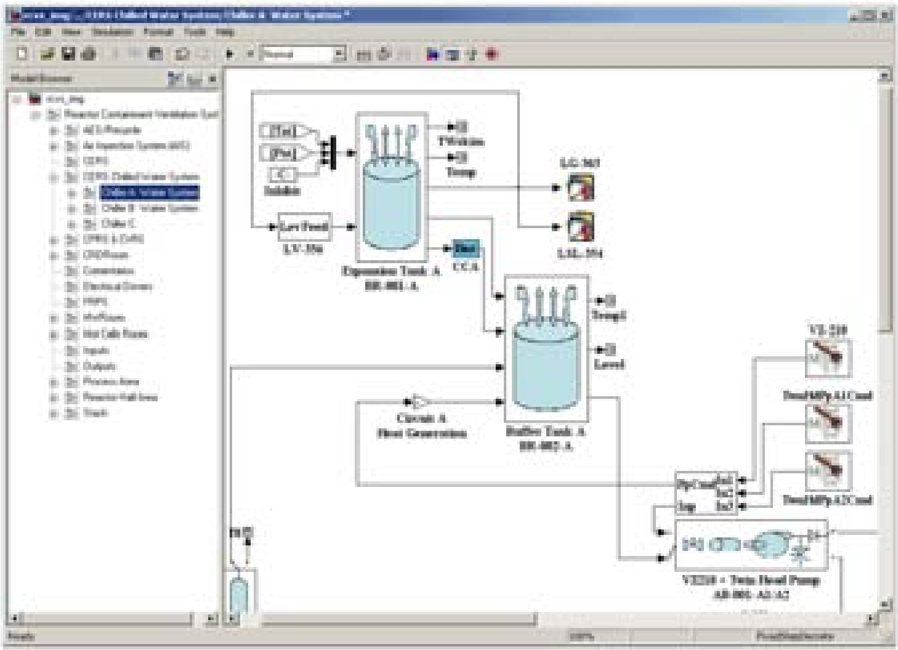


FIG. 2. Process model development environment with some library components.

that the graphical model shown is a subsystem of the chilled water system. This subsystem contains several library components that were dragged and dropped from the library into the window and then graphically connected. Tanks, pipes, isolation valves, actuators and sensors are shown in the portion of the subsystem model that is shown in Fig. 2. With this CAD-like user interface, flowsheet P&I-like diagrams can be defined for each plant system using a set of library components. The flowsheets can be grouped to implement the division of a large process into several flowsheet diagrams. The model browser allows navigating through the process.

The library components' parameters are input through menus (see Fig. 3). These component parameters are data structures that are linked to plant design input data. These parameters data structures provide the library component information about:

(a) The initial state of the component, such as nodes' temperatures, pressures, mass flow, aperture of a valve, state of a pump/fan/switch, level of a tank, precursor concentration, neutron power.

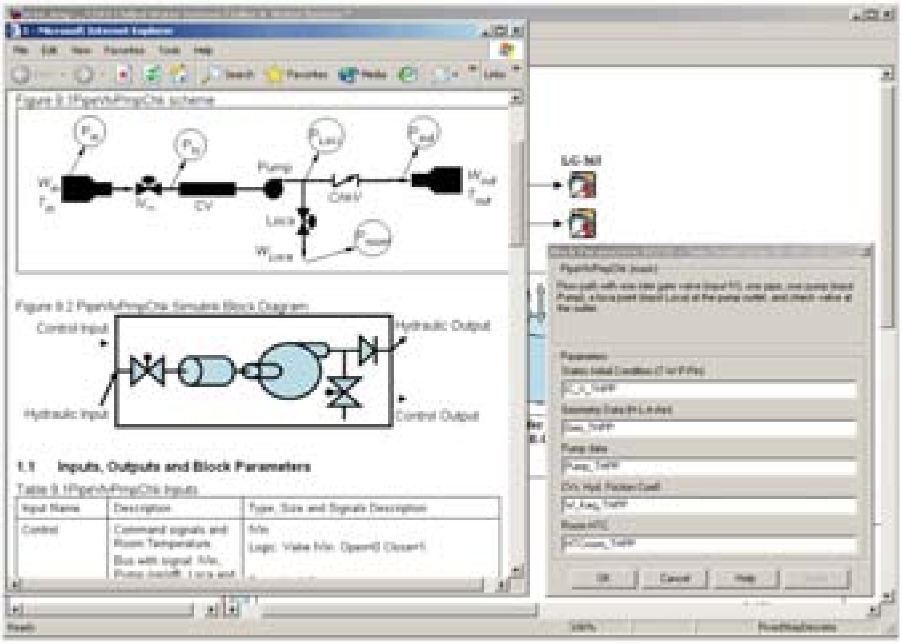


FIG. 3. Parameters of a library component with its associated on-line HTML-like help.

(b) Geometrical data, such as lengths, flow areas, levels, hydraulic diameters, heat exchange areas.

(c) Physical parameters, such as pump/fan/valve characteristic curves, hydraulic friction coefficient curve, heat transfer coefficients, kinetic parameters.

These parameters are defined in specific initialization ASCII files with high level Matlab commands.

Plant design data is collected into these files, which initialize the geometrical data structures and perform the steady state calculation. Several approaches can be implemented to tune some of the physical parameters to match the component initial state with the plant operative state.

On-line HTML-like help is available in the component parameter menu, with a description of each component input, output and parameters. All the commands and external input variables for the components are received in the input bus. Several important internal variables relevant to the component are available in the component output bus and can be selected through a bus selector menu.

The library of hydraulic components include objects such as tanks, headers, collectors, pipes, valves, control valves, check valves, flap valves, pumps, heat exchangers, cooling coils and other specific plant components. These objects are grouped into flow paths where single-phase one-dimensional mass and energy balance equations are solved in each of the control volumes. The momentum conservation is solved in an integral form over the flow path. Pressure changes and viscous losses are considered negligible on energy balance.

The core library component includes five hydraulic nodes for the core average channel. For each node, an adjacent metal node in thermal contact representing the cladding and fuel is defined. Neutron power is modelled through point kinetics, with a six delayed groups neutrons model, and eleven gamma groups to dynamically represent the decay power. The fraction of the nuclear power generated within each metal node depends on a predefined axial power distribution profile. Reactivity feedback terms from the mean fluid density, from the mean fluid temperature and from the mean fuel temperature are obtained through tabulated feedback coefficients applied to the deviation of these variables from the respective nominal values. These mean values are obtained as weighted averages within the core. The weighting depends on the power distribution within the core. The onset of nucleated boiling temperature and the saturation temperature in the average channel are used to define an ‘out of scope’ state for the component.

The library of air components includes objects such as humid air control volumes, air ducts, isolation valves, control valves, check valves (anti-return dampers), fans, cooling coils, electrical heaters and concrete structures. These humid air control volumes are connected through ducts with valves, fans, heaters and cooling coils to represent the air flow network. Mass and energy balance both for the air and vapour phase is solved within the air control volumes, ideal gas behaviour is used to calculate the internal pressure of the control volumes. Latent heat sources are defined as mass transfer liquid surfaces, sensible heat sources are defined both in the form of heat transfer surfaces and as an input to the control volume. In this way, both heat transfer to the concrete structures and the lighting/electrical heat sources are represented. Internal links of these electrical heat sources to the state of the components can be defined. The momentum equation is solved in the air ducts to define the mass flow between the air control volumes.

The library elements that define the inputs and outputs of the PMM are the sensors and actuators. The actuators define the inputs to the model and its use can be configured through a menu (Fig. 4). They are used as signal generators during the development phase of the model (debugging mode

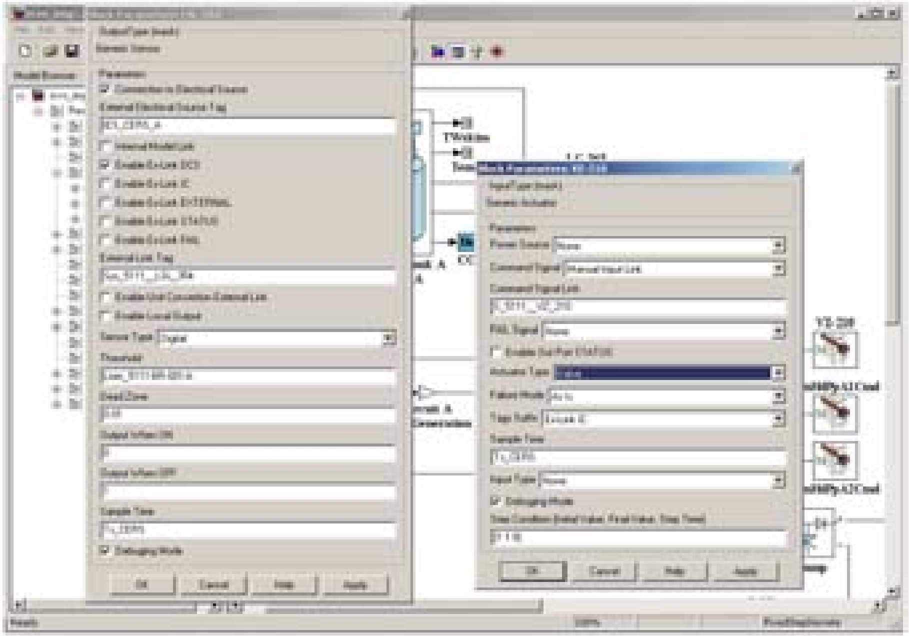


FIG. 4. Parameters for sensor and actuator library components.

checked in Fig. 4). These elements can represent a local command signal, such as the local start of a pump or manual opening of a valve (tags suffix IC, instructor console local action in Fig. 4), or an automatic action of the plant distributed control system setting the aperture of a control valve, or starting an air handling unit (tags suffix DCS). It can also be directed to an emulated hardware of the main control room (tags suffix External). These actuators can also be linked to a power source with a predefined fail mode, and receive fail signals from the instructor console.

The library components are dynamic libraries that are embedded into the MS simulation model. This allows lower code size and faster execution time in the model development phase.

The sensors define the outputs from the model and its use can be configured through a menu (Fig. 4). The signal can be output to the instructor console to represent a local sensor, such as a level glass or a manometer (Enable Ex-Link IC in Fig. 4), the instructor then can inspect or plot this variable in the IC. The signal can also be directed to the DCS (Enable Ex-Link DCS in Fig. 4), or to an external device, such as an emulated hardware of the main control room (Enable Ex-Link EXTERNAL in Fig. 4). A sensor status signal output and a sensor external fail signal input can also be added. The sensor type can be defined to be digital or analogue, in each case relevant sensor parameters are required to represent its behaviour: for digital sensors the threshold and dead zone to change state, for analogue sensors the noise variance, offset, characteristic time, low and high limits. These objects have no effect in the PMM during the model development phase (debugging mode checked in Fig. 4).

Each system of the PMM can be individually tested for steady state operative states, normal evolutions, and malfunctions within the MS simulation environment before its integration. Engineering design codes' outputs, such as RELAP [7] can be used for simulator model testing.

All system models are then integrated into the PMM, which can also be tested within the MS simulation environment. This model is precompiled to prepare all the input-output information to be used in other simulator software modules. Then it is compiled into C source code with the RTW and it is embedded into the MM.

4. INSTRUCTOR CONSOLE

Several initial conditions associated with different plant configurations can be loaded to start a simulation session. In Fig. 5, the instructor console

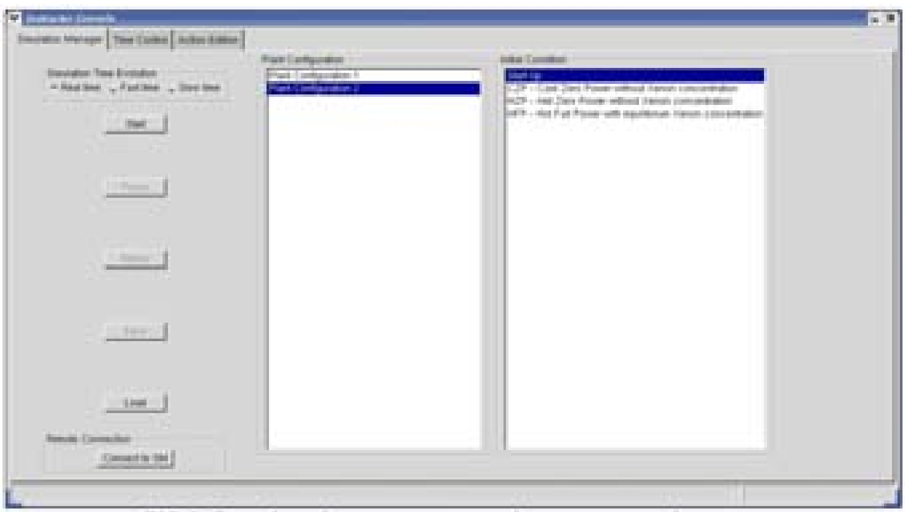


FIG. 5. General simulation management in the instructor console.

interface in the simulation manager tab is presented. In this window, some predefined initial conditions and plant configurations can be selected to be loaded for the session. The simulation can be started in real time or in slow or fast time; it can be ‘paused’, and it can be saved to be further reloaded. It can also be replayed from a previous snapshot.

In the time control tab (Fig. 6) the tasks’ time schedule and log window is presented. The tasks’ time schedule window shows all the defined events and malfunctions prepared for the simulation session. These tasks can be loaded from previous sessions and the tasks that have not been executed can be edited and modified, new tasks can also be inserted during a session. In the log window all the relevant events are presented, such as execution of tasks, inspection of variables, local actions, snapshots execution, out of scope messages. Tasks that are presented in the log window cannot be changed. At any time, the instructor can request a snapshot or it can be programmed to be executed on a fixed time-step basis.

The tasks can be created and modified in the action editor tab (Fig. 7). Each task has an execution time that is defined relative to the time when is added into the task’s schedule. A task is a set of simultaneous actions that are grouped in the task list (Fig. 7). These actions are selected from the custom tasks window where the instructor can select from a predefined set of actions over the plant components. The instructor can browse into the plant subsystems, component types within the subsystem, specific component and

FIG. 6. Tasks scheduling and log window in the instructor console.

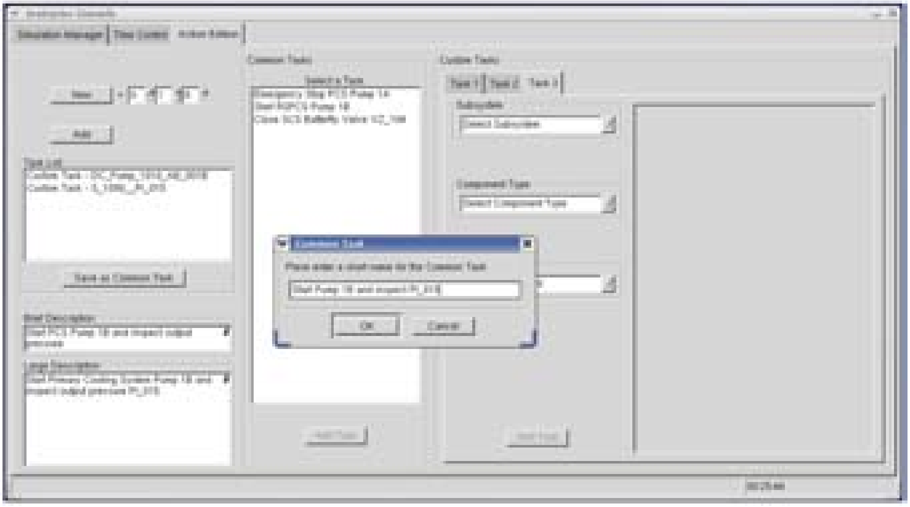


FIG. 7. Tasks loading and definition in the instructor console.

actions available for the component type. For example, the instructor can select the primary cooling system (plant system), within the system select the pumps' drivers (component type), then select a specific pump driver, and then select the commutation from remote to local (one of the actions available for this component type). The grouping of systems, component types and specific components is based on plant tags and are automatically linked to the sensor and actuator objects embedded into the MS model. All the sensors and actuators that were linked to the instructor console are presented in the proper subsystem and component type based on its tag.

Once the set of actions is completed to define the task, it can be added to the task list. If this task represents a routine plant operating procedure or an usual malfunction, it can be saved in order to be reused in another simulation session.

5. SIMULATOR HUMAN MACHINE INTERFACE

The SHMI is the software module that encapsulates the simulator SCADA. It performs the following tasks:

(a) Communication of all the inputs/outputs of the SACDA with the SM;

(b) Implementation of the links between the SCADA inputs and outputs with the corresponding PMM sensors and actuators;
(c) Timing control of the SCADA to implement the fast and slow simulation modes;
(d) Snapshot and initialization of the SCADA components, such as trend charts, control logics and mimics.

The simulator SCADA can be identical to the plant SCADA. This is the approach selected for the ANSTO RRR [8] reactor training simulator, where a Foxboro I/A Series [9] environment is used as the simulator SCADA. In this case, an engineering station (application workstation 70X, AW70X [9]), acts as an operator station and also contains a process that emulates the operation of the plant control processors (CP60 [9]), that implements the plant control strategy through regulatory logic, timing, sequential control, data acquisition, alarm detection and alarm notification. Two other stations provide additional trainee interface for supervisory control and monitoring on the I/A Series system, with access to displays, trending, alarming, historic data recording and system management functions.

Each input and output connection of the plant control processors to the field bus modules is redirected to the proper PMM sensor or actuator based on the tag of the connection.

This concept of providing the simulator with the same plant SCADA is also possible for other SCADA systems, provided that adequate emulation and interfacing capabilities are available [10, 11].

Control room hardware can also be emulated through graphical displays with touchscreen capabilities. This is used in the ANSTO RRR [8] reactor training simulator for the control rods drivers and reactor protection hardware (Fig. 8). This concept can be extended in the case of hardware-based control rooms.

6. ANSTO REACTOR TRAINING SIMULATOR

This simulation facility is being used to develop the ANSTO replacement research reactor (RRR) [8] reactor training simulator (RTS). The RTS is a full-scope and partial replica simulator. All the systems relevant to the plant normal evolutions and malfunctions listed in the DBA are included. Within these systems, both the variables connected to the plant SCADA and the local variables are included, leading to several thousands input-output variables in the PMM.

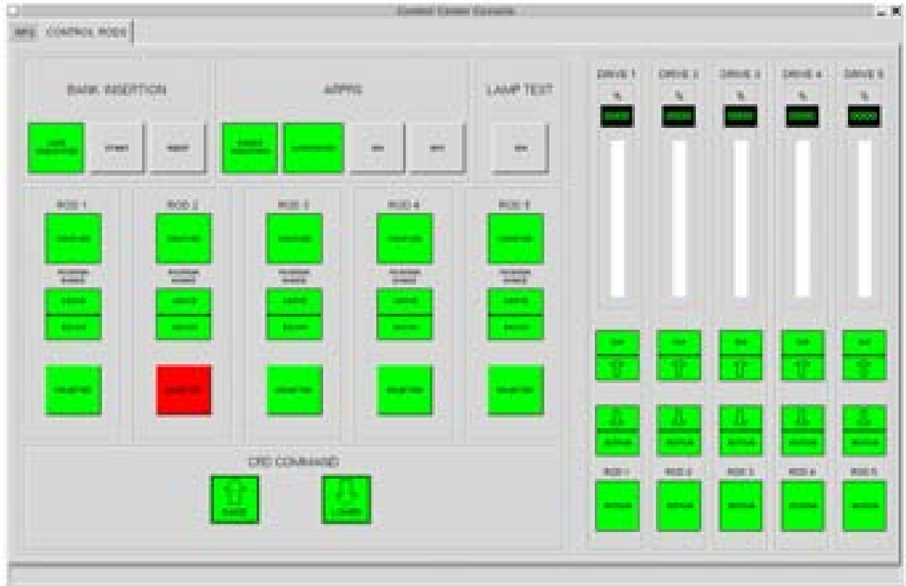


FIG. 8. Hardware emulation through graphical displays with touchscreen.

The simulator processes operation is being tested with versions of the PMM and the SHMI that include the primary cooling system, the reactor core, the reactor and service pools, the irradiation rigs, the reactor and service pool cooling system, the first shutdown system and the secondary cooling system.

System models relevant to the plant normal evolutions and malfunctions listed in the DBA were tested against RELAP outputs. The simulated models represented correctly the steady state, tendency and timing of all variables relevant to the plant normal evolutions and malfunctions included in the DBA list.

Other plant system models within the scope of the RTS are the reactor protection system, the electrical system, the reflector cooling and purification system, the second shutdown system, the nucleonic instrumentation system, the hot water layer system, the radiation monitoring system, the reactor containment and ventilation system, and the emergency make-up water system.

7. CONCLUSION

The use of Matlab, Simulink and Real-Time Workshop as the plant mathematical model development environment eases the model development process through the use of the specialized hydraulic-air-nuclear components libraries.

The libraries components model scope proved to be adequate to represent the ANSTO replacement research reactor normal evolutions and malfunctions included in the design basis accident list.

The plant mathematical model can be completely developed and tested using Matlab, Simulink and the specialized libraries; before it is integrated into the model manager using RTW.

Proprietary and more complex models can be embedded in Matlab, Simulink or in the model manager.

Sensors and actuators library objects proved to be an efficient resource to define the interface of the plant mathematical model with the instructor console and with the simulator human-machine interface.

Several options are possible for the simulator human-machine interface. As SCADA systems usually provide interfacing and emulation facilities for their products, it is possible to embed the same plant SCADA into the simulator human-machine interface. Control room hardware emulation through displays with touchscreen can also be used in the simulator human-machine interface.

REFERENCES

[1] OECD NUCLEAR ENERGY AGENCY, The Second CSNI Specialist Meeting on Simulators and Plant Analysers, Summary and Conclusions (Espoo, Finland, 1997), Rep. NEA/CNSI/R(97)36, OECD, Paris (1997).

[2] CORYS T.E.S.S, FRANCE, TRACTEBEL, BELGIUM, "ALICES: an Advanced Object-Oriented Software Workshop for Simulators", Simulators and Plant Analysers (Proc. 2nd CSNI Specialists Mtg Espoo, Finland, 1997), Rep. NEA/CSNI/R(1997)37, OECD, Paris (1998).

[3] TECHNICAL RESEARCH CENTRE OF FINLAND AND FINNTECH FINNISH TECHNOLOGY LTD., "APROS: a Multifunctional Modelling Environment", Simulators and Plant Analysers (Proc. 2nd CSNI Specialists Mtg Espoo, Finland, 1997), Rep. NEA/CSNI/R(1997)37, OECD, Paris (1998).

[4] KFKI ATOMIC ENERGY RESEARCH INSTITUTE, "GRASS: the Graphic Simulation System", Simulators and Plant Analysers (Proc. 2nd CSNI Specialists Mtg Espoo, Finland, 1997), Rep. NEA/CSNI/R(1997)37, OECD, Paris (1998).

[5] MATHWORKS INC., “MATLAB®, Simulink® and Real Time Workshop®”, 3 Apple Hill Drive, Natick, MA 01760-2098, http://www.mathworks.com
[6] DYNASIM AB, “DYMOLA®, Dynamic Modelling Laboratory”, http://www.dynasim.se
[7] INNOVATIVE SYSTEMS SOFTWARE, “RELAP/SCDAPSIM/MOD3.2”. L.L.C, 2000.
[8] AUSTRALIAN NUCLEAR SCIENCE AND TECHNOLOGY ORGANISATION, RRR Replacement Research Reactor, http://www.ansto.gov.au/ansto/RRR/
[9] THE FOXBORO COMPANY, “Foxboro I/A Series® system”, 33 Commercial Street, Foxboro, MA 02035, http://www.foxboro.com/iaseries/
[10] ABB AUTOMATION INC., “Symphony and Infi90 DCS simulation system”, 12808 West Airport Blvd, Houston, TX 77478, www.abb.com/automation
[11] SIEMENS AG, “SIMBApro PCI”, Siemensallee 84, Karlsruhe, 76187, Germany, http://www.solutions@khe1.siemens.de

BLANK

UTILIZATION PROGRAMME OF HANARO
Present and future

Y.J. KIM, B.J. JUN, K.B. PARK, I.H. KUK
Korea Atomic Energy Research Institute,
Yuseong, Daejon, Republic of Korea
Email: cpark@kaeri.re.kr

Abstract

Since the commencement of the High-flux Advanced Neutron Application Reactor (HANARO) operation in 1995, a significant number of experimental facilities have been developed and installed. Owing to the stable operation of reactor and rapid proliferation in utilization, more experimental facilities are continuously added up to satisfy the research needs arising. As a nationwide neutron research facility, HANARO is now successfully utilized in various fields such as neutron beam research, fuel and material tests, radioisotope production, and neutron activation analysis. The paper provides an overview of the present status and future plans of HANARO utilization.

1. INTRODUCTION

The High-flux Advanced Neutron Application Reactor (HANARO) is a multi-purpose research reactor with a design thermal power of 30 MW, constructed to meet national research demands for the 21st century. The HANARO has been operated since 1995 and is now utilized mainly in neutron beam research, fuel and material tests, radioisotopes production, and neutron activation analysis. A significant number of experimental facilities have been developed and installed since the beginning of reactor operation, and a continued effort of installing more facilities is still in progress. As new experimental facilities are added up, there was a rapid proliferation in utilization in terms of number of users as well as fields of applications. HANARO is establishing its status as a nationwide neutron research facility.

2. OPERATION PERFORMANCE OF HANARO

The operation of HANARO has been flexibly adjusted to meet the reactor's demands, and some reactor systems have been gradually improved. A two week operation and one week shutdown was the basic operation mode in the beginning. From 1998, the operation mode was changed to a weekly operation—at least three operation days every week for the stable supply of medical radioisotopes to meet the request of domestic hospitals. Annual operation time was about 160 days from 1998 to 2001, while the reactor power was gradually increased to meet the increasing demand as shown in Fig. 1. From mid-2002, the operation mode was changed again to a two week operation and one week shutdown so as to satisfy the rapidly increasing demand. Thereby, reactor availability reached about 210 days. In 2003, the operation mode was changed to 18 days operation and 10 days shutdown. It is almost equivalent to a three week operation and one week shutdown for the majority of reactor users, due to the reduction of operation during weekends. The operation time in 2003 was about 215 days and a similar record is expected in 2004, but the reactor availability to users will be increased because of the power increase to 30 MW. The cycle length will be increased further in the future as the demand increases.

3. EXPERIMENTAL FACILITIES

HANARO has a compact core and a spacious heavy water reflector region to accommodate many experimental holes with different neutron spectra. Three central flux traps located at the central region of the core provide a high fast neutron flux of over 2×10^{14} n/cm^{2}s. They can be utilized for

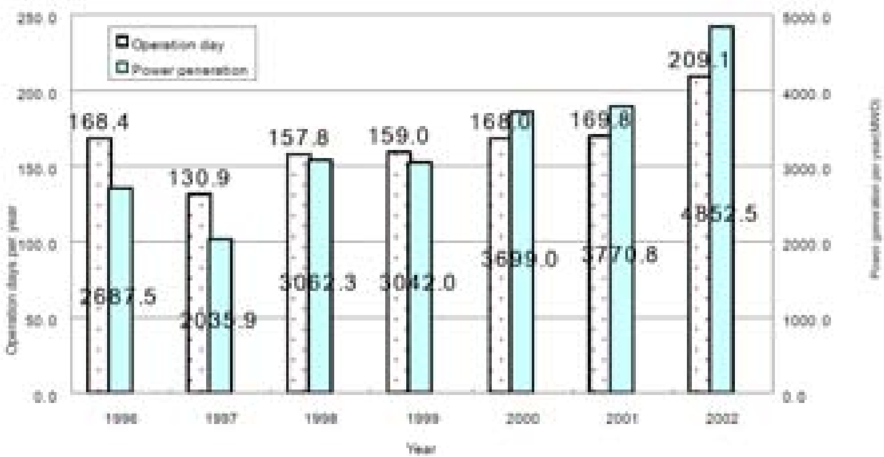


FIG. 1. Trend of the yearly power generation and operation time.

material and fuel tests. Four vertical holes in the outer core region, abundant in epithermal neutrons, are used for fuel or material tests and radioisotope production.

In the heavy water reflector region, 25 vertical holes abundant in high quality thermal neutrons are arranged for radioisotope production, neutron activation analysis, neutron transmutation doping and cold neutron source installation. Horizontally, there are seven beam ports of different types available for research in neutron scattering, neutron radiography, and medical applications. The arrangement of vertical holes and beam ports are shown in Fig. 2, together with corresponding experimental facilities.

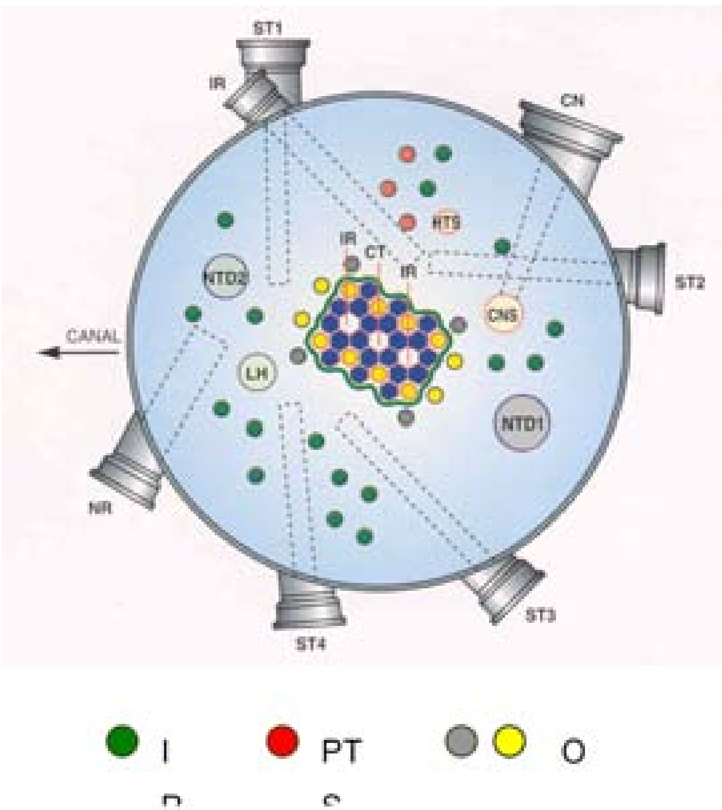


FIG. 2. Plan view of HANARO core and reflector for experimental facilities (see also the legend given in Table 1).

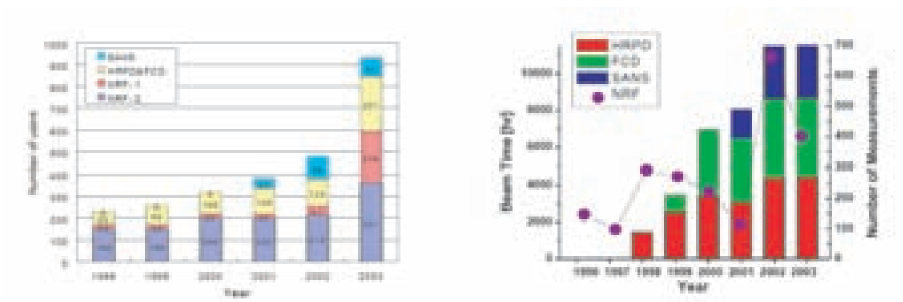
TABLE 1. LEGEND

Vertical experimental holes		Horizontal beam ports	
IR OR, CT	Capsule irradiation	ST1	Polarized neutron spectrometer
	Fuel test loop		Residual stress instrument
	RI Production	ST2	High resolution powder
LH	RI Production		diffractometer
HTS, IP	RI Production	ST3	Four circle diffractometer
PTS	Neutron activation analysis	ST4	Reflectometers
NTD	Neutron transmutation doping	IR	Triple axis spectrometer
CNS	Cold neutron source	CN	Boron neutron capture therapy
		NR	Small angle neutron spectrometer
			Neutron radiography

4. UTILIZATION OF HANARO

4.1. Neutron beam research

Since the commencement of HANARO operation in 1995, a series of neutron scattering instruments have been developed and installed. The neutron radiography facility (NRF) was installed in 1996 as the first experimental facility. The high resolution powder diffractometer (HRPD) became operational in 1998, followed by the four circle diffractometer (FCD) in 1999, the residual stress instrument (RSI) in 2000, and the small angle neutron spectrometer (SANS) in 2001, respectively. They are currently available to users and are under full utilization, as shown in Fig. 3.



Annual trend of users
(NRF-1: General users;
NRF-2: Student training)

Annual beam time allocation
(NRF: number of measurements)

FIG. 3. Yearly statistics of utilization in neutron beam research.

HRPD and SANS have been hot instruments these days, attracting a wide range of users from academia, institutes and industries. NRF is becoming a practical tool for industrial and scientific applications as neutron tomography and dynamic radiography capabilities are offered. FCD is a quite classical single crystal diffractometer whose beam time has been equally allocated for single crystal studies on the academic side and texture works on the industrial one. RSI is an instrument to measure residual stress on industrial materials. It was commissioned recently but is currently under upgrade with relocation from beam port ST2 to ST1. Along with the development of these instruments, various sample environments have been developed to meet diverse user demands and internal needs. Several low temperature environments based on closed-cycle refrigerator down to 10^{o} K are available for HRPD, FCD and high temperature furnace up to 1000^{o} K for HRPD and a special furnace fitted into the Euler cradle for the FCD can be used up to 800^{o} K. Several sample stages including heat blocks are possible for SANS.

Considerable effort over the past 10 years was put into the development of some key components such as monochromators, collimators and precision motion units with its motion controller specific to heavy load neutron instruments, etc. The development and application of a position sensitive detector (PSD) was a big success. In 2003, the development of neutron mirror techniques and devices was undertaken. After some success and some failure, techniques could be built on the fabrication of a Gd-coated foil collimator, a bent-perfect Si crystal monochromator, Cu monochromator and high power drive stepping motor controller. All of these combined enabled us to develop high performance neutron instruments in a cost effective way.

Instruments under development are the test station with a polarized neutron spectrometer (PNS) option and the reflectometer with vertical sample geometry (REF-V). PNS is the only one instrument at present in HANARO which can change its take-off angle, i.e. wavelength, and using this characteristic we will use this instrument for various testing and development purposes. The PNS and REF-V will be open to users in 2004 and 2005, respectively. During the third phase of the R&D programme starting from 2003, it is planned to develop additional instruments complying with user demand, the HIPD, the reflectometer with horizontal sample geometry (REF-H), and the triple axis spectrometer (TAS).

Most importantly, the project of constructing a cold neutron research facility (CNRF) at HANARO was launched in July 2003. The duration of the first phase of the CNRF project is five years starting from July 2003, and envisions installation of a cold neutron source, three neutron guides, and six cold neutron scattering instruments to satisfy the imminent needs for cold neutron beam.

4.2. Fuel and material irradiation tests

The nuclear fuel and material irradiation tests are one of the important missions of HANARO to support the national mid- and long-term nuclear R&D programme and basic research. The irradiation tests primarily require using the flux trap positions at an in-core region. Four types of capsule have been developed and offered to internal and external users; non-instrumented and instrumented capsules for material and fuel irradiation experiments, respectively. Figure 4 shows the yearly trend of the number of samples and irradiation time using capsules.

For material irradiation experiments, 12 capsules (2 non-instrumented and 10 instrumented capsules) have been designed, fabricated and successfully irradiated in CT, IR1 or IR2 positions since 1995. The irradiated materials were mainly reactor pressure vessel, reactor core, Zr-based fuel cladding and pressure tube materials. These irradiation tests have produced valuable data, especially for life extension of nuclear power plants. The capsules are also used to support irradiation requests by external researchers from universities for nuclear, nano-, semi-conductor and magnetic materials. For a more quantitative understanding of neutron irradiation, the precise control technology of irradiation temperature and neutron fluence irrespective of reactor operation is under development.

A non-instrumented fuel test capsule has been developed and used for new fuels such as a high burn-up PWR fuel, direct use of PWR fuel in CANDU (DUPIC) fuel, and high performance metallic fuel since 1999. In addition, new research reactor fuels developed by KAERI are being irradiated in order to support the RERTR programme. An instrumented capsule is being developed to measure fuel characteristics, such as fuel temperature, internal pressure of fuel rod, fuel elongation and neutron flux during the irradiation test of nuclear

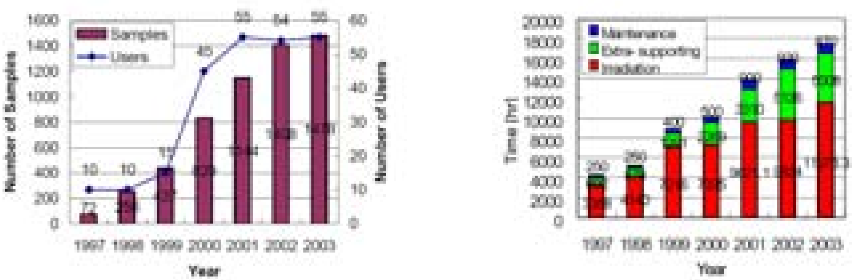


FIG. 4. Yearly statistics of irradiation test utilization.

fuel. In 2003, the centreline temperature of PWR UO_2 fuel pellets and neutron flux were continuously measured and monitored during the design verification test of an instrumented capsule. In 2004, internal pressure of fuel rod and fuel elongation, as well as fuel temperature and neutron flux, are to be measured during the irradiation test of UO_2 pellet. As the first special purpose capsule, a creep capsule was developed in order to examine creep behaviour of the zirconium alloy, SUS304, Cr-Ni steel in 2003. Other special purpose capsules for measuring creep, fatigue, crack propagation of material, for analysing gas sweep, and for measuring heat generation and power ramping of fuel will be developed in the near future.

The fuel test loop (FTL) development was started in December 2001 and will be completed by 2006. The FTL can simulate operating conditions (temperature, pressure and water chemistry) of commercial PWRs and CANDUs and, therefore, will be used for integral performance test of nuclear fuel. The loop consists of in-pile test section (IPS) and out-pile system (OPS). The number of fuel rods to be tested at a time in the IPS is limited to three at a maximum. The IPS will be installed at IR1 position.

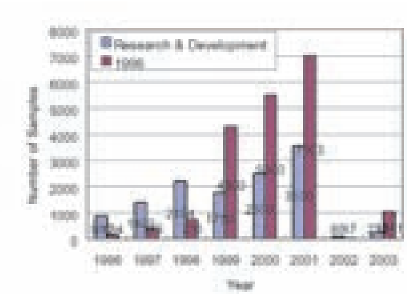
4.3. Neutron activation analysis

Neutron activation analysis (NAA) has rapidly extended its application in various scientific and technological areas. NAA has been applied to trace component analysis of nuclear, geological, biological, environmental and high purity materials, and various polymers for R&D.

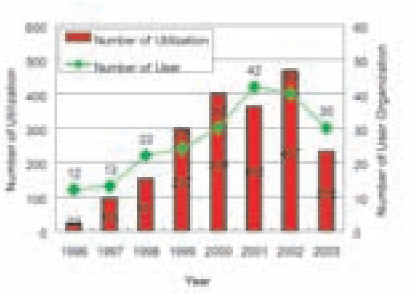
Automatic (PTS#1) and manual (PTS#2) pneumatic transfer systems were installed for NAA at three irradiation holes in the heavy water reflector region in 1995. Unfortunately, the Cd-lined hole for the epithermal NAA (PTS#1) became unavailable a few years ago due to depletion and contamination of Cd. A device for prompt gamma neutron activation analysis (PGAA) became available from 2002 for boron analysis in boron neutron capture therapy (BNCT). Yearly statistics of NAA utilization are presented in Fig. 5.

4.4. Radioisotope production

Labelled compounds including ^{99m}Tc cold kits and many radioisotopes, such as ^{131}I, ^{99m}Tc, ^{166}Ho, ^{192}Ir, ^{60}Co, are regularly produced in the radioisotope production facility (RIPF). Mass production of ^{192}Ir, ^{131}I solution and capsules commenced in 2001, and we currently supply most of their domestic demand. ^{192}Ir NDT assembly has been exported to Asian countries since September 2002. Several gauge sources used for process and quality control of industrial



Annual trend of samples.

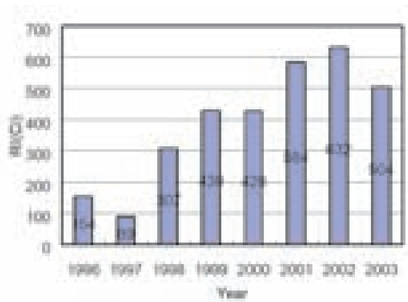


Annual trend of users.

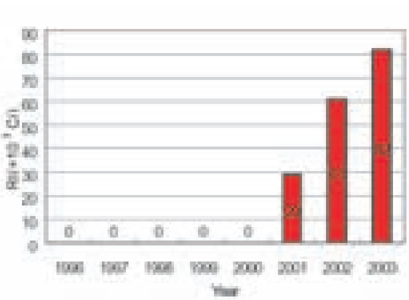
FIG. 5. Yearly statistics of utilization in neutron activation analysis.

sectors have also been supplied, based on user demand. For the quality assurance of final products, we obtained ISO 9002 certification in the field of radioisotope production. Technologies for production of fission moly using low enriched uranium and some other radioisotopes are under development. The Government of the Republic of Korea is considering construction of a new low power research reactor dedicated to radioisotope production.

Radioisotope production at HANARO has led to the development of new therapeutic modalities. ^{166}Ho chitosan complex developed by KAERI was approved by KFDA as a new therapeutic agent for the effective treatment of liver cancer. The same radio-pharmaceutical is under clinical trials for other applications as well, such as cystic brain tumour and rheumatoid arthritis, etc. KAERI has also developed other new therapeutic devices based on ^{166}Ho; ^{166}Ho-Patch for skin cancers and ^{166}Ho-Stent for esophageal, etc. Fig. 6. shows the yearly trend of commercial radioisotope production.



Medical application.



Industrial application.

FIG. 6. Yearly statistics of radioisotope production.

4.5. Boron neutron capture therapy

The boron neutron capture therapy (BNCT) facility delivers almost pure thermal neutrons that can be used for researches from cell culture study to clinical trials by focused irradiation without any interference from other utilization activities of the reactor. New promising boron compounds, whose concentration in the cancer cells is about ten times the existing ones, were developed. The facility is currently used for pre-clinical trials on animals. It is also used for dynamic neutron radiography, taking advantage of excellent beam quality and spacious experimental area.

4.6. Neutron transmutation doping

Two vertical holes are provided for neutron transmutation doping (NTD) at HANARO. A device to irradiate 5 inch ingots of 60 cm length was installed. Using this device, a commercial NTD service has been provided since the end of 2002. Excellent irradiation uniformity has been achieved with a very high accuracy to target dose. The axial variation of dose is less than 3% and one standard deviation to the target dose is about 1%. About 10 t was serviced in 2003. A device to irradiate 6 inch ingots at another hole is under development, whereas a provision is also considered for 8 inch ingots.

4.7. Other aspects

The nationwide utilization of the reactor is promoted, managed by the HANARO Steering Committee, the HANARO Users Council and User Groups. The HANARO Steering Committee, composed of Government representatives, HANARO users, the regulatory body, KAERI, and a few relevant experts, decides important policies on HANARO management and utilization. The HANARO Users Council, composed of representatives of user groups, discusses issues raised from user groups and makes recommendations to the HANARO Steering Committee. Presently, six active user groups are in operation. Each group holds regular meetings to discuss issues and to exchange information among them. A HANARO workshop is held once every year.

The Republic of Korea is leading an IAEA/RCA technical cooperation project in research reactors. Sharing regional research reactor resources is the major issue of the project. The Government of the Republic of Korea fully supports the project and encourages bilateral and multilateral collaborations in using the HANARO.

5. CONCLUSIONS

Operation of HANARO has been flexibly adjusted to meet the reactor's demands since the commencement of reactor operation in 1995. In mid-2004, the reactor is expected to achieve 30 MW operation with a cycle comprising an 18 day operation and 10 day shutdown, resulting in about 215 days of operation time a year. During the past years, a significant number of experimental facilities have been developed and installed to make efficient use of 3 vertical holes in the in-core region, 4 vertical holes in the outer core region, 25 vertical holes in the heavy water reflector region, and 7 horizontal beam ports. Owing to the stable operation of the reactor and rapid proliferation in utilization, more experimental facilities are continuously added up to satisfy the arising research needs. Among them, the most important ongoing efforts are the development of the FTL and the CNRF project. As a nationwide neutron research facility, HANARO is now successfully utilized in various fields, such as neutron beam research, fuel and material tests, radioisotope production and neutron activation analysis.

ENHANCEMENT IN THE UTILIZATION OF IEA-R1 RESEARCH REACTOR FOR RADIOISOTOPE PRODUCTION AND RESEARCH

R.N. SAXENA, R. FULFARO
Instituto de Pesquisas Energéticas e Nucleares (IPEN/CNEN-SP),
São Paulo, Brazil
Email: rnsaxena@net.ipen.br

Abstract

The paper describes the current status of the Brazilian research reactor IEA-R1 with respect to utilization and the future programme to enhance its utilization in the radioisotopes production for medical applications in the country. Plans to optimize the use of the reactor in research and other applications of nuclear techniques are also described. The reactor power is being upgraded from 2 to 5 MW and the operational cycle will be increased gradually from 60 to 120 h continuous per week. The new operational regime of the reactor will permit the production of ^{99}Mo using ^{98}Mo(n,γ) ^{99}Mo reaction on the natural target and preparation of ^{99}Mo-^{99m}Tc generators using the gel process. It is hoped that it would be possible to produce ^{99m}Tc generators with activities ranging from 250mC$_i$ to 1000mC$_i$ using the IEA-R1 reactor and to reduce the importation cost of ^{99}Mo by 15-20%. The ^{99}Mo-^{99m}Tc generators represent about 90% of all medical radioisotopes used by hospitals and clinics in the country. The new operational regime will also permit the local production of ^{131}I. Savings in the importation cost in the case of ^{131}I would amount to more than 50%. As a consequence of increased neutron flux with reactor operating at 5 MW, as well as the possibility of prolonged and continuous irradiation periods, other applications such as NTD silicon production and the production of ^{192}Ir for industrial gammagraphy and brachytherapy will have improved performance. The basic and applied research at the reactor will benefit from this new operational regime due to the availability of more intense neutron beams and a prolonged reactor operation period.

1. INTRODUCTION

The nuclear reactor is a strong neutron source for both thermal and fast neutrons and can be efficiently used for the production of radioisotopes with numerous applications in medicine, agriculture and industry and for other irradiation services. It could also be used for academic and applied research in the areas of nuclear and neutron related sciences and engineering. The extent of the utilization of the reactor is basically determined by its power level, which in turn determines the neutron flux available, as well as on its operational schedule. Small power levels and short operational cycles (only a few hours of operation each day) are particularly inconvenient both for high quality research and also for the production of useful radioisotopes. A high power level in conjunction with the prolonged operational cycle (several weeks' continuous operation) on the other hand, result in high fuel consumption and require an expensive maintenance programme. Since small research reactors are generally operated and maintained through government funding, which is usually scarce, it is obvious that efforts must be made to optimize the utilization of such reactors and to determine the power level and operational regime on the basis of cost and benefit. Social benefits derived from reactor utilization are usually given priority, however, the cost of such benefits must be determined in a realistic manner.

Brazil has four operational research reactors, all under the responsibility of the Brazilian National Nuclear Energy Commission (CNEN). Some details about the characteristics of these reactors are presented in Table 1.

As can be seen, the IEA-R1 is the only research reactor in Brazil operating at a substantial power level suitable for its utilization in wide areas of research such as physics, chemistry, biology and engineering, as well as in the production of some useful radioisotopes for medical and other applications. We will briefly review the general features of this reactor, its past operational experience and current utilization. We shall also describe briefly our future plans to optimize its performance in the areas of research and development but specially its role as a radioisotope producer.

TABLE 1. BRAZILIAN RESEARCH REACTORS

	IEA-R1	IPR-R1	ARGONAUTA	IPEN/MB-01
Criticality	September 1957	November 1960	February 1965	November 1988
Operator	IPEN-CNEN/SP	CDTN-CNEN/MG	IEN-CNEN/RJ	IPEN-CNEN/SP
Location	São Paulo	Minas Gerais	Rio de Janeiro	São Paulo
Type	Swimming pool	Triga Mark I	Argonaut	Critical assembly
Power level	2–5 MW	100 kW	200 W	
Enrichment	20%	20%	20%	4.3%
Supplier	Babcok & Wilcox	General Atomics	USDOE	Brazil

2. IEA-R1 RESEARCH REACTOR

The IEA-R1 research reactor is a swimming pool-type reactor, moderated and cooled by light water, and uses graphite and beryllium as reflector. The reactor was constructed by Babcok and Wilcox and achieved its first criticality on 16 September 1957. Although designed to operate at 5 MW, the reactor operated at 2 MW only during the early 1960s and mid-1980s on an operational cycle of 8 h/d, 5 days per week. The reactor started operating at 2 MW on 40–64 h continuous per week cycle since then. Originally charged with 93% enriched U-Al standard fuel elements, the reactor presently uses fuel elements with 20% enriched uranium (U_3O_8-Al and U_3Si_2-Al) produced at IPEN. The reactor is operated and maintained by the Research Reactor Centre (CRPq) which is also responsible for irradiation and other services.

The IEA-R1 reactor is a multidisciplinary facility and has been used during all these years intensively for basic and applied research in nuclear and neutron related sciences and engineering. The reactor has also been used for training purpose, for producing some useful radioisotopes with applications in industry and nuclear medicine and miscellaneous irradiation services. Several departments of IPEN routinely use the reactor for their R&D work. Many scientists and students at the universities and other research institutions in the country also use it quite often for academic and technological research. However, the largest user of the reactor is the staff of the CRPq with an interest in basic and applied research in the areas of nuclear and neutron physics, nuclear metrology and nuclear analytical techniques.

3. BASIC AND APPLIED RESEARCH

As mentioned, the CRPq is responsible for the operation and maintenance of the IEA-R1 research reactor as well as for irradiation services. The CRPq has a threefold mission of promoting basic and applied research in nuclear and neutron related sciences, providing educational opportunities for students in these areas, including postgraduate and undergraduate teaching, and providing irradiation services and other applications resulting from the reactor utilization. Scientific programmes at the CRPq span several multidisciplinary, fundamental and applied research areas. Specific research programmes include nuclear structure study from beta and gamma decay of radioactive nuclei and nuclear reactions, nuclear and neutron metrology, neutron diffraction and a neutron multiple diffraction study for crystalline and magnetic structure determination, perturbed angular correlation (PAC) spectroscopy using radioactive nuclear probes to investigate nuclear hyperfine interactions in solids, and neutron activation analysis, both instrumental as well as involving radiochemical separations applied to the fields of health, agriculture, environment, geology and industry. Research in the area of applied physics includes neutron radiography and instrumentation. CRPq takes its role very seriously as one of the major research reactor facilities in the country providing educational opportunities to students in their programmes in the areas of nuclear and neutron related sciences. A large part of the research work has active participation of many graduate students, affiliated to the CRPq, working for their Master's in Science and doctoral degrees, as well as some undergraduate students initiating scientific activities. Research staff members at the CRPq offer a number of undergraduate and postgraduate courses in the areas of nuclear and neutron physics, nuclear solid state physics and radiochemistry, as well as special courses for the operators of research and power reactors.

Operational experience and utilization of the reactor in basic and applied research has been quite intense in the 46 years of its operation and has been described in some of the earlier publications [1, 2]. Here we mention only briefly some of the experimental facilities which have undergone major improvements in recent years.

A major programme to upgrade the neutron diffractometer installed at the IEA-R1 research reactor was undertaken a couple of years ago and is in its final stage of conclusion. The project, realized in collaboration with Missouri University Research Reactor (MURR), includes nine position sensitive detectors (PSD), a rotating oscillating collimator and an elastically bent silicon single crystal focusing monochromator.

The PSD stack will permit simultaneous measurement of neutron intensity in an angular interval of 30°. The monochromator will permit the choice of three different neutron wavelengths.

The combination of PSD and monochromator will lead to considerable improvement in the spectral resolution and the per-point data collection rate is expected to increase by a factor of 600 over the existing configuration. Further details about the new diffractometer at the IEA-R1 reactor are described in another contribution to this conference.

The R&D work on neutron radiography at the IEA-R1 reactor dates back to 1988. The radiographs are obtained either by using conventional X ray films with gadolinium converter, or polymers using a boron based converter screen.

More recently, a real time NR system has been added to the facility. The radiography equipment is installed at the 8-inch neutron beam tube, which has a neutron flux of about 2×10^6n/s·cm^2 at the irradiation position.

Real time neutron radiography consisting of continuous visualization of images uses a scintillator converter screen, a highly sensitivity digital videocamera along with a high performance computer for image processing.

Direct examination of dynamic events under neutrons can be observed. The resulting information helps in understanding the dynamic behaviour of complex systems.

4. NEUTRON IRRADIATION AND OTHER SERVICES

We firmly believe that no matter how important the academic research may be in its own right, it does not do much good if it does not make its way to the outside world somehow. The CRPq is making an enormous effort to enlarge the scope of services and applications resulting from reactor utilization so that more and more benefits of these applications could be offered to the society. Some of the products and services offered by our Centre find their way to the petroleum industry, aeronautical and space industry, medical clinics and hospitals, semiconductor industry, environmental agencies, universities and research institutions. We produce special radioisotopes, such as ^{41}Ar and ^{82}Br for the industrial process inspection, ^{192}Ir and ^{198}Au radiation sources used for brachytherapy, ^{153}Sm (EDTMP) for pain palliation in bone metastases, calibrated gamma sources of ^{133}Ba, ^{137}Cs, ^{57}Co, ^{60}Co, ^{241}Am and ^{152}Eu used in clinics and hospitals practicing nuclear medicine and research laboratories.

We offer regular services of non-destructive testing by real time neutron radiography, multielement trace analysis by NAA and miscellaneous neutron irradiation of samples for research applications. Regular irradiations are carried out to produce some primary radioisotopes for the radiopharmaceuticals centre at IPEN.

Neutron irradiation of silicon single crystals for doping with phosphorus (NTD) was developed at IPEN in the early 1990s. A simple device for irradiating silicon crystals with up to 4 inch diameter and 40 cm length located in the graphite reflector was soon installed in the reactor for a commercial irradiation service. The important feature of the irradiation procedure, to obtain excellent axial homogeneity in the distribution of neutron dose received by the crystal, is to irradiate two silicon crystals, each 20 cm long, placed one on top of another, simultaneously. After exactly 50% of the required neutron dose to attain the final target resistivity, the crystals are pulled out of the reactor and their positions are interchanged. The irradiation then continues until the remaining dose is complete. The precise detection of the 50% fluence is the key to success of this method and it is accomplished by measuring the neutron dose using self-powered detectors placed close to the irradiation tube, as well as by cobalt monitors which are irradiated with each silicon ingot. Details about the design of the irradiation rig and its performance can be found else where [3, 4]. A new device for irradiating up to 5 inch diameter crystals with 50 cm length was later installed in the same position, replacing the older rig. Silicon irradiation capacity at the reactor for 5 inch diameter crystals is about 3–4 t/a, considering target resistivities in the range of 50–60 Ωcm.

5. RADIOISOTOPES PRODUCTION

The beginning of radioisotope production at IPEN for medical applications dates back to the late 1950s when a small experimental production of ^{131}I was started. In 1961, with the reactor operating at 2 MW and for 8 h/d, 5 days per week, the production of ^{131}I was sufficient to meet the demand of existing hospitals and clinics practicing nuclear medicine in the country. In subsequent years, IPEN began to produce several other radioisotopes used in nuclear medicine, such as ^{24}Na, ^{32}P, ^{35}S, ^{42}K, ^{51}Cr, ^{198}Au, as well as the synthesis of several labelled compounds. For this purpose, a radiopharmaceuticals department was established and equipped with hot cells and radiochemical processing laboratories. National demand for these isotopes increased rapidly during the 1970s and it was no longer possible to meet the requirements of the hospitals and clinics with the existing production. IPEN, therefore, began to import primary radioisotopes, such as ^{131}I, ^{32}P and ^{51}Cr, which could not be produced in our reactor in sufficient quantities and high enough specific activities. These radioisotopes were processed at IPEN to produce pharmaceuticals and distributed to medical centres.

Until 1980, all the ^{99m}Tc generators used in the country were imported. Due to the rapidly increasing demand and the high cost of importation, the radiopharmaceuticals centre started to make its own ^{99m}Tc generator kits with automatic elution system using fission ^{99}Mo purchased from Canada. Today, ^{99}Mo-^{99m}Tc generators in the form of several kinds of radiopharmaceuticals represent more than 90% of all the radioisotopes distributed to hospitals and clinics in the country. Projections for the year 2003 indicate production of about 14500C_i of ^{99m}Tc generators with individual kit activity ranging from 250 mC_i to 2000 mC_i and distribute them to about 260 hospitals and clinics all over the country, benefiting more than two million patients. In addition to ^{99m}Tc generators, the radiopharmaceuticals centre also makes and distributes radioactive preparations for medical use based on ^{131}I (1050Ci/a), as well as smaller quantities of ^{51}Cr (2Ci/a), ^{32}P (3Ci/a) form the imported radioisotopes. The shorter lived radioisotope^{153}Sm (40C_i/a) is regularly produced in our own reactor and distributed in the form of ^{153}Sm (EDTMP) for treatment as pain palliation in bone metastases.

These radioisotopes and radiopharmaceuticals represent annual receipts from sales in the order of \$9–10 million for IPEN (estimates for the year 2003). The importation costs for the reactor produced primary radioisotopes, on the other hand, are of the order of \$5 million. A recent survey has shown that the demand for ^{99m}Tc generators and ^{131}I preparations is steadily increasing at the rate of 10% and 30% a year, respectively, and expected to continue increasing in the years to come. In order to meet a continuously increasing demand of these radioisotopes in the country, IPEN will have to face increasing importation costs easily reaching \$6–8 million in just a couple of years. This is quite cumbersome for an institution such as the IPEN, whose main funding comes from a federally approved budget. Importation cost factor aside, increasing reliance on practically one or two foreign suppliers of a vital radionuclide such as ^{99}Mo is certainly a strategic disadvantage for the country.

Considering these factors, as well as the possibility of producing, in our own reactor, radioisotopes such as ^{99}Mo, ^{192}Ir, ^{131}I, and ^{125}I, among others, an important decision was taken by the higher administration of IPEN a few years ago to upgrade the reactor power to 5 MW and to gradually increase the operational cycle to 120 h continuous per week. Initial plans to produce ^{99}Mo from fission were given up, principally due to lack of necessary funds, but also due to technical problems associated with complex radiochemical processing and the management and storage of the highly radioactive waste generated. It was decided to produce ^{99}Mo by ^{98}Mo(n,γ)^{99}Mo reaction using the natural Mo target and ^{99m}Tc generator using the gel process. To achieve the goal of producing high specific activity sources of ^{99}Mo, resulting in the ^{99m}Tc generators in the range of 250mC_i to 1000 mC_i activities, from our reactor it was found necessary to raise the reactor power to 5 MW and to adopt an operational cycle of 120 h continuous per week. Four main areas received particular attention: (1) optimization of reactor systems, structures and components; (2) optimization of reactor fuel elements production; (3) optimization of radiochemical processing facilities for radiopharmaceuticals production; and (4) an effective programme for spent fuel management. IPEN assigned definite priorities for these projects at the institutional level.

The proposed upgrading of the reactor power to 5 MW, as well as continuous 120 h operation per week was enthusiastically welcomed by most of the users of the reactor, as they saw in this the opportunity to increase the productivity of their services and products. The experimental facilities installed in the beam holes of the reactor for academic and applied research will benefit, largely due to an increase in the intensity of neutron beams but also due to increased data acquisition time available per week. Stimulated by new possibilities, many of the users set out to modernize the existing facilities to explore research areas, which were considered inaccessible due to lack of neutron source intensity. The quality of data in neutron diffraction or scattering experiments, for example, is determined mainly by the counting statistics and it should now be possible to obtain good quality data in an experiment of shorter duration, thus considerably reducing the systematic errors.

6. MODERNIZATION OF THE REACTOR

During the last several years, many changes in the reactor system and components have been made in an effort to extend the lifetime of the reactor and secure its safe operation. It is worth remembering that IEA-R1 is one of the oldest reactors of its kind in the world and has been operating for over 45 years with an excellent safety record. The utilization of our reactor has been quite intense during all these years, without presenting any major problems or the occurrence of incidents. Some of the important improvements made in the reactor systems during the last several years include the following:

— Modification of the reactor core from 6 × 5 to 5 × 5 using LEU fuel elements;
— Installation of a central beryllium irradiating element;
— Replacement of 13 graphite reflectors with beryllium;
— Installation of four isolation valves in the primary cooling system;
— Repairs in the cooling tower and pipelines;
— Installation of a new ventilation and air conditioning system;
— Improvement in the control instrumentation;
— Replacement of some of the old radiation monitoring system;
— Installation of an emergency spray core cooling system.

With these modifications introduced in the reactor, a new safety analysis report was prepared and submitted to the regulatory body. Authorization from the regulatory body was obtained in September 1997 for commissioning the reactor at 5 MW. The reactor operated at 5 MW for six months. However, since other projects including optimization of fuel element production, chemical processing facilities for ^{99m}Tc using gel process and spent fuel storage facility had not yet been implemented to the full extent, the reactor power was reduced back to 2 MW. The reactor has since been operating at 2 MW on a continuous 60 h per week cycle. From 1998–2003, substantial progress was made to implement the reactor fuel production programme. The fuel fabrication centre acquired the know-how and capacity to produce 15–18 fuel elements of the type U_3Si_2/a. At the same time, the chemical method for producing ^{99m}Tc generator by the gel process was developed and is in the final stages of its implementation. In 1999, all the spent fuel elements stored in the reactor pool since its first criticality (a total of 127 elements) were transferred to the United States of America under a bilateral agreement (USDOE–IPEN/CNEN). The reactor pool at present has storage space for more than 100 spent fuel elements. The available pool storage space should be sufficient for about 5–6 years of reactor operation at 5 MW: 120 h per week. The fuel consumption is estimated to be 16–18 elements per year in this regime of operation. A new project for spent fuel management and storage is being initiated at IPEN to investigate the possibility of an alternate dry storage space.

An ageing management and refurbishment programme for the IEA-R1 reactor components and systems is a continuous and ongoing activity of our research reactor centre. For example, the reactor pool water treatment and purification system is being replaced.

The present control elements of the reactor have begun to show signs of ageing and we plan to replace them with identical elements (fork type Ag-In-Cd), fabricated at IPEN, before the end of 2003.

7. CONCLUSIONS

The reactor modernization programme introduced several years ago at IPEN and its effective implementation during all these years with solid investments will guarantee safe and continuous operation of the reactor to achieve the goal of producing some of the important primary radioisotopes, such as ^{99}Mo and ^{131}I among others in our reactor. An important saving in the foreign exchange is expected to result from the substitution of importation. Estimated figures are \$600 000–800 000 for ^{99}Mo; \$250 000–300 000 for ^{131}I and \$50 000–80 000 for ^{192}Ir. Reactor operation under new conditions will also permit the production of ^{125}I seeds used for the treatment of prostate cancer, which are at present imported.

As a consequence of increased neutron flux available (maximum thermal neutron flux of 2×10^{14} n/cm^2·s at 5 MW power) and extended operation period, other applications and services, such as NTD silicon production, neutron radiography and neutron activation analysis, will have better performance. The improved operational regime of the reactor will stimulate renewed interests in other applications, which are at present at experimental stages, such as boron neutron capture therapy (BNCT) and the colouration of topaz. Neutron beam research will benefit due to the availability of more intense neutron beams. A viability study has recently been made for the possibility of installing a low angle neutron scattering (SANS) facility at our reactor. This activity was supported by the IAEA through a research contract. It should be emphasized that academic research and postgraduate teaching at our reactor centre are very important programmes in the effective utilization of the reactor. More than 70 postgraduate and undergraduate students affiliated with the centre are carrying out research towards Master's in Science and doctoral degree programmes, as well as formal training programmes through the utilization of experimental facilities at the reactor. Research scientists and professors from universities and other research institutions and their students have free access to the research facilities at the reactor centre. It is expected that the reactor will begin its operation at 5 MW for 60 h continuous per week in 2004, and we will gradually increase the operational period to 120 h per week in mid-2004.

REFERENCES

[1] FULFARO, R., et al., "Experience and research with the IEA-R1 Brazilian reactor", IAEA-SR-77, paper presented at IAEA Sem. on Research Reactor Operation and Use, Jülich, 1981.

[2] MAIORINO, J.R., "The Brazilian research reactor IEA-R1", Optimization of Research Reactor Utilization for Radioisotope Production (Mtg Jaeri, Tokai-Mura, 1995), Rep. IAEA Advisory Group, Vol. 1, IAEA, Vienna (1995).

[3] CARBONARI, A.W., et al., An irradiation rig for neutron transmutation doping of silicon in the IEA-R1 research reactor, Nucl. Instrum. Methods Phys. Res. **B 83** (1993) 157.

[4] CARBONARI, A.W., et al., "Neutron doping of silicon: Some experiments at the IEA-R1 research reactor", ICARID-94, 1994, paper presented at Int. Conf. on Applications of Radioisotopes and Radiation in Industrial Development, Bombay, 1994.

NEW IPEN–CNEN/SP NEUTRON POWDER DIFFRACTOMETER

C.B.R. PARENTE, V.L. MAZZOCCHI
Comissão Nacional de Energia Nuclear,
Instituto de Pesquisas Energéticas e Nucleares (IPEN–CNEN/SP),
São Paulo
Email: cparente@curiango.ipen.br

Y.P. MASCARENHAS
Universidade de São Paulo (Campus de São Carlos),
Instituto de Física,
São Carlos

Brazil

Abstract

A new IPEN–CNEN/SP neutron powder diffractometer is under construction at the 2 MW thermal IEA-R1 research reactor. It is an upgrading of the old IPEN-CNEN/SP multipurpose neutron diffractometer. The old diffractometer was a single detector instrument with a boron trifluoride (BF_3) detector and a flat copper mosaic single crystal monochromator. The main modifications introduced in the old instrument are installation of a position sensitive detector (PSD) and a bent perfect single crystal monochromator. A rotating oscillating collimator, placed at the entrance to the detector shielding, eliminates parasitic scattering from furnace or cryo-refrigerator heat shields in the vicinity of the sample, while only reducing the scattered intensity by about 10%. The collimator also makes the PSD less sensitive to ambient background leaking in through the shielding entrance. Placed at a distance of 1600 mm from the sample, the PSD spans an angular range of 20º of a diffraction pattern, resulting in quite a good resolution for the instrument. An extended powder diffraction pattern can be obtained by moving the detector and collecting the data in segments. In order to increase the neutron beam flux at the sample position, a focusing Si monochromator will be installed in the instrument. With a take-off angle of 84°, the monochromator can be positioned to produce four different wavelengths, namely 1.111, 1.399, 1.667 and 2.191 Å. A beam shutter will protect the operator during sample manipulation or installation of any device in the monochromatic beam. In comparison with the former instrument, the new diffractometer will have better resolution and will be about 600 times faster in data acquisition.

1. INTRODUCTION

A new IPEN–CNEN/SP neutron powder diffractometer is under construction at the 2 MW thermal IEA-R1 research reactor. It is an upgrading of the old IPEN–CNEN/SP multipurpose neutron diffractometer. The old diffractometer was a single detector instrument with a boron trifluoride (BF_3) detector and a flat copper mosaic single crystal monochromator. The main modifications introduced in the old instrument are listed in Table 1 presented below.

The new instrument has several parts. All parts belonging to the detector system, except for the detector shield, were imported from Instrumentation Associates Inc. (IA), in the United States of America. Other parts, such as the main neutron shield, detector shield, open collimators and beam shutter, were mostly constructed at the IPEN machine shop. In the discussion that follows, a brief description of the main parts of the new diffractometer is given.

TABLE 1. MAIN MODIFICATIONS INTRODUCED IN THE OLD INSTRUMENT

Modified parts	Old diffractometer	New diffractometer
Detector	A single boron trifluoride (BF_3) neutron detector	A position sensitive detector (PSD)
Monochromator	A Cu flat mosaic single crystal	A nine-blade unit, each blade a Si bent perfect single crystal
Take-off angle ($2\theta_M$)	36°	84°
Wavelength (λ)	1.137 Å	Four different λs available
In-pile collimator	A Soller collimator	An open collimator
Incident-beam collimator	A Soller collimator	A pyramidal open collimator
Scattered-beam collimator	A Soller collimator	A rotating-oscillating collimator
Detector shield	A cylindrical shield	A pyramidal shield
Main neutron shield	A semicylindrical shield	A parallelepipedal shield + the old semicylindrical shield
Beam shutter	A beam port door (a normal component of the reactor)	A shutter formed by two contrarotating drums with peripheral square channels

2. POSITION SENSITIVE DETECTOR

The position sensitive detector (PSD) consists of 11 linear detector elements, clamped together at each end to form a rigid plane. A linear detector element is a proportional counter manufactured by Reuter-Stokes Inc. The 25 mm outside diameter stainless steel cylindrical tube of an element has a wall thickness of 0.25 mm and an active detector length of 610 mm. The anode wire is nickel chrome with a diameter of 0.015 mm. The specific resistance of the wire is about 8000 Ω. The gas fill of the counters is 8 atm of ^{3}He for neutron detection, and 4 atm of Ar for stopping the reaction products (with 0.5% of CO_2 for quenching).

In operation, each end of a detector element is connected to a charge sensitive preamplifier and the detector anode is maintained at a bias of 2000 V. When a thermal neutron strikes the detector element, it can be captured by a ^{3}He atom in the fill gas. The capture creates ^{1}H and ^{3}H products with enough kinematical energy to ionize the gas. The well known nuclear reaction expresses such an interaction:

$$^3\mathrm{He} + \mathrm{n} \rightarrow {}^3\mathrm{H} + {}^1\mathrm{H} + 768\ \mathrm{kev}$$

Electrons created in the detector fill gas by the reaction by-products (^{1}H and ^{3}H) are drawn to the detector anode, injecting (with gas multiplication) a charge pulse on the detector anode. The positive ions are drawn to the detector wall. This signal propagates to each end of the detector element, is amplified by the preamplifiers and is passed to analogue-to-digital converters (dual-ADC modules). With this configuration, the linear detector element becomes a position sensitive detector. Figure 1 is a schematic drawing for the equivalent circuit representation of the detector-preamplifier assembly.

It is possible to understand many of the characteristics of these detector systems using a simple DC analysis. The charge injected at x divides into two currents (I_A and I_B) that flow to the left and right through the anode resistance and the input impedances of the two preamplifiers. Kirchoff's Law and charge conservation are sufficient to show that:

$$Q_B/(Q_A+Q_B) = r = (\rho x+Z_A)/(\rho L+Z_A+Z_B) \quad (1)$$

In equation (1), Q_A and Q_B are the charges that pass through the A- and B-side preamplifiers, ρ is the anode resistivity, L the anode (detector) length,

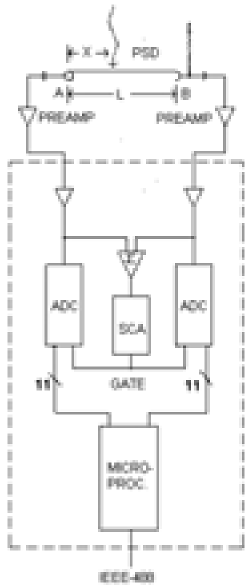


FIG. 1. Schematic diagram of the position sensitive detector electronics for a single detector element.

Z_A and Z_B are the preamplifier input impedances and x is the neutron capture coordinate measured from the A-end of the detector element. Equation (1) illustrates the linear relationship between the charge ratio $Q_B/(Q_A+Q_B)$ and the neutron capture position, x, and is the basis for the use of these devices and position sensitive detectors.

Figure 1 is a block diagram of the signal processing electronics for a single linear position sensitive proportional counter detector element, like that depicted in Fig. 2. Figure 3 is a schematic diagram showing the configuration of the detector system electronics. This system effectuates control and data

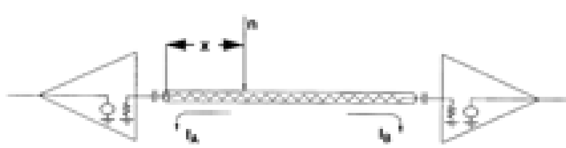


FIG. 2. Equivalent circuit representation of the detector-preamplifier assembly for a single linear position sensitive detector element.

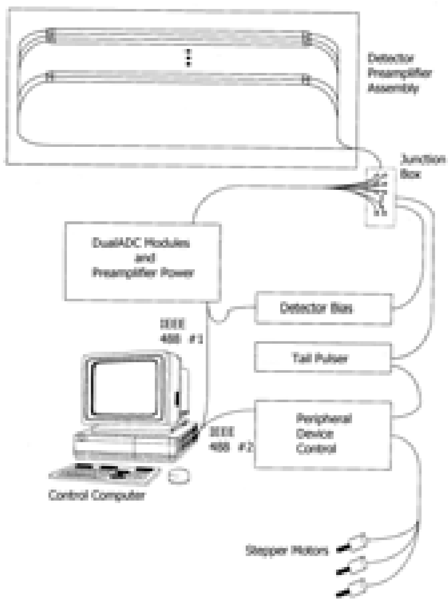


FIG. 3. Schematic diagram of the signal processing electronics for one linear position sensitive detector element. The dual-ADC module is the region enclosed in the dashed line.

acquisition in the PSD neutron powder diffractometer using adequate software for this purpose. The detector, enclosed in the upper box, includes the detector elements and preamplifiers. These are wired directly to a junction box that distributes the signals to the dual-ADC position decoding electronics. The junction box also connects the preamplifier power supply, electronic alignment tail pulser and high voltage bias supply to the detector assembly. The peripheral control system, in turn, connects to the motion controls for the monochromator, diffractometer, rotating-oscillating collimator and ancillary equipment (refrigerators, furnaces, etc.). Figure 4 is a photograph of the PSD array constructed for the new diffractometer.

3. FOCUSING SILICON MONOCHROMATOR

In order to increase the neutron beam flux at the sample position, a focusing silicon monochromator will be installed in the instrument. The unit is made of 9 vertically stacked silicon blades, approximately 5 mm thick, 14 mm high, 190 mm length, mechanically bent in the horizontal plane and quasi-bent

FIG. 4. Back view of the PSD array. Preamplifiers are identified by the large yellow capacitors at each end of the detectors. On the left of the detector is the junction box.

by segmentation in the vertical plane. All blades originate from the same plate, ensuring their correct relative orientation. At 84° take-off angle the following reflections/wavelengths (Å) can be attained: 533/1.111, 511/1.399, 331/1.667 and 311/2.191. Switching between 533, 511 and 311 reflections only requires rotating the crystal around the vertical $(01\bar{1})$ zone axis. Switching to 331 requires flipping the monochromator bottom up with a full circle stage.

Figure 5 shows the monochromator installed in a goniometer used for its orientation in the neutron beam. The goniometer has three movements: a rocking around a vertical axis, a tilting around a horizontal axis and a displacement along an axis normal to the monochromator surface. All movements are remote controlled via torque transmitters electrically coupled to torque receivers. In the 1960s, when the old diffractometer was built and stepper motors were not yet available, these devices were commonly employed in instrumentation. They were popularly known as ‘selsyns’. The goniometer was used for many years in the old diffractometer. It was slightly modified to support the focusing silicon monochromator.

4. DETECTOR SHIELD

To cut neutrons from the ambient background, a shield for the PSD was designed and constructed at IPEN. This shield has the form of a truncated pyramid coupled with a parallelepipedal box. The shield is made essentially of

FIG. 5. Front view of the focusing silicon monochromator. It is installed in the goniometer used for its orientation in the neutron beam.

double walled high density polyethylene panels. Neutron trapping is provided by a mixture of paraffin and boric acid filling the panels. Two arms fixed in a large 29 inch diameter rotary table support the shield. The table provides the instrument with the angular movement 2θ for the detector. Placed inside the parallelepipedal box, at a distance of 1600 mm from the sample, the PSD spans an angular range of 20 of a diffraction pattern. Depending on the desired extension for the experimental pattern, a complete pattern can be obtained by a 2θ angular scanning performed with only a few steps (maximum 2θ range about 120°). A smaller rotary table, placed underneath and concentric with the larger one, provides the $\omega(\theta)$ movement. Both tables are driven by a computer controlled geared mechanism. The assembly formed by rotary tables and mechanism is the same that was used in the old diffractometer.

Figure 6 shows the detector shield already finished. At the top of the shield, inserted into a narrow opening, an aluminium plate is used for the calibration of the PSD. It is a calibration mask. The mask is covered with a cadmium foil, for neutron absorption, and has several vertical equally spaced apertures to allow the neutrons to reach the detector only in certain positions. These apertures are necessary for the calibration process.

FIG. 6. The detector shield ready to be installed in the diffractometer.

5. ROTATING-OSCILLATING COLLIMATOR

A rotating-oscillating collimator (ROC), placed at the entrance to the detector shield, eliminates parasitic scattering from furnace or cryo-refrigerator heat shields in the vicinity of the sample, while only reducing the scattered intensity by about 10%. The collimator also makes the PSD less sensitive to ambient background leaking in through the shielding entrance. Both detector shield and ROC are counterbalanced by several lead bricks, placed on a basis fixed in an opposite position in the two supporting arms.

Figure 7 is a photograph of the ROC. The plates were removed in order to protect them before installation. In the upper part of the device, a stepper motor is coupled to a worm gear that makes the central region of the collimator assembly oscillate over a small angular range. Reversion is provoked by two microswitches placed at both ends of the movement. The microswitches are barely seen in the photograph.

6. OTHER COLLIMATORS

Two other collimators were constructed for the instrument: the in-pile and the incident-beam collimators. The in-pile collimator is an open collimator, i.e. it has no plates. It is inserted into the beam tube no. 6 of the reactor and is used to guide neutrons toward the monochromator, forming the polychromatic

FIG. 7. The rotating-oscillating collimator (ROC). Plates were removed for the safe handling of the device before its installation in the diffractometer.

beam. A central square tube guides the neutrons. The incident-beam collimator is also an open collimator. An inner duct, with a pyramidal form, has appropriate dimensions to allow focusing of the monochromatic beam on the sample. It is placed after the monochromator, at 84° (take-off angle) from the polychromatic beam.

Figure 8 is a photograph of the in-pile collimator separated into parts. The larger tube is its body. It is inserted into the beam tube. On the right of the body, a tube filled with barite concrete serves as a plug to cut the radiation off during maintenance. It is inserted into the body instead of the collimator. The tube with a ‘cage’ in the middle is the collimator itself. It is inserted into the body for normal operation of the instrument. The cage will accommodate a (future) sapphire filter to cut fast neutrons off from the polychromatic beam. Figure 9 is a photograph of the incident-beam collimator.

7. BEAM SHUTTER

A beam shutter was designed, constructed and installed in the main neutron shield. It protects the operator during sample manipulation or installation of any device in the sample position. The beam shutter is formed by two 500 mm diameter × 500 mm length contrarotating drums with 92 × 92 mm^2 peripheral square channels. The drums are filled with barite concrete for neutron

FIG. 8. Parts forming the in-pile collimator.

FIG. 9. The incident-beam collimator.

shielding and coupled to each other by a pair of identical gears. The drums rotate towards opposite directions supported by pairs of ball bearings. When the channels are aligned, they are in the right position to allow the neutron beam, coming from the reactor, to pass and reach the neutron monochromator. Owing to the gear coupling of the drums, moving one of them towards a certain direction, the other moves to the opposite direction. Consequently, the channels go to opposite positions, shutting the passage of neutrons. The shutter is driven by a 180 VDC/0.11 A electric motor, provided with a 1100:1 reduction gearbox. A metallic gear attached to the drive shaft of the motor is coupled to a rubber toothed strap fixed on the cylindrical surface of one of the drums. They form a sort of rack and pinion coupling. Movement and positioning of the shutter is controlled by an electronic control module, also designed and constructed at IPEN. The frontal panel of the electronic module has three coloured push buttons (red, black and green) that command movement and positioning of the beam shutter. The status of the system is indicated by three coloured lamps stacked on a pedestal (red, yellow and green for, respectively, beam on, intermediary position and beam out). The pedestal is placed on the top of the main neutron shield of the diffractometer for maximum visibility.

8. MAIN NEUTRON SHIELD

In order to avoid creation of a large ambient background in the diffractometer, a massive shield was designed and constructed at IPEN. This main neutron shield also accommodates the beam shutter, the focusing silicon monochromator and the incident-beam collimator. The main shield of the old diffractometer is now used as an additional shield, together with the new one. Both are supported by movable platforms to allow access to the beam port in the reactor wall.

Figure 10 is a schematic drawing of the beam shutter showing it in both *beam out* and *beam on* conditions.

Figure 11 is a photograph of the two neutron shields, the old and the new, in the position they will remain for the normal operation of the diffractometer. On the top of the shield are seen the indicator of the beam shutter status and a plug to be inserted into a vertical hole, where the monochromator is placed. The assembly of the two rotary tables used for the 2θ and $\omega(\theta)$ movements is also seen in the photo. Finally, Fig. 12 is a layout of the new diffractometer installed at the IEA-R1 research reactor. In this layout, the new main neutron shield is not pictured, but its position is that shown in Fig. 11.

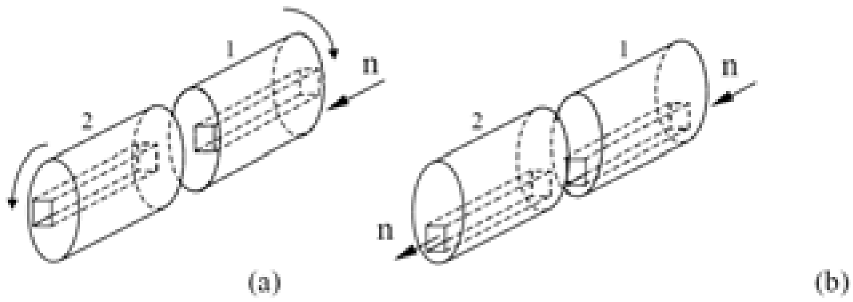


FIG. 10. Schematic drawing of the beam shutter in two different situations. On the left (a), the channels are in opposite positions corresponding to the beam out condition. On the right (b), the channels are aligned corresponding to the beam on condition.

FIG. 11. The old and the new main neutron shields.

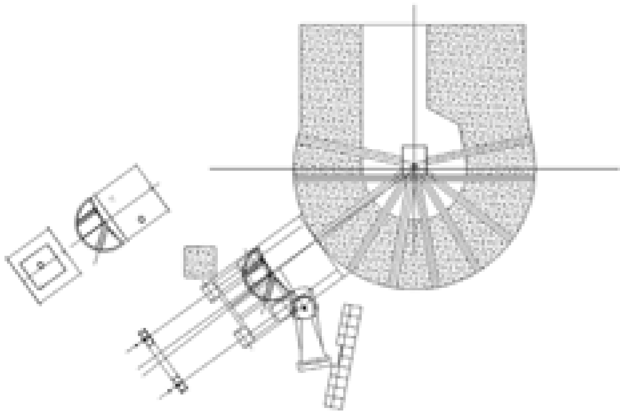


FIG. 12. Layout of the new diffractometer.

9. CONCLUSION

In comparison with the former instrument, the new diffractometer will have better resolution and will be about 600 times faster in data acquisition. The new diffractometer is presently in the final steps of construction. The end of the construction is foreseen for the next month. Installation and calibration of the PSD system still depend on the coming of an expert from IA. The utilization of this new instrument will be open for the Brazilian and Latin-American scientific and technological communities.

ACKNOWLEDGEMENT

The authors acknowledge the financial support given by Fundação de Amparo à Pesquisa do Estado de São Paulo (FAPESP), under project no. 95/05173-0, and Ministério da Ciência e Tecnologia, under project RXENZCS. They also acknowledge the inestimable cooperation of several people, particularly of Francisco E. Sprenger, Roberto M. Schoueri, Cláudio Domienikan, Fábio de Toledo, Edno A. Lenhatti and Alberto de Jesus Fernando.

BIBLIOGRAPHY

BERLINER, R., et al., A large area position sensitive neutron detector, Nucl. Instrum. Methods **185** (1981) 481.

BERLINER, R., et al., "Steinsvoll. Hardware and software system design for high resolution linear position sensitive proportional counter detector arrays", Data Acquisition Systems for Neutron Experimental Facilities (Proc. Int. Workshop, 1997), Frank Laboratory of Neutron Physics, Joint Institute for Nuclear Research, Moscow (1997).

POPOVICI, M., YELON, W.B., Focusing monochromators for neutron diffraction, J. Neutron Res. **3**(1) (1995) 1–25.

POPOVICI, M., YELON, W.B., A high performance focusing silicon monochromator, J. Neutron Res. **5** (1997) 227–239.

TOMPSON, C.W., et al., A position-sensitive detector for neutron powder diffraction, J. Appl. Crystallogr. **17** (1984) 385–394.

EIGHT YEARS OF NEUTRON ACTIVATION ANALYSIS IN GHANA USING A LOW POWER RESEARCH REACTOR

B.J.B. NYARKO, E.H.K. AKAHO, Y. SERFOR-ARMAH, E.K. OSAE
National Nuclear Research Institute,
Ghana Atomic Energy Commission,
Legon-Accra, Ghana
Email: bjbnyarko@yahoo.co.uk

Abstract

Ghana acquired a miniature neutron source reactor (MNSR) with assistance from the IAEA. The reactor, known in Ghana as Ghana Research Reactor-1 (GHARR-1), was manufactured by the China Institute of Atomic Energy (CIAE). The reactor was commissioned in March 1995 and has since been operational. The maximum neutron flux is 1×10^{12} $ns^{-1}{\cdot}cm^{-2}$ when it is being operated at full power of 30 kW. GHARR-1 is used mainly for neutron activation analysis. For the past eight years of its operation, the reactor has been used for the analysis of many samples of different origin. In the paper, some of the work carried out during this period is presented and the contribution made to the socioeconomic development of Ghana is discussed. The areas addressed include commercial activities, research and training.

1. INTRODUCTION

The miniature neutron source reactor (MNSR), known in Ghana as Ghana Research Reactor-1 (GHARR-1), developed and constructed by the China Institute of Energy (CIAE), is a small, simple, reliable and safe reactor [1]. MNSR adopts the pool-tank structure, and employs highly enriched uranium as fuel, light water as moderator and coolant, and metal beryllium as reflectors. The reactor is cooled by natural convention. The rated thermal power of the MNSR is 30 kW; the corresponding thermal neutron flux is 1.0×10^{12} $n{\cdot}cm^{-2}{\cdot}s^{-1}$. Since the small excess reactivity of the MNSR can not compensate the negative reactivity effect of the moderator and equilibrium xenon poisoning, it can not be continuously operated at the rated flux level for too long, but it can be operated continuously at $1 \times 10^{11}$$n{\cdot}cm^{-2}{\cdot}s^{-1}$ of the flux level for a longer time. The refuelling mode of this reactor is to totally change the old core with a new one, the lifetime being more than ten years.

The MNSR or GHARR-1 is simple in structure, easy to use and manage, and low in the cost of construction and operation. The MNSR possesses very good safety features, it has very small radiation effects on the site personnel and the public, and it can be built in the city areas with dense population. MNSR is used as a neutron activation analysis tool for research institutions, universities and hospitals; it can also be used to prepare radioisotopes with medium and short lived half-lives and it is a good tool for training and education. GHARR-1 was commissioned in March 1995 and has since been operational. The reactor GHARR-1 is used mainly for neutron activation analysis, and for research and training. In this paper, we present some of the major activities and work carried out during the past eight years.

2. NEUTRON ACTIVATION ANALYSIS

Neutron activation analysis (NAA) has been developed as a highly valuable analytical technique especially with the advent of nuclear reactors [2–4]. NAA is widely applied today to investigate biomedical, environmental, industrial, geological, and archaeological problems, etc. Due to its sensitivity, accuracy and precision, it is considered an ideal method for determination of a large number of trace elements in several samples. Trace elements play an important role in human life. Certain trace elements are essential, non-essential, or toxic to humans, animals and plants. Several trace elements, which are present in air, water, food and soil in concentrations greater than the maximum permissible level—mostly due to an increase in environmental pollution from anthropogenic activities—produce damage to animals and plant organisms and are generally dangerous to human health. The existence of a trace element in an industrial product may influence the quality of the product. The concentrations of trace element in an object of art have often been indicative of the origin of the object. At GHARR-1 Centre, the high advantages of NAA couple with the importance of trace elements to humans have given rise to a wide application of this technique. The various areas of interest and areas of active research and development using the Ghana MNSR are listed below:
Development and validation of analytical methods:

— k_0 – NAA standardization;
— Relative method;
— Flux mapping;

Radioanalytical chemistry QA/QC:

— Preconcentration;
— Microwave digestion;
— Speciation;

Geochemistry:

— Hydrochemistry;
— Geochemical mapping;
— Mineral exploitation;
— Soil fertility studies;

Environmental and health related studies:

— Analysis of medical plant and drugs;
— Food/IDD;
— Air pollution monitoring;
— Modelling: biomonitoring air pollution.

2.1. Development and validation of analytical methods

For the past eight years, we have focused on the development of improved instrumental neutron activation analysis (INAA) methods. The relative and k_0 – NAA methods have been established at the NAA laboratory for analysis of various types of materials for over two-thirds of the elements on the periodic table [5–8]. These methods have been validated with standard reference materials (SRMs). The SRMs used included: NIST Estuarine Sediment 1646, Orchard Leaves 1571, Coal Fly Ash 1633a, IAEA soil-7, Hey powder V-10, etc. The standardization methods also serve as a quality control procedure for the laboratory and are used to complement each other. We have used thermal and epithermal neutrons as target projectiles. Flexible boron (B_4C) was used as thermal neutron filters for the analysis of elements such as Au, As, Br, I, Rb, Th, U, W, etc., in different matrices [9]. The neutron spectrum characteristics for k_0 – NAA standardization methods have been determined in two inner and two outer irradiation sites of GHARR-1. The neutron spectrum characteristics determined were the ratio of thermal to epithermal neutron flux (f) and the epithermal neutron spectrum shape factor () for the Hogdahl convention and the modified spectra index $r()(T_n/T_0)$ and the neutron temperature T_n dependant on the Westcott –g factor $g(T_n)$ for the Westcott-formalism [7, 8, 10].

2.2. Radioanalytical chemistry and quality control

The laboratory for radioanalytical chemistry deals with preconcentration, digestion, speciation and quality management of the analytical procedure. The laboratory has carried out speciation of arsenic and Hg in water and sediments [11, 12]. The quality management systems in place at the moment are constant and regular calibration of the instruments and validation of methods. The laboratory has also participated in some proficiency tests with the IAEA and AFRA. Presently, the laboratory is in the process of preparing for certification and accreditation under IAEA project RAF/04/018.

2.3. Geochemistry

The geochemistry section of the NAA group has focused on the analysis of rocks, soil and water samples. Various rock types in the country have been analysed for their major, minor and trace elements for multipurpose geochemical mapping of Ghana [13–15]. Different types of soil in Ghana have been analysed for a variety of purposes, i.e. soil fertility and pollution studies. Groundwater analysis is also an ongoing project under this section [16–17].

2.4. Environmental and health related studies

In relation to human health, nutrition and environmental monitoring, the Centre has concentrated on the following areas: the analysis of medicinal plants and drugs. In this area, the Centre, in collaboration with the Centre for Scientific Research in Plant medicine (CSRPM) Mampong-Akuapim, have analysed different kinds of herbal medicine for their elemental content. The results served as a database for the CSRPM. In addition, three papers have been published [18–20]. We have also applied NAA for the study of chromium in diabetic and non-diabetic patients [21]. Zinc intake and absorption by humans was studied for Nigerian inhabitants.

In the area of food and nutrition, samples analysed included cassava and vegetables from mining and industrial areas [22], iodized salts [9], breast milk, baby food [23–24], cocoa products and many more.

The environmental studies have focused on pollution due to mining, industrial and marine pollution. Biological indicators (lichens and seaweeds), sediments and water have been used to study the degree of pollution in Ghana due to anthropogenic activities [25–28].

Over the years, we have analysed many samples for both research and commercial purposes. Figure 1 depicts the number of samples irradiated each year from 1996 to 2002.

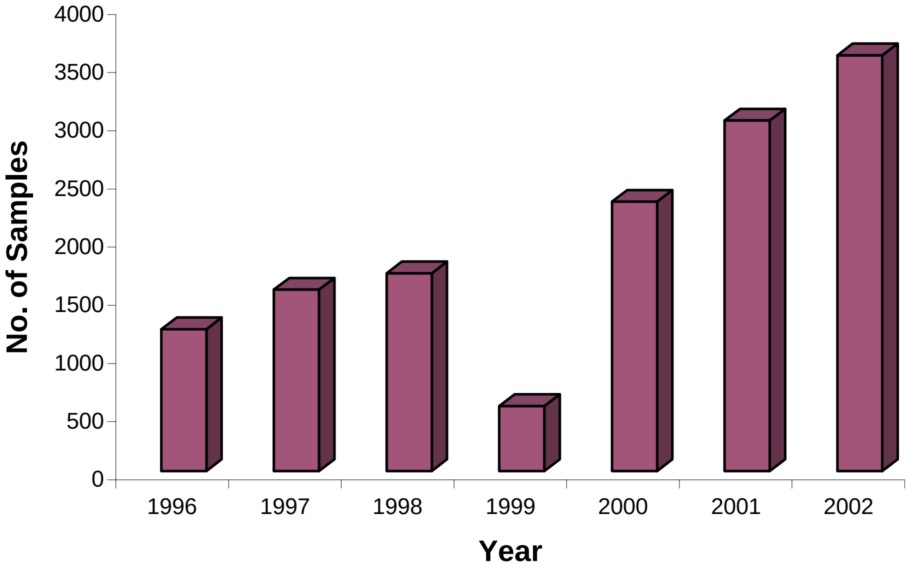


FIG. 1. Number of samples irradiated per year.

3. TRAINING

For the past eight years, universities and polytechnics from Ghana and Nigeria have used the reactor for teaching and learning nuclear science. In all 13, Bachelor of Science dissertations have been produced and one HND project work was carried out at the Centre from the University of Ghana, Kwame Nkrumah University of Science and Technology (KNUST), University of Cape Coast and Accra Polytechnic. There are eight Masters theses as of December 2002, from the University of Ghana and KNUST. One doctoral student from Nigeria did part of her work at the Centre using the reactor [29]. Students from junior and senior secondary schools from all over the country visit the Centre to acquaint themselves with the application of nuclear science.

4. COMMERCIAL WORK

The Centre has rendered services to many companies and industries. The analyses carried out included: crude oil, inlet oil, fuel oil, etc. for the Tema Oil Refinery. We have also worked for Ashanti Goldfields (Bibiani) Ltd, for the analysis of environmental samples and foodstuffs from the operation area.

Other companies are VALCO, Bogoso Gold Ltd, West African Gas Pipeline Project, Abosso Goldfields Ltd., and Panbros salt industry, just to mention a few. The commercial activities generated modest funds, which helped to sustain the running of the facility.

5. CONCLUSION

We would like to conclude by saying that the MNSR reactor, installed and commissioned in Ghana in 1995, has been used extensively to support the socioeconomic development of the country. The GHARR-1 Centre has received enormous support and financial assistance from the IAEA/AFRA. This has greatly enhanced our performance in the area of research and commercialization, and capacity building. It is our cherished hope that these activities will be enhanced in the years to come to meet the expectations of local and international standards.

ACKNOWLEDGEMENT

Our sincere thanks go to the IAEA/AFRA for the supply of almost all the equipment at the Centre and provision of standards. We also thank the IAEA for the human resource development put in place to train the staff, which has made the utilization of the Ghana MNSR a success. Acknowledged with thanks is the IAEA's sponsorship for this conference. We also thank the team of reactor operators and all the technicians at the Centre for their input.

REFERENCES

[1] CHINA INSTITUTE OF ATOMIC ENERGY, Miniature Neutron Source Reactor General Descriptions, (1992) 1–2.

[2] HOSTE, J., "Isotopic neutron sources for neutron activation analysis", User's Manual, IAEA-TECDOC-465, IAEA, Vienna (1988).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Practical Aspects of Operating a Neutron Activation Analysis Laboratory, IAEA-TECDOC-564, IAEA, Vienna (1990).

[4] DE SOETE, D., et al., Neutron Activation Analysis, Wiley Interscience, Hoboken, NJ (1978).

[5] OSAE, E.K., et al., Standardization of GHARER-1 gamma spectroscopy system for instrumental neutron activation analysis, J. KNUST (1997).

[6] NYARKO, B.J.B., Calibration and validation of APTEC gamma ray spectroscopy system at GHARR-1 Centre for INAA, Master's thesis, 1999.
[7] NYARKO, B.J.B., et al., Application of NAA standardization methods using a low power research reactor, J. Radioanal. Nucl. Chem. **257** 2 (2003) 361–366.
[8] AKAHO, E.H.K, NYARKO, B.J.B., Characterization of neutron flux spectra in irradiation sites of MNSR using the Westcott-formalism for the k0 neutron activation analysis method, J. Appl. Radiat. Isot. **57** 2 (2002) 165–273.
[9] NYARKO, B.J.B., et al., Epiboron instrumental neutron activation analysis for the determination of iodine in various salt samples, J. Radioanal. Nucl. Chem. **251** 2 (2002) 281–284.
[10] AKAHO, E.H.K., et al., Determination of neutron spectrum parameters of GHARR-1 inner irradiation sites for k0 standardization NAA, J. Appl. Sci. Technol. (2001) under review.
[11] ADOTEY, D.K., Environmental impact of mercury pollution due to small-scale gold mining activities in the Central and Western regions of Ghana, Master's thesis, 2003.
[12] CARBOO, D., et al., "Development and validation of speciation analysis using nuclear techniques: Speciation of As(III)/As(V) in borehole and tap waters in the Obuasi area", 1st Research Co-ordination Mtg Rep. for the CRP, 333-F2-RC-841, IAEA, Vienna (2002).
[13] DAMPARE, S.B., "Geochemistry and provenance of lower Birim diamondiferous fields", paper presented at 11th Faculty of Science Colloquium, University of Ghana, Legon-Accra, 2001.
[14] OSAE, E.K., et al., "Geochemical analysis of selected Ghanaian rock types: A prelude to a multipurpose geochemical mapping project", paper presented at IAEA/AFRA 3rd Int. Conf. on Research Reactor Utilization for Socio-economic Development, Rabat, 2003.
[15] ASIEDU, D.K., Geochemistry of Palaeoproterozoic metasedimentary rocks from the Birim diamondiferous fields of Southern Ghana: implications for provenance and crustal evolution at the Archaean-Proterozoic boundary, Geochem. J. (2003) under review.
[16] SETIGA, M., Determination of U and Th in and around the Weija lake using INAA methods, Bachelor of Science dissertation, KNUST, 2002.
[17] TIGME, J., Instrumental neutron activation analysis of groundwater (borehole) samples, Bachelor of Science dissertation, University of Ghana, 2002.
[18] SERFOR-ARMAH, Y., et al., Multi-elemental analysis of some traditional plant medicines used in Ghana, J. Trace Microprobe Technol. **3** 3 (2002) 419–427.
[19] SERFOR-ARMAH, Y., et al., Activation analysis of some essential elements in five plant medicines used in Ghana, J. Radioanal. Nucl. Chem. **250** 1 (2001) 173–176.
[20] SERFOR-ARMAH, Y., et al., Application of NAA to plant medicine in Ghana, A Review, J. Radioanal. Nucl. Chem. **257** 1 (2003) 125–128.
[21] GYAMFI, A., Neutron activation analysis of Cr in type II diabetic blood and food samples, Accra Polytechnic, Legon-Accra, 2000.

[22] DANSO K.E., et al., Determination of some mineral composition of cassava (*Manihot esculenta Crantz*) using INAA, J. Radioanal. Nucl. Chem. **250** 1 (2001) 139–142.

[23] JONAH, S.A., Routine monitoring of infant foods by INAA with a low power reactor, J. Trace Microprobe Technol. **21** 2 (2003) 405–411.

[24] DANQUAH, D., Epiboron INAA of Bromine and Iodine in infant formula, Bachelor of Science dissertation, University of Ghana, Legon-Accra, 2002.

[25] BOADU, M., et al., Determination of arsenic in some water bodies, untreated ore and tailing samples at Konogo in the Ashanti region of Ghana and its surrounding towns and villages by INAA, J. Radioanal. Nucl. Chem. **249** 3 (2001) 443–446.

[26] SERFOR-ARMAH, Y., et al., Elemental analysis of green and brown seaweeds from the coastal belt of Ghana, J. Radioanal. Nucl. Chem. **242** 1 (1999) 193–197.

[27] SERFOR-ARMAH, Y., Rhodophta seaweed species as bioindicators for monitoring toxic element pollutants in the marine ecosystem of Ghana, J. Water Air Soil Pollut. **127** 1–4 (2001) 243–254.

[28] NYARKO, B.J.B., et al., "Biomonitoring of trace-element air pollution in a gold mining area in Ghana using the generalized k0-standardization NAA method", poster presentation, Isotopic and Nuclear Analytical Techniques for Health and Environment (Proc. Int. Conf. Vienna, 2003), Isotopic and Nuclear Analytical Techniques for Health and Environment, C&S Papers Series No. 22, IAEA, Vienna (2004) CD-ROM.

[29] BRIGGS-KAMARA, M.P., Multielement analysis of epidermal creams by X-ray fluorescence spectrometry and instrumental neutron activation analysis, doctoral thesis, River State University of Science and Technology, Port Harcourt, Nigeria, 2001.

EXTENSIVE UTILIZATION OF TRAINING REACTOR VR-1

K. MATEJKA, L. SKLENKA
Czech Technical University in Prague,
Faculty of Nuclear Sciences and Physical Engineering,
Department of Nuclear Reactors,
Prague, Czech Republic
Email: matej@troja.fjfi.cvut.cz

Abstract

The paper describes one of the main purposes of the VR-1 training reactor utilization, i.e. the extensive educational programme. The educational programme is intended for the training of university students (all technical universities in the Czech Republic) and selected nuclear power plant personnel. At present, students can go through more than 20 different experimental exercises. An attractive programme, including demonstration of reactor operation, is also prepared for high school students. In addition, research and development works and information programmes proceed at the VR-1 reactor.

1. INTRODUCTION

The Czech Republic uses nuclear energy for a wide range of purposes. It not only uses nuclear energy for the production of electricity (e.g. the operational Dukovany nuclear power plant, four reactors VVER-440 and the Temelin nuclear power plant, two reactors VVER-1000), the Czech Republic also takes part in uranium mining, nuclear facilities production, nuclear waste management, R&D (from nuclear and reactor physics to radiobiology) and health physics.

The VR-1 Vrabec (known as ‘Sparrow’) training reactor, operated at the Faculty of Nuclear Sciences and Physical Engineering, Czech Technical University in Prague, was started up on 3 December 1990. Particularly, it is designed and operated for the training of students from Czech universities, preparation of experts for the Czech nuclear programme, as well as for certain R&D work, and for information programmes in the sphere of non-military nuclear energy use (public relations).

TABLE 1. BASIC TECHNICAL PARAMETERS OF THE VR-1 REACTOR

Rated power	1 kW (thermal), 5 kW for short time period
Fuel	IRT-3M type, ^{235}U enrichment 36% (imported from the Russian Federation)
Reactor vessels (pools)	Made from 08CH18N10T stainless material: Diameter (2300)/height (4720)/wall thickness (15 mm)
Reactor shielding	Above core: water (3000 mm) Side: water (900 mm) + super heavy concrete (950 mm)
Temperature	20° C (according to the ambient temperature)
Core cooling	Natural convection
Pressure	Atmospheric
Control system	5–7 control rods UR-70 with Cd absorbator as follows: 3 safety shutdown (scram) rods, 2 control rods and 0–2 experimental rods (according to the core configuration)
Operating power measurement	4 wide-range non-compensated fission chambers of RJ 1300 type (imported from Poland)
Independent power protection	4 pulse corona boron counters SNM-12 (imported from the former USSR)
Control devices	Microprocessor control system that performs especially: neutron flux density measurement, evaluation of the power and the rate of its change, information processing, connection with operators, controlling the control rods and the neutron source, giving the safety and safety signals.
Neutron flux	$2–3.10^{13} m^{-2}.s^{-1}$ according to the core configuration
Neutron source	Am-Be, 185 GBq, emission rate of $1.1 \, . \, 10^{7}$ s^{-1}

2. ACTUAL BASIC TECHNICAL PARAMETERS OF THE VR-1 REACTOR

The VR-1 training reactor is a pool-type light water reactor based on enriched uranium with maximum thermal power 1kW(th) and for a short time period up to 5kW(th). The moderator of neutrons is light demineralized water (H_2O) that is also used as a reflector, a biological shielding and a coolant. Heat is removed from the core with natural convection. The reactor core contains 14 to 18 fuel assemblies IRT-3M, depending on the geometric arrangement and kind of experiments to be performed in the reactor. The core is accommodated in a cylindrical stainless steel pool vessel, which is filled with water. UR-70 control rods serve the reactor control and safe shutdown. The parameters if the VR-1 reactor are indicated in Table 1.

3. VR-1 EXPERIMENTAL EQUIPMENT AND DEVICES

The main experimental equipment and devices on VR-1 reactor for training and research utilization are as follows:

— Radial and inserted (tangential) channel;
— Vertical channels 56 mm, ∅ 38 mm, ∅ 28 mm, ∅ 23 mm, and ∅ 12 mm;
— Manipulators;
— Measurement facilities of the delayed neutrons;
— Measurement facilities of the irradiation samples;
— Experimental facility (sub-critical assembly with graphite, fuel pins EK-10 type and fluoride salt) BLAZKA (study of selected neutron property of transmutation technology with fluoride salt) (see Fig. 1);
— Experimental probe BLANKA 02 for study of selected fluoride salt properties;
— Experimental facility BLANKA 417 for study of neutron and gamma spectra in special arrangement of active core (see Fig. 1);
— Measurement devices at the VR-1 laboratory (dosimetric equipment—new radiation monitoring network, neutron and gamma detectors, portable dosimetric devices, etc., multi-channel MCA and MCS analysers, special low current pico-amperemeters and other measurement devices);
— Different neutron chambers (^{3}He proportional counter 65NH25, ^{10}B proportional counter CPNB25, ^{235}U fission chamber CFUL01, ^{10}B compensated ionization chamber CC54 or ^{10}B corona detector SNM type);
— Semiconductor HPGe spectrometric system with high resolution and high efficiency is used for gamma measurement of irradiation samples.

4. SE OF THE VR-1 REACTOR

Students from technical universities and from natural sciences universities come to the reactor for training. The training provides students with experience in reactor and neutron physics, dosimetry, nuclear safety and nuclear installation operation.

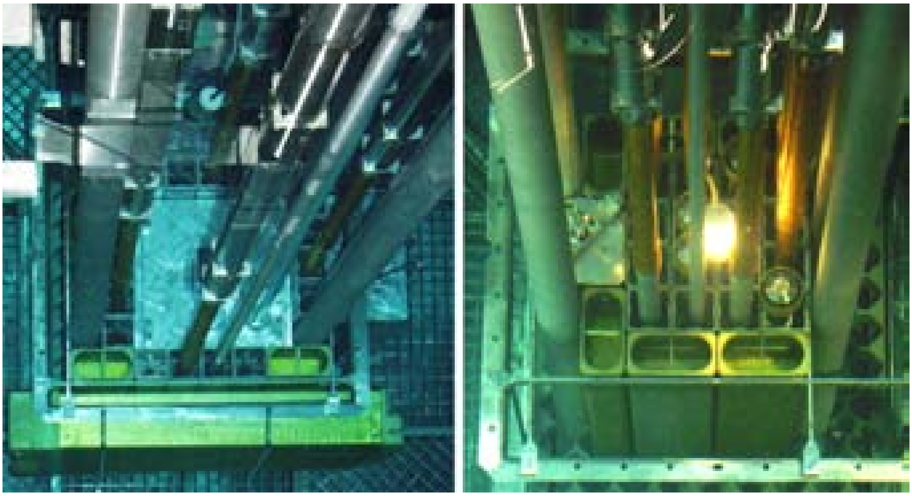


FIG. 1. The core of VR-1 reactor with sub-critical assembly BLAZKA (left) and BLANKA 417 (right).

Approximately 200 university students become familiar with the reactor (through lectures, experiments, experimental and diploma works, etc.) every year. About 12 different faculties from Czech universities (e.g. the Czech Technical University and Charles University in Prague, the Technical University and Masaryk University in Brno, Western Bohemia University in Plzeň, the Technical University in Ostrava) use the reactor. International cooperation with European universities in Austria, Germany, Hungary, the Netherlands, Slovakia, and the United Kingdom is frequent. A practical course on reactor physics in the framework of the European Nuclear Engineering Network (ENEN) has been newly introduced. Further training courses have been included to provide special training for selected specialists from Czech and Slovak nuclear power plants.

Scientific research respects reactor parameters and requirements of the so-called clean reactor core (free from a major effect of the fission products). Research on VR-1 is mainly aimed at the preparation and testing of new educational methodologies, the investigation of reactor lattice parameters, reactor dynamics study, research in the control equipment field, neutron detector calibration, etc.

Information services and promotional activities in the nuclear power field are important parts of the reactor operation. Many visitors, mainly high school students, come to the reactor. The reactor staff prepares an attractive programme including reactor operation. Every year, more than 1500 high school students come to visit the reactor, as do many foreigner visitors.

5. EXAMPLE OF THE EDUCATIONAL PROGRAMME

Currently, students can go through more than 20 experimental exercises. Many study materials (including textbooks [1], experiment procedures, diagrams and protocols) have been prepared. The following is a list of some of the experiments performed on the VR-1 reactor:

- Startup and operation of the VR-1 reactor;
- Properties of neutron detectors for nuclear reactor control;
- Measurement of delayed neutrons;
- Measurements of reactivity by various methods;
- Calibration of control rods by various methods;
- The critical or basic experiment;
- Determination of the effect of various materials on the reactivity of the reactor;
- Measurement of thermal neutron flux density;
- Study of nuclear reactor dynamics:
- Studying the reactor response to the negative reactivity change;
- Studying the reactor response to the positive reactivity change;
- Studying the influence of the bubbly boiling to the reactivity of the VR-1 reactor;
- Simulation of the selected operating statuses of the power reactor of the VVER type;
- Study of the properties of sub-critical multiplying assembly.

The experimental content of training courses is optional according to the interest of users.

6. PERSPECTIVE OF VR-1 TRAINING REACTOR

The plan for the training reactor VR-1 for the next 10 years covers essential activities (less important activities describe the annual plan for each year) in five fields: education activities, research activities, public relations activities, international cooperation, and human resources, innovation and new equipment. All our activities are prepared at a high level of nuclear safety, radiation protection, physical protection and emergency preparedness, and a good standard of ‘safety culture’.

6.1. Education activities

Education activities include the following:

— Keeping the status quo with respect to the number of user universities and schools, number of students and number of offered experimental exercises;
— Improving existing experimental exercises and establishing new, according to requests, users from universities and nuclear engineering companies, for example: study of neutron noise and its application, study of thermal effects, study of digital control systems, study of transmutation technologies ADTT, study of neutron detectors.

6.2. Research activities

Research activities include the following:

— Seeking research activities which can use advantages of ‘clean’ core without temperature, pressure, burn-up feedback, etc.;
— Continuing the study of the digitally controlled nuclear research reactors;
— Continuing the development of control equipment the VR-1 reactor;
— Continuing wide cooperation with Czech and Slovak institutions;
— Studying transmutation technologies ADTT.

6.3. Public relations activities

Public relations activities include the following:

— Keeping the status quo in the field number of user schools and number of visitors (see Fig. 2);
— Developing new demonstration experiments for secondary/high school students.

6.4. International cooperation

International cooperation includes the following:

— Continuing close cooperation with universities in Austria, Germany, Hungary, Slovakia, etc.;

FIG. 2. Students at the training reactor VR-1.

- Participation in the Reduce Enrichment of Research and Test Reactors (RERTR) Program in cooperation with the Nuclear Research Institute in Řež (Czech Republic);
- Participation in European Nuclear Education Network (ENEN) and NEPTUNO;
- Participation in World Nuclear University (WNU);
- Participation in IAEA training courses;
- Additional courses for potential users.

7. CONCLUSION

The experiences with the VR-1 operation have been excellent for the last 13 years. There was no accident regarding nuclear safety or radiation protection during the whole period of the use. Operation of the reactor is widely included in many study branches and it significantly contributes to the education of our students and also to the wider public in terms of conditions and acceptability of the use of nuclear energy in the Czech Republic.

REFERENCES

[1] MATEJKA, K., et al., Training reactor VR-1, Czech Technical University, Prague (April 1998).

BIBLIOGRAPHY

KROPIK, M., MATEJKA, K., “Upgrade of the Vr-1 training reactor”, in Enlarged Halden Reactor Programme Group Meeting, Halden, OECD Halden Reactor Project, Vol. 1, OECD, Paris (2002), pp. 47–58.

MATEJKA, K., SKLENKA, L., LISTÍK, E., “Conversion (utilizing LEU instead of HEU) of research reactors in the Czech Republic”, Reduced Enrichment for Research and Test Reactors (Proc. 21st Int. Mtg Sao Paolo, 1998), IAEA, Vienna (1999).

MATEJKA, K., SKLENKA, L., KROPÍK, M., KOLROS, A., “Educational programme on the training reactor Vr-1, Lisbon 1999”, Research Reactor Utilization, Safety and Management (Symp. Proc. Lisbon, 1999), IAEA-CSP-4/C, IAEA, Vienna (2000), pp. 33–34.

MATEJKA, K., BOUDA, J., 10 Years of Training Reactor Vr-1 Successful Operation (Proc.Workshop Prague, 2001) Vol. B, CTU, Prague (2001), pp. 712–713.

MATEJKA, K., “Safety operation of training reactor VR-1”, Topical Issues in Nuclear Safety (Proc. Int. Conf. Vienna, 2001), IAEA, Vienna (2002), pp. 212–217.

MATEJKA, K., RATAJ, J., UHER, J., Experimental Study of Ads at the Faculty of Nuclear Sciences and Physical Engineering, CTU in Prague (Proc. Workshop Prague, 2002) Vol. B, CTU, Prague (2002), pp. 818–819.

POLACH, S., MATEJKA, K., “Education of students and experts and personnel preparation on training reactor VR-1”, Research Facilities for the Future of Nuclear Energy, Proc. ENS Mtg, Brussels (1996), pp.113–119.

TEACHING AND TRAINING AT RA-6 REACTOR AND THEIR CONTRIBUTION TO THE RESEARCH FACILITIES DEVELOPMENT

O. CALZETTA LARRIEU, H. BLAUMANN
Centro Atómico Bariloche,
Bariloche, Argentina
Email: calzetta@cab.cnea.gov.ar

Abstract

The RA-6 reactor at the Bariloche Atomic Centre was designed mainly as a teaching tool. During its almost 21 years, after the first criticality, it was used as a support for several graduate and post graduate careers at the Balseiro Institute, depending on the Argentine National Atomic Energy Commission and the Cuyo National University. Besides these tasks, a big work was done in research and development using the synergy produced by the close relationship between the students and researchers. Main characteristics of the reactor are described. An outline of use for teaching and training is given. Research activities resulting from appropriate personnel management and students' cooperation that allowed achieving different grades of development are described.

1. INTRODUCTION

The RA-6 reactor is owned and operated by the Argentine National Atomic Energy Commission. It is located at the Bariloche Atomic Centre, Bariloche, Argentina.

The reactor was designed and built entirely in Argentina and it has been in operation since October 1982. The main design goal was to be a school reactor to support the nuclear engineering career and to be a nuclear experiment facility for the Nuclear Engineering Department. Figure 1 shows the RA-6 reactor building.

2. REACTOR DESCRIPTION

The RA-6 is a pool-type reactor, cooled and moderated by light water with 500 kW of nominal power. Figure 2 shows a view of the reactor block.

FIG. 1. RA-6 reactor building.

The reactor core configuration is variable inside an 8 × 10 grid and consists of about 28–32 fuel assemblies with U 90% enriched uranium and a number of graphite reflectors.

There are two different fuel assemblies:

- — The standard one (horizontal cross-section 81 mm × 77 mm, height 750 mm) has 19 plates with about 8 g of U235 each.
- — The control assembly that is similar to the previous one but four plates (number 2, 3 and 17, 18) were removed and replaced by two channels in which a fork control device can move up and down.

The control rods are made in cadmium with stainless steel clads. Normally, the core configurations have five control assemblies, four in the central area and one in the periphery. Their drive mechanisms are installed on the top of the pool at the mechanisms bridge.

Each reflector element is inside an aluminium box with the same outside dimensions as a fuel element.

The open pool itself is a stainless steel tank of 2.4 m in diameter and 10 m in depth.

Mainly two kinds of irradiation devices may be used to irradiate different types of specimens or samples. For long periods, irradiation hermetic aluminium cans are normally used and positioned in special boxes inside the

FIG. 2. A-6 reactor block.

core. A pneumatic irradiation facility is generally installed inside the core. The rabbit plastic capsule can be easily inserted and removed during reactor operation and is used for short periods of irradiation of samples.

For 15 years, an internal and an external graphite thermal columns were positioned in one lateral core face, but in 1997 were replaced by the internal filter and the external port of the BNCT facility.

There are five irradiation beam tubes, two of them from the external side of the reactor block to the core crossing the concrete shield and pool water. The other three only go up to the external pool liner and special devices are needed to reach the core.

Other important features of the reactor are:

— The instrumentation and the safety and control logic system are over qualified, which means more complete than it would be necessary for this small plant, in order to be a powerful tool to introduce the students and trainees in more complex systems.
— The same situation described above was carried out for the ventilation system.
— A second control room that provides all the information about plant's parameters is used to perform some supervised experiments.

3. TEACHING AND TRAINING

The reactor has been strongly involved in the nuclear engineering career of the Balseiro Institute through graduate and postgraduate courses and Master's and PhD theses during the last 20 years.

At graduate level, the reactor is used for the following courses: reactor physics, radiation protection, activation analysis, radiation physics and dosimetry, nuclear instrumentation and control, nuclear measurements and plant maintenance.

In this frame are performed experiments, including:

— Critical loading determination;
— Critical control rod position determination;
— Neutron flux profile measurements;
— Thermal power determination;
— Xenon poisoning;
— Reactivity coefficients determination;
— Control rod calibrations;
— Nuclear detectors calibrations and measurements;
— N16 control system;
— Neutron radiography;
— Gamma prompt analysis.

At post graduate level, the reactor provides an important support for the Medical Physics Magister career, the Nuclear Energy Technological Applications Specialist career and the IAEA–CNEA Radiation Protection Course.

It has also provided training for the exploitation teams of the NUR (Algeria) and MPRR (Egypt) reactors commissioned in the last decade and for the personnel involved in the development and operation of its facilities. Operators of the Argentinian nuclear power plants Atucha I and Embalse received specific training in reactor physics and more than 10 worldwide professionals were trained, supported by IAEA fellowships.

The narrow relationship between development and teaching in the reactor's group can be highlighted by the fact that several PhD theses were done in the last years [1, 2, 3] and several more are carrying on.

4. FACILITIES DEVELOPMENT

The irradiation facilities initially included in the reactor design were improved, focusing on specific interests with an appropriate personnel management, and students' cooperation that allowed achieving different grades of development:

The neutron activation analysis laboratory (LAAN) has been widely used for standard applications. This laboratory has been working also in the study of heavy metals in aquatic systems, with emphasis in mercury, analysing by instrumental neutron activation analysis bed sediments, suspended load, lake sediment cores, and biota, mainly fish muscle and liver. These studies include sediment core dating using ^{210}Pb and ^{137}Cs determined by gamma ray spectrometry, bioindication experiments with lichens for atmospheric transport and mussels in water bodies, and determinations of mercury methylation potentials in lake sediments using ^{197}Hg tracer. This work includes also the activation of materials with fast neutrons, measuring nuclear parameters such as cross-sections, half-lives, gamma ray energies and yields and, finally, forensic studies.

The boron neutron capture therapy (BNCT) facility [4, 5] has been completed, with an hyperthermal beam. Figure 3 shows a schematic representation of the BNCT facility.

The BNCT facility has been used for 'in vivo' and 'in vitro' experiments in hamsters and misses which contributed to the development of new applications of BNCT:

— Given that the hamster cheek pouch is the most widely accepted model of oral cancer, in vivo experiments were performed in order to assess the response of hamster cheek pouch tumours, pre-cancerous tissue and

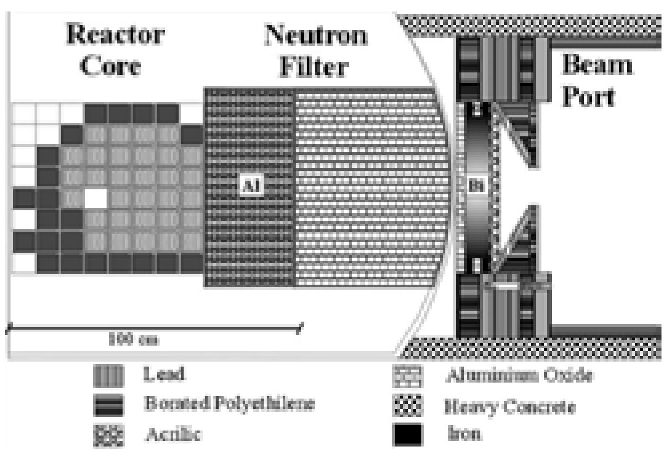


FIG. 3. Schematic representation of the BNCT facility showing the main components of the beam port.

normal tissue to BPA-mediated BNCT employing the thermalized epithermal beam. BNCT leads to complete remission by 15 days post-treatment in 78% of tumours and partial remission in a further 13% of tumours with virtually no damage to normal tissue [6].

— The nude mice model closely resembles the behaviour of undifferentiated thyroid carcinoma, which occurs in human beings. The animals were implanted in their lower back with human UTC ARO cells, and were irradiated with the hyperthermal neutron beam. Irradiations were performed in groups of eight mice placed on a mobile plate against the port's external shielding. The animals were distributed into four groups: control; NCT (no BPA); BNCT #1 (350 mg/kg b.w. BPA); and BNCT #2 (600 mg/kg b.w. BPA). For the BNCT groups, physical doses of 4.3 and 4.8 Gy in tumour were achieved, while the highest physical dose to normal tissue was 2.8 Gy. According to the experimental results [7], it may be concluded that, in all animals of group four (600 mg BPA/kg b.w, irradiation), regardless of the initial tumour volume, a 100% control of tumour growth was obtained at least during the first 30 days of followup. Also, when the initial volume of the tumour was 50 mm^3 or less, a complete regression (i.e. disappearance) of the tumours was observed in 50% of the mice, regardless of the BPA dose.

The BNCT facility also has been recently used for the treatment of human malignant melanomas. The neutron radiography facility is normally used for teaching, training and services. The prompt gamma neutron activation analysis (PGNAA) is being upgraded in the frame of the IAEA project "New Applications on PGNAA". It will be used not only for standard applications but also for in vivo applications. The neutron transmutation doping technique (NTDT) was successfully applied for doping small crystals used in solid state detectors.

Other examples of research and development, and teaching to be mentioned are:

— Iridium sources for brachytherapy;
— Failed fuel elements detection system;
— N16 power control system.

REFERENCES

[1] ARRIBÉRE, M.A., Determinación de secciones eficaces promediadas sobre un espectro de fisión para las reacciones umbral inducidas sobre elementos livianos. Estudio de su importancia analítica, doctoral thesis in nuclear engineering, Instituto Balseiro, Universidad Nacional de Cuyo and Comisión Nacional de Energía Atómica.

[2] SUÁREZ, P., Estimación semiempírica de secciones eficaces umbral promediadas sobre el espectro de fisión de ^{235}U, doctoral thesis in nuclear engineering, Instituto Balseiro, Universidad Nacional de Cuyo and Comisión Nacional de Energía Atómica.

[3] RIBEIRO GUEVARA, S., Determinación de secciones eficaces de reacciones con neutrones, discriminando los aportes a los estados metaestables y fundamental, doctoral thesis in nuclear engineering, Instituto Balseiro, Universidad Nacional de Cuyo and Comisión Nacional de Energía Atómica.

[4] CALZETTA LARRIEU, O., BLAUMANN, H., LONGHINO, J., "RA-6 reactor mixed beam design and performance for NCT trials," in Proceedings of the Tenth International Congress on Neutron Capture Therapy (SAUERWEIN, W., MOSS, R., WITTIG, A., Eds), Monduzzi Editore, International Proceedings Division, Bologna (2002) pp. 155–158.

[5] BLAUMANN, H.R., et al., Boron neutron capture therapy of skin melanomas at the RA-6 reactor: a procedural approach to beam set up and performance evaluation for upcoming clinical trials, accepted and to be published by Med. Phys.

[6] KREIMANN, E.L., et al., Boron neutron capture therapy for the treatment of oral cancer in the hamster cheek pouch model, Cancer Research (Advances in Brief) **61** (2001) 8638–8642.

[7] DAGROSA, M.A., et al., Experimental application of boron neutron capture therapy (BNCT) to undifferentiated thyroid carcinoma (UTC), accepted and to be published by Int. J. Radiat. Oncol. Biol. Physic.

BLANK

THE CONTRIBUTION TO INTERNATIONAL SAFEGUARDS AND SECURITY BY A SMALL UNIVERSITY RESEARCH REACTOR

H. BÖCK, M. VILLA
Atominstitut der Österreichischen Universitäten,
Vienna, Austria
Email: boeck@ati.ac.at

Abstract

Due to the proximity of the Atominstitut der Österreichischen Universitäten to the IAEA, many research projects have been carried out and completed successfully during the past years, especially in the field of nuclear safeguards and security, some of them have just started recently. In the paper, a survey on selected cooperation projects is given to show how a small university institute can contribute significantly to international safeguards and security research. Small university laboratories have the advantage of low cost but highly qualified personnel who can concentrate on investigation and test series which would largely exceed the possibilities of international organizations and are, therefore, a typical example of successful outsourcing for the benefit of both partners.

1. INTRODUCTION

In recent years, international safeguards and security of radioactive sources has become an increasing concern to countries, to the public and to international organizations. The main target of all international efforts is to prevent the spread and the illicit trafficking of special nuclear materials or radioactive sources. The development of highly specialized instruments being hand-held, portable or fixed installed is performed in cooperation with the IAEA, mainly in the United States of America and in Europe. However, even small research institutions, such as the Atominstitut der Österreichischen Universitäten in Vienna can contribute significantly to the improvement both of hardware and software, as well as to the practical test in different radiation fields under laboratory conditions. The Atominstitut belongs to the Vienna University of Technology and operates a 250 kW TRIGA type reactor mainly for education and training of students in the nuclear field. All relevant information can be found on the homepage (www.ati.ac.at). One advantage of the Atominstitut is the proximity to the IAEA—in fact, it is the closest nuclear facility to the IAEA headquarters. In past years, the Atominstitut cooperated very closely with several divisions in the IAEA to test software and hardware of newly developed nuclear instruments under variable environmental conditions.

2. DESCRIPTION OF SELECTED PROJECTS

2.1. Basic features of CdZnTe and CdTe detectors

In recent years, room temperature solid state detectors (CdZn and CdZnTe detectors) become increasingly important in nuclear safeguards, as they have a better gamma resolution than NaJ crystals and they do not need any external cooling such as HP Ge detectors. This brings definitely many advantages for safeguards inspectors working under field conditions. However, the behaviour of such detector types has to be tested under various environmental conditions [3]. Different geometrical detector designs, such as hemispherical or planar detectors, have been developed and have been subjected to environmental and radiation test at the Atominstitut under an IAEA project. CdZnTe has become the material of choice for many applications in radiation detection systems. The critical point of this detector type is the incomplete charge collection which limits the use in spectroscopic applications. This problem arises from the different mobility-lifetime product ($\mu\tau$) of holes and electrons. The mobility-lifetime product ($\mu\tau_h$) of holes is significantly lower than that for the electrons ($\mu\tau_e$). The difference between the collection time of electrons and holes results in an increase of low energy tailing in dependence of energy because holes are trapped very quickly and cannot contribute to the information of a full energy signal. Various methods in electronic correction, in modification of CdZnTe material properties, in contact structure and crystal geometry and arrangements have been tried to minimize this problem.

2.1.1. Co-planar grid detectors

The application of the co-planar grid detectors (CPG) electrode structure creates an electron only collection device that allows for a reduction in tailing caused by the trapping of charge in the CdZnTe crystal. For this technique, the anode is divided into two sets of connected electrode grids, with each set coupled to an independent preamplifier. One set of grids (the collecting anode) is held at a slightly more positive potential than the noncollecting set (Fig. 1). The preamplifier is then connected to a differential amplifier and the resulting

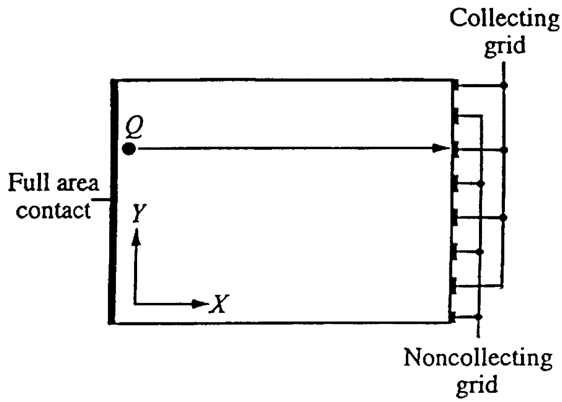


FIG. 1. Basic structure of a CPG detector.

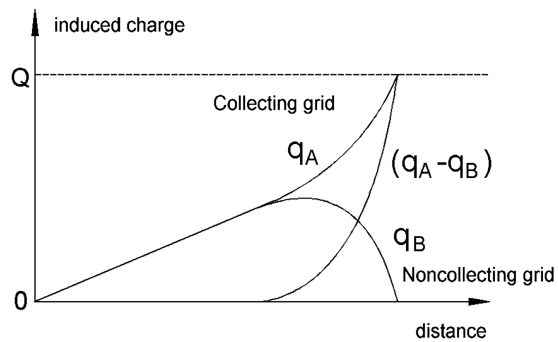


FIG. 2. Induced signal of the two electrodes and the subtracted signal.

signal is fed to an extended shaping amplifier. Spectra recorded with this detector type exhibit a very good peak shape. The absence of low energy tailing simplifies the processing by spectrum evaluation software to extract peak energy and peak area.

2.1.2. Multi-element detectors

The detector structure of the multi-element detectors (CAP), in principle, is planar. But the design is with a full area anode and an extended cathode corresponds to the quasi-hemispherical concept (Fig. 3).

The internal field is modified in such a way that there is a low field region corresponding to the volume defined by the dimension of the cathode

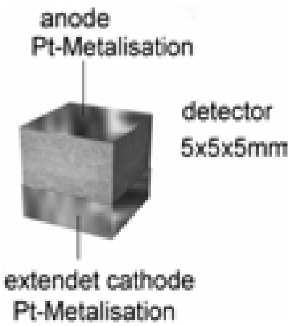


FIG. 3. Structure of one detector element of a CAP detector.

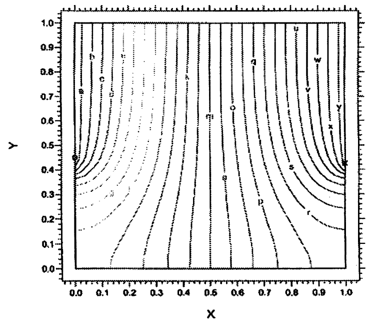


FIG. 4. Resulting electric field distribution.

extension, with the linear, higher field region starting above the extended cathode (Fig. 4). This field structure will result in a predominantly electron only device, at least for interactions taking place in the low field region. Carriers generated in this region will have to traverse the low field, with the holes migrating towards the cathode and the electrons towards the anode. As the lifetime for the electrons is about 10 times longer than for the holes, the electrons have a high probability of traversing the low field region and arriving in the high field region. Once in the high field, the electrons will induce charge in the normal manner on the anode. In contrast, the holes, due to their low mobility and short lifetime, will induce only negligible charge on the cathode.

2.1.3. CZT detector

The CZT detectors have a hemispherical design (Fig. 5) to achieve single charge collection. Their hemispherical geometry modifies the internal field in such a way that the extension of the cathode and the use of a small area anode results in a concentration of the electric field lines in the region of the anode. Electrons generated in the bulk of the detector volume can, due to their relatively long lifetime, travel to the high field region. The induced signal due to this motion is small but increases in proportion to the increased electric field when the electron approaches the anode. The holes travel in opposite directions but contribute much less to the spectrum.

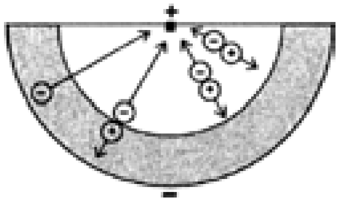


FIG. 5. Schematic drawing of a hemispheric geometry.

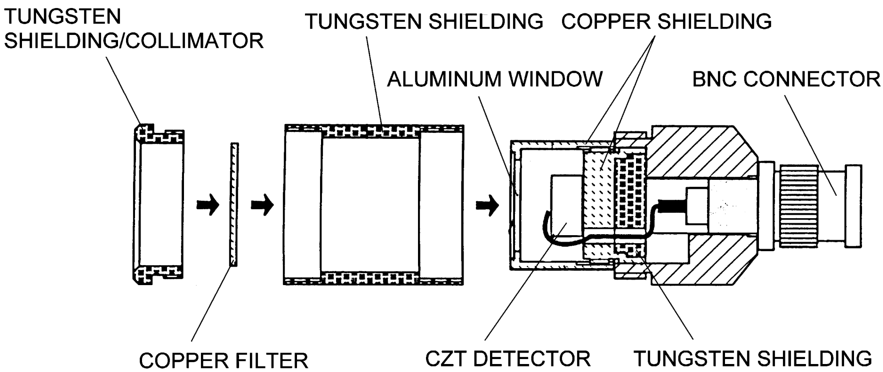


FIG. 6. Cross-section of a CZT 500s.

2.2. Projects involving CdZn- and CdZnTe detectors

2.2.1. Establishment of a gamma spectra catalogue with commercially available CdZn- and CdZnTe detectors [1, 2]

In view of the application of these detectors in safeguards and security surveillance, a spectra catalogue was established using different detector types and different radiation sources. The nuclides were selected according to following criteria:

— Nuclear medical isotopes;
— Isotopes used for illicit trafficking;
— Fission products;
— Activation products during reactor operation;
— Naturally occurring nuclides;
— Nuclides present during decommissioning;

— Measurable half-life;
— Special nuclear material.

In total, about 40 nuclides were used and gamma ray spectra were measured with different detector types. The collected data were imported into a commercial graphic tool to provide the desired standardized output. The catalogue is intended not only for users and application planners, but also for researchers modelling gamma spectra and developing spectrum processing software.

2.2.2. Performance tests of CdTe und CdZnTe detectors under various environmental conditions [3]

The performance of any detector is defined by several parameters, as listed below:

— FWHM as a function of energy;
— Peak area comparison;
— Relative detection efficiency as a function of energy;
— Peak to Compton ratio;
— Relative photo-peak efficiency at 662 kev peak of Cs-137;
— FWHM for 662 keV peak of Cs-137;
— Performance at high and low temperatures;
— High count rate performance;
— Long term stability.

These tests were carried out with five commercially available detectors and allowed the selection of one optimal detector with best performance. Under the same project, different software for isotope identification was tested and some identification problems were eliminated.

2.2.3. Characterization of NaI(Tl) and BGO crystals in view of identification of U and Pu enrichment [4]

The CdTe- and CdZnTe detectors discussed have several advantages for safeguards purposes. However, for other applications such as hand-held isotope identifiers to prevent illicit trafficking, the sensitivity of these detectors is too small and therefore the necessary measuring time for identifying hidden nuclear sources is too long. Therefore, there are plans to combine the good resolution/low sensitivity CdZnTe detectors with the poor resolution/high sensitivity NaI(Tl) or BGO detectors. In this case, the latter detectors have to undergo similar environmental and radiation tests, as mentioned for the detectors in Section 2.2.2.

In addition, two different scintillation detectors, sodium iodide thallium doped (NaI(Tl)) and Bismuth Germanate $Bi_4Ge_3O_{12}$ (BGO), were tested for their application in spectroscopy of special nuclear material (SNM) and for industrial and medical isotopes in view of the development of a new isotope identification programme and a new method to identify the degree of uranium enrichment. After performance tests of the detectors, the optimal detector was chosen for spectroscopy. The purpose was to provide a database of gamma spectra for programmers to support isotope identification software development and testing for use in hand-held devices. Preference is given to spectra of samples, which have many afflictions and are more difficult to obtain for developers and manufacturers, such as gamma spectra of special nuclear material (SNM) and medical isotopes with a short half-life. In addition, the spectra included various ranges of measurement times, sample geometries and absorbers. The recorded spectra were used to test two isotope identification and one enrichment identification software, the latter to be included in the next generation of multipurpose hand-held gamma search instruments/isotope identifiers.

2.2.4. Detectors systems to prevent illicit trafficking at border stations [6]

Since several years and especially after 11 September 2001, the IAEA has initiated a programme to develop detector systems of various technical levels to locate, verify and identify illicit trafficking of SNM or radioactive sources. The detectors systems can be subdivided into:

— Simple pocket-type instruments;
— Hand-held instruments;
— Fixed installed instruments.

The minimum performance requirements and specifications of these detectors are summarized in an IAEA TECDOC (see Ref. [5]). Presently, the technical performance of such detector systems is developing very rapidly and practical experience is obtained from applications at border stations. Out of this, one recent problem to be solved is the so-called 'innocent alarms' originating from medical isotopes incorporated in airline passengers after nuclear medical treatment. Experience at the airport in Vienna showed that approximately 95% of alarms triggered with a fixed installed monitor originate from about three of the most widely used nuclear medical isotopes. Innocent alarms are annoying for customs officials and result in the disregard of alarms.

Therefore, software is tested to suppress alarms from medical isotopes by comparing the spectrum measured with a spectrum library of current medical isotopes. To test the hardware and software, the system was installed and successfully tested at one major Viennese nuclear medical centre, where a number of treated patients with incorporated nuclear medical isotopes pass by daily.

2.2.5. Test of portable radiation search tools at the TRIGA reactor [7]

The portable radiation search tools (PRST) device contains a scintillating fibre optics detector with integrated polyethylene moderator for combined neutron and gamma detection, as well as a separate NaI(Tl) scintillation counter for additional gamma ray indication. After passing the compact counting electronics, the signals from both sensors are evaluated in real time on the builtin display and stored by a flashcard memory for later read-out. The detector parameters can be adjusted according to the expected characteristics of the radiation field to be monitored. A small membrane control panel and a Microsoft® Windows™ based software are available to perform these settings.

The system was designed with maximum sensitivity in a reasonable volume and weight in order to facilitate portable, or at least transportable, application under field conditions. The main purpose of the device is to permit the identification of undeclared radioactive material to support additional protocol verification at facilities under IAEA safeguards. Additionally, the PRST can be employed for nuclear trafficking verifications and, therefore, supports nuclear weapon arms control proliferation prevention.

The following tests for the evaluation of the PRST were carried out:

— Determination of the neutron and gamma ray sensitivity of the device by means of different radiation sources;
— Comparison with other commercial and scientific instruments;
— Verification of the gamma ray discrimination in the neutron channel of the fibre optics detector;
— ‘Close to reality’ test, especially regarding moderation and shielding effects from the human body;
— Functional and preliminary usability test;
— Proposals for necessary modifications in design and procedure.

The radiation sources employed for testing of the PRST units in comparison with other devices include the thermal column of the TRIGA Mark-II research reactor at the Atominstitut der Österreichischen Universitäten, a ^{137}Cs gamma ray source and two neutron sources (unmoderated ^{252}Cf and $^{239}Pu/^{9}Be$) emitting different neutron energy spectra.

2.2.6. Radiation test of IAEA seals and digital surveillance devices [8]

Electronic seals are highly sophisticated sealing devices, however, their application in nuclear facilities under intense radiation conditions for safeguards purposes has always been questionable. The influence of intense gamma and/or neutron radiation for an extended period has been considered as an important failure factor for ultra low power electronic devices. Therefore, it was highly desirable to assess more systematically the reliability behaviour of new types of electronic seals and some other seal components, such as fibre optic cables, etc. under controllable radiation conditions.

Typical investigations carried out in cooperation with the IAEA were:

- Evaluation of the functionality of the VACOSS 5E electronic seal in neutron radiation fields.
- Evaluation of the functionality of the IRES electronic seal prototypes in neutron radiation fields.
- Assessment of the optical properties of the fibre optic sealing cable for COBRA seals after irradiation in gamma and neutron fields.
- Assessment of the functionality of transponder components after irradiation in gamma and neutron fields.

One important task of this project was the investigation and radiation behaviour of the fibre optic cable (FOC). The original application area of this type of fibre optic cable is mainly illumination. It is quite a cheap cable, not intended and not optimized for transfer of information data, therefore, the transmission losses are very high: 650 dB/km—a value which has to be considered for the seal design and application. The mechanical and environmental properties of the FOC are high but not extreme, the operating temperature range is -40 °C to +70° C. Deterioration of the optical properties of the cable starts at higher humidity values; the cable tends to absorb water, especially on the cutting surfaces.

The main optical parameter assessed during this investigation was the transmission factor. In particular, the optical coupling of the FOC to the measuring device appeared to be very critical. This is based on the particular structure of the FOC containing a high number of individual optical fibres.

The evaluation of the field test of the new IAEA's digital surveillance devices showed the significance of the single event upsets (SEU) for the reliability of these highly sophisticated ultra low power electronic devices. The main reason for the failures observed was assumed to be neutron radiation effects in the semiconductor devices of cameras. In order to assess the influence of the neutron radiation on the safeguard containment and surveillance devices (C/S) mentioned, a more detailed evaluation of the neutron dose is necessary. Unfortunately, all standard commercially available neutron dosimeters are based on biological tissue equivalent measurements. Therefore, it is essential to evaluate appropriate new devices, which may deliver Sievert values as a measure for the intensity of neutron generated charged particles responsible for SEUs.

Two commercially available dosimeters produced by a company from the United States of America were custom made neutron micro-dosimeters using Silicon, respectively, boron doped Si radiators in order to measure the charge particle spectra and indirectly estimate the SEU frequency caused by neutron radiation in low power IAEA safeguards electronic equipment implemented at nuclear facilities. The following tasks were completed for the evaluation of the Hawk Si (Si-B) dosimeters:

— Assessment of the sensitivity for the micro-dosimeters relative to other available neutron dosimeters.
— Comparison of the tissue equivalent reading of a neutron rem-counter with the Si equivalent reading of the Hawk for the known neutron spectrum of a Pu-Be source and Am-Be source and a thermal neutron field.

The projects described and a few others were all carried out in cooperation with the IAEA and in the same time resulted in six diploma theses and two PhDs in the period between 1996 and 2002. Most of the cited references are also available in English as an Atominstitut der Österreichischen Universitäten progress report.

3. SUMMARY AND CONCLUSIONS

Small university laboratories have the advantage of low cost but highly qualified personnel who can concentrate on investigation and test series which would largely exceed the possibilities of international organizations and are, therefore, a typical example of successful outsourcing for the benefit of both partners.

ACKNOWLEDGEMENT

The authors want to express their thanks to the IAEA, especially to R. Arlt and R. Carchon for their continuing support and interest in the described projects.

REFERENCES

[1] ZWICKELSTORFER, H., Gamma Spektren Katalog mittels CdZnTe Detektoren Part 1, Diplomarbeit Technical University Vienna, May 2000.
[2] KNABL, M., Gamma Spektren Katalog mittels CdZnTe Detektoren Part 2, Diplomarbeit Technical University Vienna, March 2001.
[3] SWOBODA, M., Evaluierung von CdZnTe Detektoren in Hinblick auf deren Einsatz in tragbaren Isotopenidentifizierungsgeräten, Diplomarbeit Technical University, Vienna, June 2002.
[4] HRKAC, G., Characterization and analysis of NaI(Tl) and BGO crystals and their use in identification of Pu- and U-enrichment, Diplomarbeit Technical University, Vienna, June 2003.
[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Detection of Radioactive Materials at Borders, IAEA-TECDOC-1312, IAEA, Vienna (2002).
[6] SCHRENK, M., Reduction of the frequency of innocent alarms in border monitors, Diplomarbeit Technical University, Vienna, in progress.
[7] HAJEK, M., FUGGER, M., VILLA, M., BÖCK, H., Evaluation of the Portable Radiation Search Tool (PRST), Rep. AIAU 23303, Atominstitut der Österreichischen Universitäten, Vienna (April 2003).
[8] BÖCK, H., Investigation of Radiation Effects on Seals and Performance of NDA Devices, Rep. AIAU 22308, Atominstitut der Österreichischen Universitäten, Vienna (August 2002).

BLANK

CURRENT TENDENCIES AND PERSPECTIVES OF THE DEVELOPMENT OF RESEARCH REACTORS IN THE RUSSIAN FEDERATION

N.V. ARKHANGELSKY
Nuclear Energy Department,
Ministry of the Russian Federation on Nuclear Energy,
Moscow

B.A. GABARAEV, V.V. KCHMELSCHIKOV,
YU. N. KUZNETSOV, I.T. TRETIYAKOV
Research and Development Institute of Power Engineering,
Moscow
Email: tretijakov@entek.ru

Russian Federation

Abstract

For more than 50 years, many research reactors were constructed under Russian projects, and that is a considerable contribution to world reactor building. The designs of research reactors constructed under Russian projects appeared to be so successful, that they permitted the capacity to be raised and the range of their application to widen. The majority of Russian research reactors being middle-aged are far from having their designed resources exhausted and are kept on the intensive run still. In 2000, the so-called "Strategy of nuclear power development in Russia in the first half of the 21st century" was elaborated and approved. The national nuclear power requirements and possible ways of its development determined in this document demanded to analyse the state of the research reactors base. The analysis results are presented in this report. The main conclusion consists in the following: on the one hand, quantity and experimental potentialities of domestic research reactors are sufficient for the solution of reactor materials science tasks; and on the other hand, the reconstruction and modernization appears to be the most preferable way for research reactors development for the near term outlook. At present, modernization and reconstruction work, and work on the extension of operational life of high powered multipurpose MIR-M1, SM-3, IRV-1M, BOR-60, IVV-2M and others are conducted. The Ministry of the Russian Federation on nuclear energy (Minatom) supports the development of research reactors, intended for carrying out the fundamental investigations on the neutron beams. To this end, Minatom gives financial and professional support with a view to complete the reactor PIK construction in PINPh and the reactor IBR-2 modernization in JINR. In the future, the prospect for the research reactors branch in Russia is to acquire the following trends:

limited number of existent scientific centres, based on the construction sites, with high flux materials testing research reactors, equipped with experimental facilities; reactor PIK, satisfying Russian and foreign requirements for beam experiments in the range of nuclear physics, condensed matter physics and other fundamental investigations; pulsed reactors. Eventually, ADS draw growing attention lately; these plants can compete with research reactors as neutron sources for some task solutions. In the paper, the results of new engineering developments by the designs of reactors under reconstruction and modernization are presented, as well as the results for experimental arrangements of different purposes for these reactors: IREN; steamwater loop PVP-3 (reactor MIR.M1); Medical channel for BNCT (reactor IRT, MEPhI). The problem of selection of a high-powered universal research reactor type and design of the next generation also exists in the face of Russian experts. They have five to ten years to solve this problem. Russia is ready for cooperation in the field of development, construction and utilization of new research reactors, which can be built in Europe instead of decommissioned ones.

1. INTRODUCTION

The history of research reactor (RR) utilization commences from the date when the F-1 reactor reached criticality, i.e. late in 1946. The F-1 reactor is a research reactor in the true sense of the words.

World nuclear engineering and technology have passed an enormous pathway of development during the time of their existence. Over this pathway, there were the times of both the ‘nuclear’ boom of the 1960s and 1970s and post-Chernobyl radio-phobia by the end of the 1980s. Research reactors, being an integral part of nuclear engineering, went through those periods together. In that respect, the Russian Federation has made a worthy contribution to reactor construction, preserved its potential and keeps developing it under the complicated social and economic conditions of the last 10 to 15 years.

Chernobyl, as well as social and political shocks, left prints on the 1990s. This period can be remembered in a positive manner in connection with updating the SM-2 (presently SM-3) reactor. Other research reactors of the former USSR partially left the limits of the ‘new Russia’ responsibility zone, but those which remained within its limit were either discontinued or had to keep existing under hard financial conditions.

The rate of RR development and service slowed down that allowed for all interested organizations from Minatom to operating utilities to adapt to new realities. Although the adaptation was painful and possibly is not yet completed so far, this process, as such, is undoubtedly useful. Operating RR have undergone or still keep undergoing ‘natural selection’ for their right to exist. At the same time, safety standards were renewed (made more rigorous).

Some progress in recent years has revealed itself in the development of Russian RR. This is observed in both preparations of the PIK reactor for commissioning and projects on RR updating, in development of works aimed at extending experimental potentialities and area of operating RR application. The ever growing balancing in Russian public attitude to nuclear technologies also furthers this.

2. UP TO DATE STATES OF RESEARCH REACTORS

The Russian Federation possesses 95 nuclear research facilities (NRF), of them 39 are research reactors of different power levels, 41 are critical test-rigs (CTR) and 15 are sub-critical test-rigs (SCTR). Of them, 64 installations are in operation (including those under updating), one is under construction, the remaining ones are either preserved or under decommissioning. Table 1 presents the 'principle' research reactors in service in the Russian Federation.

Out of 39 research reactors, the PIK reactor is under construction, 25 are in operation, 3 are shutdown for updating and 10 are under decommissioning. Updating projects are developed or under development for some operating research reactors and equipment for replacing is manufactured. This will permit minimizing the reactor shutdown period necessary for dismantling works and assembling of replacing equipment. The PIK reactor, the only one under construction in the Russian Federation, is supposed to be ready for commissioning in about two years.

All the Russian RR, except for the IRT-T reactor, are located in the European part of the country: the major portion of them is concentrated in scientific and educational centres possessing developed infrastructure and qualified personnel.

The reactors presently in service operate with different degrees of usage intensity. The utilization factor (UF) of the most powerful RR falls within the range 0.65–0.90. Nevertheless, all in all there is still a reserve in the time for use, scope of experiments and neutron beams with regard to operating RR.

3. PERSPECTIVES

On 31 July 2003, the President of the Russian Federation has stated unambiguously at RFNC VNIIEF in the city of Sarov: "… Russia should be and will remain a great nuclear power"[1]. This means that the Russian nuclear industry and research reactors constituting its integral part have a reliable prospect.

TABLE 1. PRINCIPLE RESEARCH REACTORS* IN THE RUSSIAN FEDERATION

Name	Location	Type of reactor	Year of commissioning/ updating	Power (MW)
VK-50	Dimitrovgrad RIAR	BWR-prototype	1965	200.0
SM-3		Tank	1961/1992	100.0
MIR-M1		Pool/channels	1966/1975	äî 100.0
BOR-60		Fast breeder	1969	60.0
RBT-6		Pool	1975	6.0
RBT-10/1		Pool	1983	10.0
RBT-10/2		Pool	1984	7.0
IVV-2M	Zarechny	Pool-type IRT	1966/1982	15.0
IRV-1M	Moscow region	Pool-type IRT	1974/now	2.0/4.0
IR-8	Moscow RSC ‘KI’	Pool-type IRT	1957/1981	8.0
F-1		Graphite pile	1946	0.024
OP		Tank WWR	1960/1989	0.3
VVR-M	Gatchina, IPPN	Tank WWR	1959	18.0
VVR-TS	Obninsk	Tank WWR	1964	15.0
IRT-MEPhI	Moscow	Pool-type IRT	1967/1975	2.5
IRT-T	Tomsk	Pool-type IRT	1967/1984	6.0
IBR-30**	Dubna, JINR	Fast pulsed	1969/now	0.03/0.022
IBR-2	Dubna, JINR	Fast pulsed	1984/now	2.0/2.0

* Table 1 does not cover RR of lower power levels, among those the pulse reactors of aperiodical action.

** In the course of updating, the IBR-30 reactor is transformed to the ADS facility IREN as an intense resonance neutron source.

RR are known to perform studies on the matters brought up by:

— Power production and heat supply;
— Fundamental science;
— Applied areas of nuclear technologies’ application (in particular, biology, geology, medicine, etc.);
— As well as weapon complex, which is beyond the scope of our consideration.

All the above lines are intrinsic to Russian industry.

Nuclear technology also has its own tasks, for which RR is an irreplaceable tool as well. The matter in question is, first of all, safety of the fuel cycle and the problem related to it, that is, utilization of radioactive wastes (RAW).

3.1. Nuclear power production

On 25 May 2000, the Government of the Russian Federation approved the Strategy of developing nuclear power production in the first half of the 21st century [2], the document governing prospective and lines in the development of nuclear power (NP) in the coming 50 years.

Yu. Roumyntsev, the Minister of Atomic Energy, said in his speech at the 47th session of the IAEA General Assembly:

In last May, the Russian Government approved a fundamental set of documents, which is the power production strategy up to 2020The Strategy says directly, I quote, '…it is expedient for the increased needs of the country economy in electric power to be reimbursed on account of growth in electric power production of nuclear power plants…' This will result in that the share of electric power production by NPP will grow from 15% in 2003 up to 23% by 2020....In this connection, a complete set of measures is being implemented in Russia for substantiating possibility and economic expedience of the works on enhancing safety of nuclear installations in service and extending their service life"[3].

The above measures envisage a set of tasks covering operation safety and optimization of NPP operating cycles to reactor material science, of which the solution cannot but involve the studies with the use of RR.

3.2. Fundamental studies

One of the major factors hampering development of fundamental studies in the Russian Federation is related to financial problems. However, during the recent three years, stable growth is observed in the funds allotted for this line from the federal budget. Understanding the importance of fundamental studies in the Russian Federation, Minatom provides substantial financial support to the Russian Academy of Sciences (RAS) for completing the PIK reactor construction.

Since 2000, Minatom provides partial financing for upgrading the IBR-2 reactor located in the city of Dubna (JINR). The reactor is intended mostly for studying the physics of condensed matter by the time-of-flight method. Development of documentation is presently carried out step by step for individual components of the reactor equipment. Their manufacture has been partially commenced. It is stipulated for the IBR-2 to be discontinued by the end of 2006 for dismantling works and mounting the equipment to be replaced. The restored reactor named IBR-2 is to be brought to power in 2010. Finally, it is planned for the ADS facility IREN (a source of resonance neutrons) intended for studies in nuclear physics to be located in JINR instead of decommissioning the IBR-30 reactor-buster.

3.3. Applied sciences

Russian research reactors are potentially able to render services in solving a tremendous number of tasks having applied character. Similar investigations and works are presently under way as well. However, the long recoupment period of expenses for these investigations and limited free resources available with potential customers restrain their rapid development. The progress in the economy of the Russian Federation recently, in the last two to three years, and optimistic prognosis for the future allow for activation of such studies to be expected. For instance now, the projects on medical beam application have already been developed with the help of reactors VVR-Ts in the city of Obninsk and IRT in MEPhI.

3.4. Safety tasks in operation and fuel cycle

The above tasks are of paramount importance for the country that possesses a substantial nuclear component in the total power production balance. Therefore, safety studies were not discontinued during the most difficult economic period of the 1990s. The electric power generated by Russian nuclear power plants is predicted to grow that will require new studies and tests with the use of RR. These studies are related to the whole set of problems in nuclear power production from updating fuel presently in operation to the matters of spent fuel reprocessing and radioactive waste utilization.

For example, with the object to provide a most representative substantiation of VVER reactor safety, the PVP-3 loop-type facility project is being developed for the MIR-M1 reactor. With the help of a fuel assembly fragment, the facility will allow for the process of design-basis and severe accidents running followed by loss of coolant to be simulated.

For some reasons, the matter of spent fuel utilization is not of paramount importance for the the Russian Federation. The fuel is considered to be a strategic reserve for the future nuclear power rather than waste. Nevertheless, the Russian Federation is ready to consider its participation in international projects in the capacity of both a designer of facilities intended for this purpose and a country providing its services for carrying out experiments with the help of its RR.

Lastly, developing the technology of a closed-type fuel cycle and reactors based on natural (inherent) safety brings forward a large set of tasks for Russian research reactors to study as well.

4. TENDENCIES

The tendencies in development of research reactors in the Russian Federation are governed by plenty of factors. Among those, the following three seem to be most significant:

— Rates and line of nuclear power development;
— The country's economic status;
— Public confidence in safety and usefulness of nuclear power and nuclear technologies.

4.1. Rates and line of nuclear power development

Analyses of Russian research reactors status were carried out by a group of Minatom experts with the object to determine trends and prospective of their development. The work was based on theses of the Strategy [2]. Principles and recommendations of the above group are as follows:

— Research reactors are an integral part of nuclear science and engineering, nuclear power safe and efficient existence and development are impossible without their application;
— Russia possesses a set of research reactors of different types with a large spectrum of experimental abilities. Presently and in the near future, Russian RR will meet the requirements of the nuclear power and fundamental science tasks under study and prospective ones. The PIK reactor commissioning will strengthen this position yet more;
— The common world trend of research reactors ageing is applicable to the Russian Federation to the full extent.
— The Russian RR proved to be so successful that it has allowed for their experimental abilities to be substantially extended through updating;
— The most optimal way for developing research reactors in the next 10 to 15 years is gradual reconstruction and updating of reactors in operation and adaptation to the requirements imposed by experiments.

4.2. The country's economic status

Stable growth of the economy proceeds in the Russian Federation during recent years. The existing forecasts depict a favourable outlook for the future. With such a development of events in the Russian Federation, the energy consumption will increase that will affect the growth of nuclear component in energy balance of the country. On the other hand, nuclear power has indisputable advantage with regard to its influence upon ecology in the absence of severe accidents that should also strengthen its position in the Russian Federation in the future.

4.3. Public confidence in safety and usefulness of nuclear power and nuclear technologies

This factor is extremely important because nuclear technologies applied for meeting the needs and requirements of people are very efficient and very often irreplaceable. In addition to power heat supply, it is a wide spectrum of possibilities for solving particular problems from seawater demineralization in arid regions to radiation medicine. Every success in application of nuclear technologies for satisfying the needs of humankind is objectively in favour of building up positive public attitude to nuclear technologies. It goes without saying, that any accident followed by the release of radioactive substances beyond stated limits and, more than that, people exposure causes irreparable harm to nuclear science and engineering.

4.4. Basic tendencies of development

In the observable future, the reactor research entity of the Russian Federation will involve a limited number of high flux updated facilities presently in operation that will be supplemented with the PIK reactor being able to meet Russian and foreign needs in carrying out beam experiments on the problems of fundamental physics. The pulse reactor line will also be kept on. The ADS facilities will be developed mainly within the framework of international projects. Thus, the principle trends in research reactor development can be formulated in the following manner:

— Research reactors of the Russian Federation presently ensure solving the tasks they are facing and have a reserve for solving the tasks of the near future;

— Development and perfection of research reactors have an encouraging prospect, which is subject to correction depending on the growth rate of nuclear power plants and the country's economic status;
— The problem of the operating research reactors ageing and ensuring safety of their operation is under solution and will be solved in the next five to ten years by way of their updating;
— In addition to the matters of safety insurance, it is envisaged for experimental potential of the facilities to be given attention in the course of the operating research reactors modernization to solve practical tasks;
— The need for creating new research reactors in the near future may happen to be involved in the development of power producing technology for a closed fuel cycle based on power reactors of inherent safety;
— The task of selecting type and design of powerful research reactors of the next generation confronts Russian specialists but they have five to ten years to solve it;
— The Russian Federation is ready to cooperate with the countries interested in both work related to the application and updating of research reactors in operation, as well as the development and creation of research reactors of the new generation in the future.

5. RESULTS OF NEW DESIGN DEVELOPMENTS ON THE PROJECTS OF RR UNDER UPDATING

5.1. Neutron resonance source (IREN) as an example of profound updating

The neutron resonance source (IREN) facility [4] is designed for nuclear and physical studies by the flight-time method within the resonance range of neutron energies. The facility is mounted in situ and inside the building of the IBR-30 reactor under decommissioning. The existing infrastructure of eight neutron beams with flight bases of 10 to 1000 m, experimental hall and pavilions for measurements are kept on and used in full under the IREN project.

The IREN facility operates in the buster mode. The primary energy is generated by the linear electron accelerator (LUE-200), of which the parameters are presented in Table 2. Parameters of fuel blanket (sub-critical assembly) and tungsten target are also given in Table 2.

TABLE 2. IREN FACILITY TECHNICAL DATA

Parameter	Description/value
Basic parameters of LUE-200 accelerator:	
-beam average power, kW	10.0
-electron energy, MeV	200
-pulse current, A	1.5
-pulse duration, ns	≤ 250
-repetition rate, Hz	150
-average acceleration gradient, MeV/m	~35
Basic characteristics of sub-critical assembly (SCA):	
-maximum multiplication factor (K_{eff})	<0.98
-fission thermal power, kW	up to 12
-fuel	Pu
-fuel element diameter, mm	11.2
-height of active portion, mm	180
-number of fuel elements in sub-critical assembly, pcs.	up to 180
-coolant	helium
-end-face reflector in fuel elements	$W(^{10}B)_2$
-side reflector	$Ta\ (^{10}B)_2$
-effective pulse duration of fast neutron, μs	0.42
-neutron generation intensity, 1/s	10^{15}
Target-converter basic parameters of:	
-target material	W
-target dia., mm	38
-target location	in SCA centre
-heat power, kW	<10.0
-coolant	helium

A general view of the facility is presented in Fig. 1 while Fig. 2 gives a 3D image of the sub-critical assembly with target.

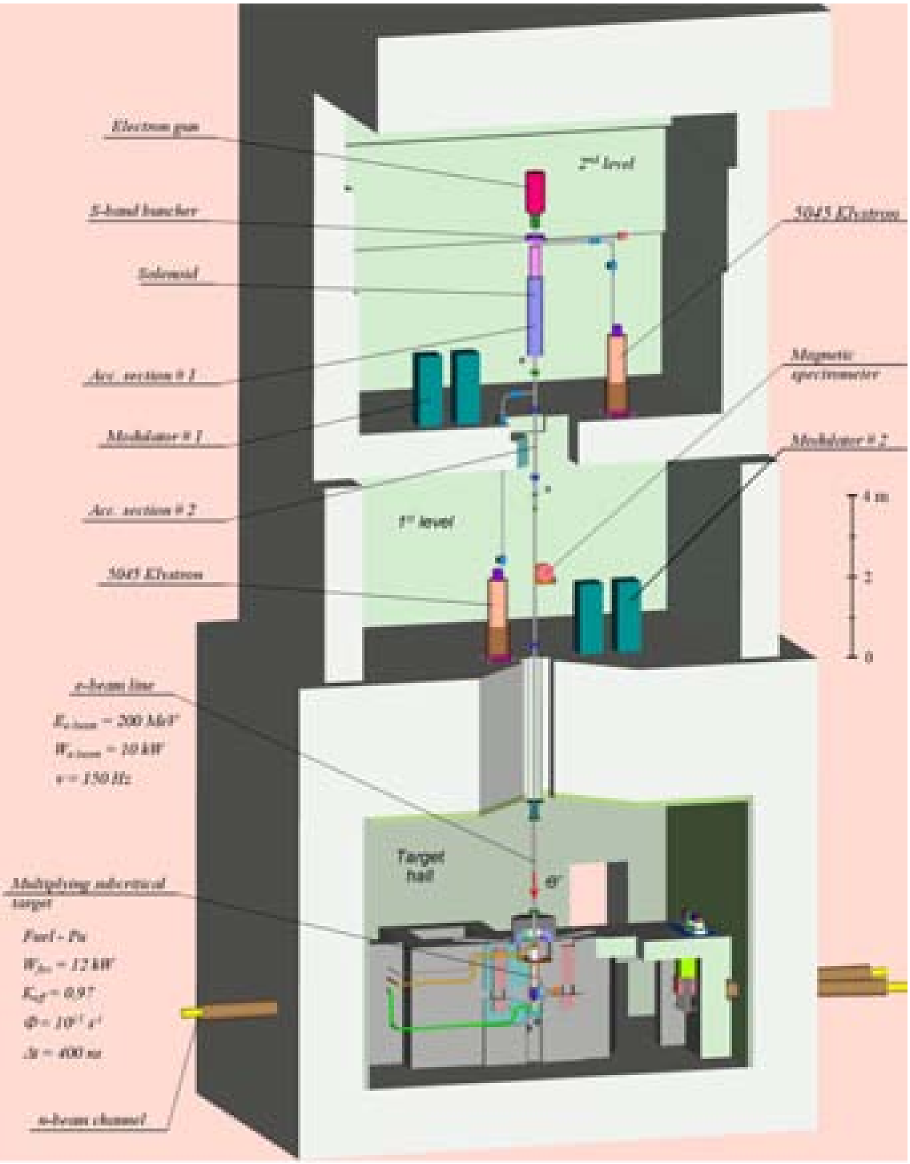


FIG. 1. IREN facility.

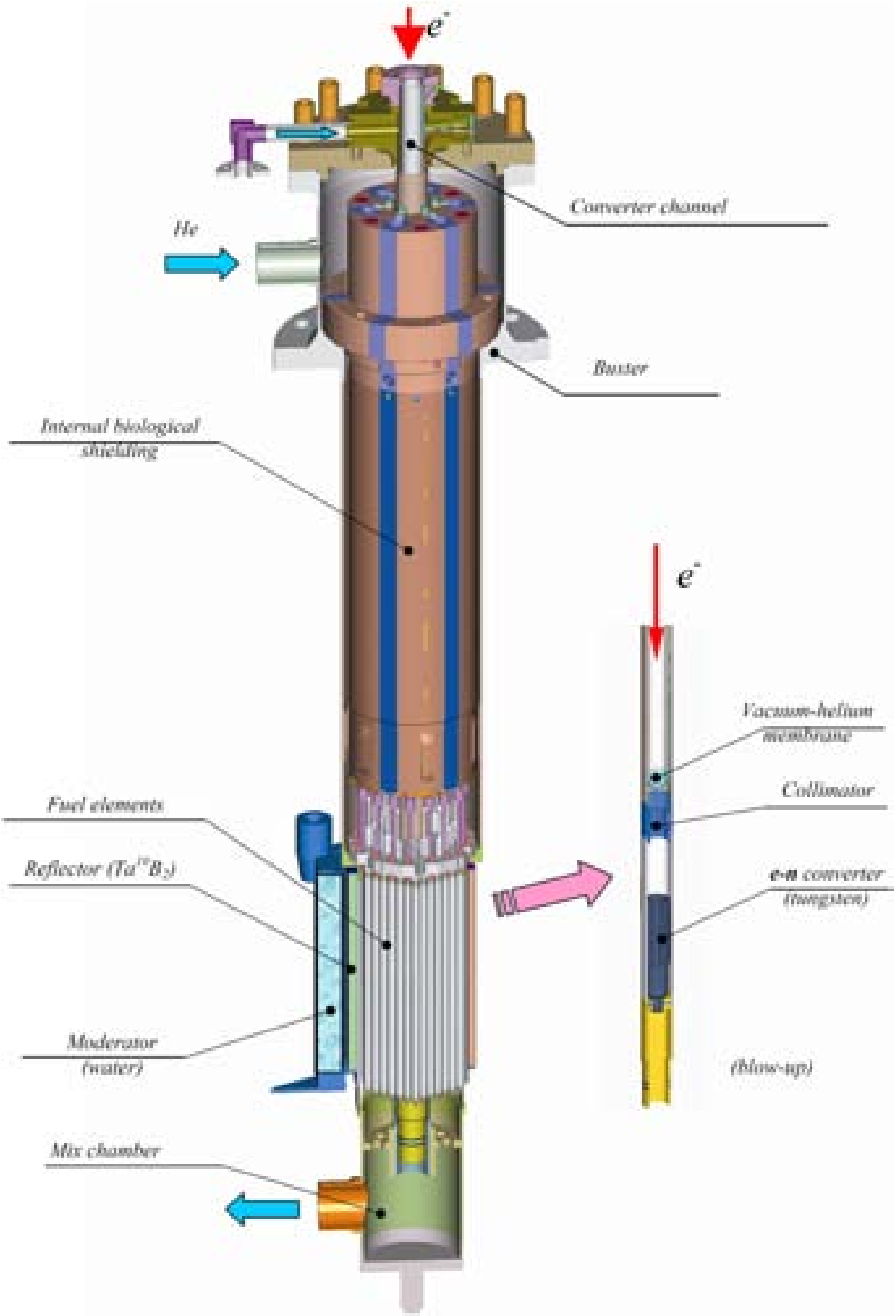


FIG. 2. The multiplying sub-critical core (sectional view).

5.2. PVP-3 steam-water loop (MIR-M1 reactor)

The PVP-3 loop-type facility [5] is planned for application in the MIR-M1 reactor located in the city of Dimitrovgrad. It is designated for in-reactor integrated simulation studies on design accident (including utmost design accident), beyond design-basis (severe) accidents initiated by loss of coolant as applied to the VVER-type reactors. The purpose of the studies is obtaining of proved experimental information to verify and validate computation codes applied for justifying VVER safe operation. Principle sketch of the PVP-3 facility is given in Fig. 3.

The equipment marked in yellow is located in free rooms available in the MIR-M1 reactor building. The experimental channel is installed inside the reactor core proper. Technical data of the facility are given in Table 3.

Creation of the PVP-3 facility will substantially extend abilities of the MIR-M1 reactor as what concerns investigations in safety of fuel for the VVER-type power reactors

5.3. Medical channel for BNCT

The purpose of creating a neutron beam in the IRT-MEPhI reactor is realization of the neutron-capture therapy technology (NCT) for treatment of localized tumours in organs.

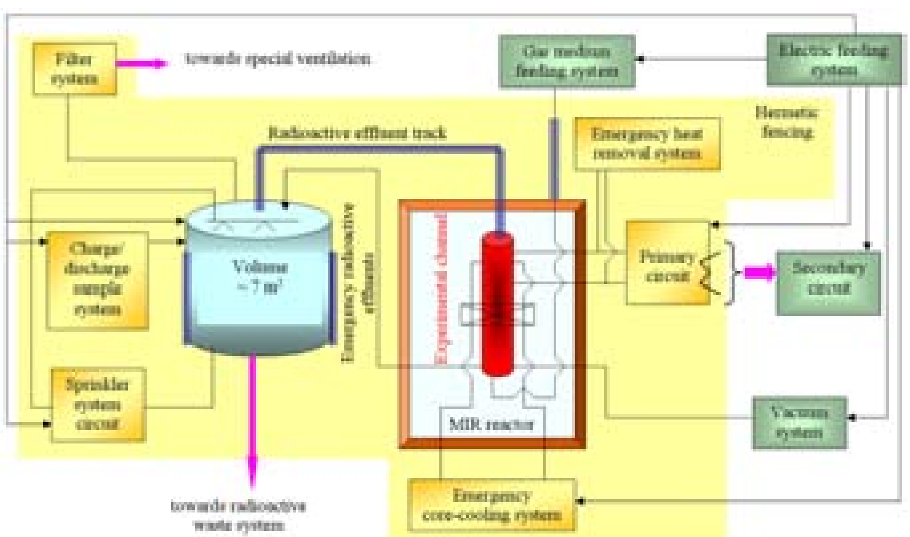


FIG. 3. The PVP-3 loop-type facility scheme.

TABLE 3. PVP-3 TECHNICAL DATA

Parameter	Description/value
Location	MIR-M1 reactor
Thermal power (DBA/SA), kW	max. 500/max. 100
Number of fuel elements in mock-up assembly, pcs	up to 19
Height of active portion in mock-up assembly, mm	1000
Water-loop parameters:	
-pressure, MPa	max. 16.0
-flow-rate, m^3/h	up to 10.0
-water temperature at loop-channel inlet, °C	up to 290
Gas-loop parameters:	
-working fluid	superheated steam, helium
-pressure, MPa	max. 3.5
-working medium flow-rate g/s	up to 3.0
-maximum achievable temperature of working medium, °C	up to 2300
Volume of discharge capacity, m^3	~7

The medical channel [6] is located in the thermal column niche at the IRT-MEPhI reactor and is designed for reliable and safe extraction of the neutron beam having required parameters to exposure premises (experimental hall) at a distance of 2.65 m from the core centre:

$F_t = 3.6\cdot10^9\ cm^{-2}\cdot c$; $F_{et} = 1.1\cdot10^9\ cm^{-2}\cdot c$; $D_n = 3.1\cdot10^{-13}$ Gy·cm^2;
$D = 15\cdot10^{-13}\cdot$Gy·cm^2.
F_t – thermal flux; F_{et} – epithermal flux; D_n – fast neutron dose;
D – photon dose.

The channel consists of the following basic components (Figs 4 and 5):

— Area for beam preliminary formation, involving aluminium-containing unit and lead board (50 mm);
— Slide-valve turning device with a neutron guide, covered with a lead screen (40 mm) in a thin (1 mm) zirconium casing;
— Emergency safety slide-valve;
— Collimating device.

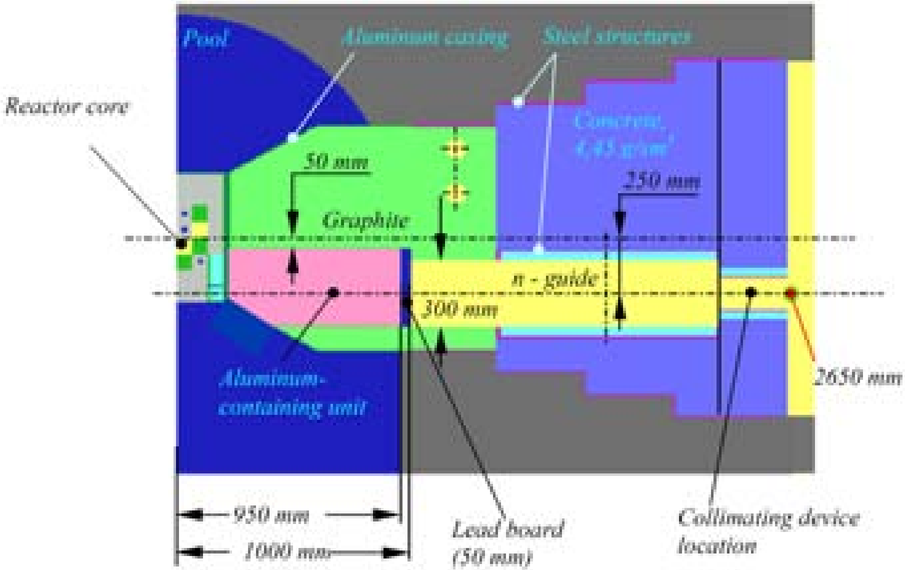


FIG. 4. Horizontal cross-section.

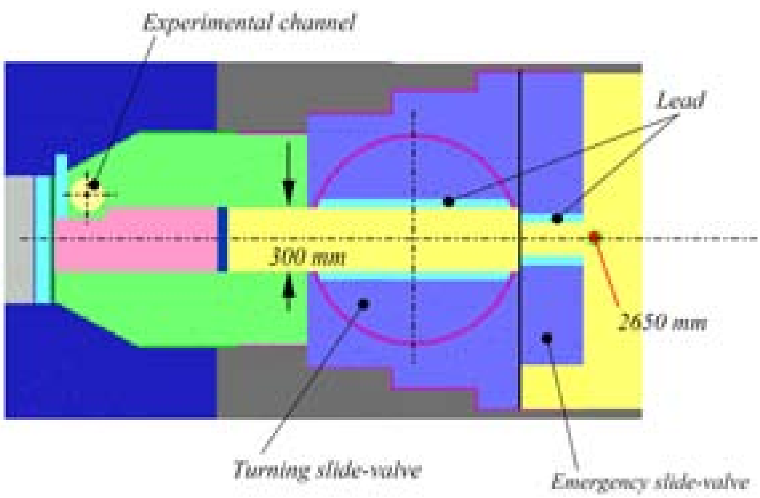


FIG. 5. Vertical cross-section.

REFERENCES

[1] Meeting of President V. Putin of the Russian Federation with scientists of the Nuclear Centre VNIIEF (Sarov, 31 July 2003), http://www.minatom.ru/

[2] MINATOM, Strategy of Nuclear Power Development in Russia During the First Half of the 21st Century, Basic Grounds, Minatom, Moscow (2000).

[3] Request of the Minister of Atomic Energy of the Russian Federation to the 47th session of the IAEA General Conference, 15 September 2003, http://www.minatom.ru/

[4] TRETIYAKOV, I.T., et al., Operational experience of the research reactor-booster IBR-30 and plans for its modification—creation of pulse neutron source IREN.

[5] KUZNETSOV, YU.N., et al., "The in-pile research of design-basis and several accidents in pressurized water reactors", Nuclear Engineering (Proc. 8th Int. Conf. Baltimore, 2000), CD-ROM ICONE8, Baltimore, MD (2000).

[6] BULANOV, A.V., LAPATKIN, A.V., MURATOV, U.G., TREYAKOV, I.I., HACKERESOV, G.A., Medical Channel for Neutron-catch Type Therapy at Reactor IRI-2500 (MPHEI), Yearly rep., NIKIET (ENTEK) Research and Development Institute of Power Engineering, Moscow (2002), p. 53.

RESEARCH REACTOR DECOMMISSIONING AND WASTE MANAGEMENT

(Session 3)

Chairpersons

M. PEŠIC
Serbia and Montenegro

C.S.B. PIANI
South Africa

Co-Chairperson

R. BURCL
IAEA

BLANK

INTERNATIONAL RADIATION SAFETY RECOMMENDATIONS ON DECOMMISSIONING

L. JOVA SED, D. REISENWEAVER
Division of Radiation and Waste Safety,
International Atomic Energy Agency,
Vienna
Email: l.a.jova-sed@iaea.org

Abstract

The IAEA Safety Standards for decommissioning state that the regulatory body shall establish requirements for the decommissioning of nuclear facilities, including conditions on the end points of decommissioning. One of the main important issues is that the operator shall be responsible for all aspects of safety of the facility during its lifetime, including the decommissioning activities. The paper mainly addresses the activities associated with the decommissioning of research reactors, primarily with decommissioning after planned final shutdown. It is intended to provide guidance to national authorities and operating organizations for the planning and safe management of the decommissioning of such installations.

1. INTRODUCTION

More than 670 research reactors have been built to date, of which some 278 are now operating in 59 countries. A large number of these research reactors will be facing permanent shutdown in the near future. In the past, decommissioning of nuclear power plants and research reactors was performed on an ad hoc basis using the same framework of regulations as applied during the operational period of a reactor. In order to provide a consistent and harmonious approach to the decommissioning of research reactor installations, as well as to incorporate lessons learned from previous decommissioning efforts, IAEA Member States have expressed the need for decommissioning guidance within the context of an internationally accepted programme of safety standards.

Safety requirements for the decommissioning of nuclear power plants and research reactor facilities are currently addressed in IAEA Safety Standards Series No. WS-R-2, Predisposal Management of Radioactive Waste, Including Decommissioning Requirements [1]. IAEA Safety Standards Series No. WS-G-2.1, Decommissioning of Nuclear Power Plants and Research

Reactors Safety Guide [2], provides recommendations, on the basis of international experience, to assist in fulfilling the basic requirements for decommissioning as set out in Ref. [1]. IAEA Safety Reports Series No. 26 on Safe Enclosure of Nuclear Facilities During Deferred Dismantling [3] was recently published. This report provides information that will help ensure that a nuclear installation, which has been or will be placed into a safe, long term storage mode is maintained in a safe state until the final decommissioning activities are performed.

2. REQUIREMENTS

The IAEA safety requirements for decommissioning [1] state that the regulatory body shall establish requirements for the decommissioning of nuclear facilities, including conditions on the end points of decommissioning. One of the main requirements is that the operator shall be responsible for all aspects of safety of the facility during its lifetime, including the decommissioning activities until its completion.

A mechanism for providing adequate financial resources shall be established to cover the costs of radioactive waste management and, in particular, the cost of decommissioning. It shall be put in place before operation begins and shall be updated, periodically.

A safety assessment of the proposed decommissioning strategy shall be performed and the strategy's implementation shall not begin until approval has been received by the regulatory body. The safety and environmental impacts of the decommissioning activities shall be assessed and managed so that hazards are mitigated.

Unless otherwise required by the regulatory body, a decommissioning plan shall be prepared for each facility, to show that decommissioning can be accomplished safely. Account shall be taken of the eventual need to decommission a facility at the time it is being planned and constructed. The decommissioning plan shall be reviewed regularly and shall be updated as required to reflect, in particular, changes in the facility or regulatory requirements, advances in technology and the needs of decommissioning operation. The preferred option for the decommissioning of research reactors is the immediate dismantling strategy. If it is intended to defer the immediate dismantling, it shall be demonstrated in the final decommissioning plan that such an option is safe. If the shutdown of a facility is sudden, it shall be brought to a safe state before the decommissioning strategy is implemented in accordance with an approved decommissioning plan.

Decontamination and dismantling techniques shall be chosen which minimize waste and appropriate means shall be in place for safe managing any waste that might be generated during the decommissioning process. A quality assurance programme shall be established for the decommissioning process.

Before a site may be released for unrestricted use, a survey shall be performed to demonstrate that the end point conditions, as established by the regulatory body, have been met. If the site cannot be released for unrestricted use, appropriate control shall be maintained to ensure protection of human health and the environment.

The IAEA Safety Guide [2] mainly addresses the radiological hazards resulting from the activities associated with the decommissioning of nuclear reactors, primarily with decommissioning after a planned final shutdown. Many of the provisions are also applicable to decommissioning after an abnormal event that has resulted in serious facility damage or contamination. In this case, the Safety Guide may be used as a basis for developing special decommissioning provisions, although additional considerations will be necessary.

3. CONSIDERATIONS DURING REACTOR OPERATION

The requirements for decommissioning should be considered at the design stage for a new reactor installation or as soon as possible for existing installations. The later in the reactor lifetime consideration is given to facilitating decommissioning, the more difficult and costly decommissioning may become. This may be due to a lack of adequate records and information, the need to install or modify equipment, the increased complexity of decommissioning activities, and the incurring of unnecessary radiation doses as a result of aspects of the design interfering with decommissioning activities.

Accurate and relevant records should be kept of the operating phase of the installation in order to facilitate the successful implementation of decommissioning. If these records have not been or are not being maintained, such record keeping should be initiated as soon as possible. During operations, consideration should be given to minimizing the extent of contamination of structures and surfaces, segregation of different categories of wastes, and avoidance and prompt cleanup of spillages and leaks.

4. PLANNING AND SAFETY ASSESSMENT FOR DECOMMISSIONING

Successful decommissioning depends on careful and organized planning. As pointed out previously, a decommissioning plan should be prepared for each reactor. The extent of such plans, their content and the degree of detail required may differ, depending on the complexity and hazard potential of the nuclear installation, and should be consistent with national regulations. Three stages of planning are envisaged: initial, ongoing and final. For a given reactor, the degree of detail will increase from the initial to the final decommissioning plan. This planning process will result in the production of a decommissioning plan, as described in the following discussion.

4.1. Initial planning

An initial plan for decommissioning should be prepared and submitted by the operating organization in support of the licence application for the construction of a new reactor, although the level of detail in the initial plan will necessarily be lower than that in the final decommissioning plan. The main focus of this plan will be to establish the amount of funds needed to eventually implement the decommissioning strategy. In cases where an operational plant does not have an initial plan for decommissioning, a decommissioning plan reflecting the operational status of the installation should be prepared without undue delay.

4.2. Ongoing planning

During the operation of a reactor, the decommissioning plan should be reviewed, updated and made more comprehensive with respect to technological developments in decommissioning, incidents that may have occurred including abnormal events, amendments in regulations and government policy, and cost estimates and financial provisions. The decommissioning plan should evolve with respect to safety considerations, based on operational experience and on information reflecting improved technology.

4.3. Final planning

When the timing of the final shutdown of a nuclear reactor is known, the operating organization should initiate detailed studies and finalize proposals for decommissioning. Following this, the operating organization should submit an application containing the final decommissioning plan for review and approval by the regulatory body. The decommissioning plan may require amendments or further refinements as decommissioning proceeds, and may require further regulatory approval. A list of items to be considered for the final decommissioning plan is presented in [2].

5. SAFETY ASSESSMENT FOR DECOMMISSIONING A RESEARCH REACTOR

A safety assessment should form an integral part of the decommissioning plan. The operating organization is responsible for preparing the safety assessment and submitting it for review by the regulatory body. The safety assessment should be commensurate with the complexity and potential hazard of the installation and, in case of deferred decommissioning, should take into account the safety of the installation during the period leading up to final dismantling.

Radiological and non-radiological hazards involved in the proposed decommissioning activities should be identified in the safety assessment leading to the definition of protective measures to ensure the safety of workers, the public and the environment, i.e. to meet national requirements for radiation protection, nuclear safety and other safety and environmental protection requirements. The protective measures may require the established safety systems for operational installations to change, but the acceptability of such changes should be clearly justified in the safety assessment.

The safety assessment will aid in the identification of engineering and administrative arrangements that should be in place to ensure the safety of the decommissioning process during all phases of the implementation of the decommissioning strategy and will aid in the choice of a particular decommissioning option.

If deferred dismantling is instituted, the safety documentation and decommissioning plans should be periodically reviewed to ensure that they represent current installation conditions and regulatory requirements.

The safety assessment may identify a number of significant non-radiological hazards during the decommissioning phase that are not normally encountered during the operational phase of a reactor. These include, for example, hazardous materials that may be used during decontamination, dismantling and demolition activities, and the lifting and handling of heavy loads. Most of these non-radiological hazards will be covered by regulations, but a good safety culture will help to ensure that such tasks are carried out safely.

6. FINANCIAL ASSURANCE FOR DECOMMISSIONING

The operating organization shall plan for adequate financial resources to ensure the decommissioning of a nuclear reactor. Especially in the case of deferred decommissioning, where there may be long safe enclosure periods, these financial provisions should be reviewed periodically and adjusted as necessary to allow for inflation and other factors such as technological advances, waste costs and regulatory changes. Responsibility for this review may reside with the operating organization, the regulatory body or other parties, depending on the national legal framework.

The cost of decommissioning should reflect all activities described in the decommissioning plan, for example, planning and engineering during the post-operation phases, the development of a specific technology, decontamination and dismantling, conducting a final survey, and management of radioactive waste. The cost of maintenance, qualification of personnel, surveillance and physical security of the reactor installation should be taken into account, especially if any phase of decommissioning is deferred for an extended period of time. For existing research reactor installations with no financial assurance mechanism for decommissioning, such a mechanism should be established without undue delay.

7. CRITICAL TASKS ON DECOMMISSIONING

A survey of radiological and non-radiological hazards is an important input for the safety assessment and for implementing a safe approach during the work. It should be conducted to identify the inventory and location of radioactive and other hazardous materials. In planning and implementing surveys, existing records and operating experience should be used to provide the necessary data. A characterization report should be prepared which documents the information and data obtained during the characterization process. The report should be retained as part of the official records of the installation. An inventory of all hazardous material present in the installation should be conducted and documented.

The removal of nuclear fuel from the reactor installation at the end of its operational lifetime should preferably be performed as part of operations. Its timely removal from the installation is beneficial and will simplify monitoring and surveillance requirements. Other activities associated with decommissioning may be conducted concurrently with fuel removal, but potential interference should be evaluated. If the fuel is not removed as part of operations, the decommissioning plan and cost estimates must reflect this.

At the beginning of decommissioning, all readily removable radioactive sources (operational waste, sealed sources, liquids) should be removed for reuse, storage in approved location or disposal. The removal of sources will normally result in a significant reduction of the radiation hazards.

In the case of safe enclosure, structures and systems may have to perform their safety functions for longer periods than their initial accepted design life. This is important for active containment devices. Care should be taken to ensure that proper maintenance is performed and to assess their integrity and efficiency regularly. Similar considerations may also apply to non-radiological hazards that may arise in the installation, including those due to toxic material, flammable liquids or vapours, heavy metals or asbestos.

Decommissioning of a reactor may be aided at certain stages by partial or total decontamination. Decontamination may be applied to internal or external surfaces of components and systems, structural surfaces and the tools employed in decommissioning. The process of decontamination associated with decommissioning can be conducted before, during or after dismantling. The decontamination should be optimized, taking into account personnel exposures, cost, safety of technique and waste generated.

Dismantling may create new hazards that should be considered. Dismantling techniques should be simple, reliable and proven. It is important that these techniques minimize the generation of liquids and secondary waste as well as the adverse impact on surrounding area and systems.

Maintenance is important during deferred decommissioning since part of the safety of the installation may rely on systems that have to retain their capability to perform for extended periods of time. Periodical monitoring of all the safety related components of the installation should be incorporated into the decommissioning plan.

At the completion of the decontamination or dismantling activities, a survey of the residual radionuclides at the reactor site needs to be performed to demonstrate that the residual activity complies with the criteria set by the national regulatory authority and the decommissioning objectives have been fulfilled. This survey may be carried out in phases, as decommissioning work is completed, to enable parts of the site to be released from regulatory control. However, it must be ensured that these released areas are not re-contaminated due to subsequent activities.

8. MANAGEMENT DURING DECOMMISSIONING

The operating organization needs to have, or have access to, competent staff to cover areas such as safety requirements of the licence, radiation protection, waste management, quality management, etc. Personnel should be competent to perform their assigned work safely. The management and staff involved in the decommissioning project should be made aware of and trained, if necessary, in the methods of minimizing the waste generated in the assigned tasks. Such methods include installation of contamination control tents, containment of spills, and segregation of radioactively contaminated waste from those wastes that are not radioactively contaminated.

The safety assessment should consider the consequences of there being insufficient personnel with plant specific expertise. It would be of benefit to make use of personnel with experience in both operation and decommissioning. In some cases, contractors may be used to carry out all or some aspects of decommissioning. This is likely to occur when decommissioning is deferred or when plant personnel may not have the required expertise. Financial considerations may also require a greater use of contractors. Examples of such activities include the use of specific decontamination processes and dismantling/demolition activities. Appropriate levels of control, supervision and training must be provided to ensure safety.

The organizational structure to be employed during decommissioning should be described in the decommissioning plan. In the description of the organizational structure, there should be a clear delineation of authorities and responsibilities among the various units. This is particularly necessary when the operating organization uses outside contractors. In this case, all licence conditions apply to them. It must be remembered that the licence always maintains overall responsibility of safety, even if contracts are used.

A team composed of decommissioning specialists and appropriate site personnel should be formed to manage the decommissioning project. Although new competences may be required for the decommissioning phase, attention should be given to the retention of key personnel who are familiar with the installation during its operational phase. Since deferred decommissioning may continue for several years, it is essential to document the historical knowledge represented by personnel associated with the reactor installation before final shutdown. This information should be accessible to decommissioning workers for use during active decommissioning phases.

The radiation protection programme should be clearly discussed in the decommissioning plan. This programme should ensure that radiation protection is optimized and that doses are kept within appropriate limits. Although the principles and aims of radiation protection during operations and during decommissioning are fundamentally the same, the methods and procedures of implementation of the radiation protection may be different. During decommissioning, special situations may need to be considered, which may require the use of specialized equipment and the implementation of certain non-routine procedures. Those involved in the execution of the radiation protection programme need to be properly trained and have access to appropriate equipment for carrying out radiation surveys, including equipment for measuring external dose rates and surface contamination levels and for sampling air concentrations. The radiation protection programme should be periodically reviewed.

All decommissioning work is planned and carried out using work procedures and radiation work permits, with adequate involvement of radiation protection expertise to determine the required radiation protection measures. Moreover, the promotion of awareness of safety issues should be accorded high emphasis in planning and implementation. Those charged with the day to day responsibility for radiation protection should have the resources, access to decommissioning management and the independence necessary to effect an adequate radiation protection programme. In order to control all decommissioning activities, the operating organization should document and record the results of the main activities.

The decommissioning plan should specify the requirement for on-site and off-site monitoring during decommissioning. On-site monitoring should provide information to identify and assist in mitigating the radiological hazards. It should also be used in the planning of specific decommissioning activities. It should ensure that all potential release points are monitored. On-site monitoring should consist not only of personnel monitoring but also of spatial monitoring for airborne contaminants. The off-site monitoring programme inherited from the operational period will require modification appropriate to the conditions existing during decommissioning. Discharges of radionuclides via airborne and liquid pathways must be controlled, monitored and recorded, as required by the regulatory body or other relevant competent authority. On-site and off-site monitoring, radiation and contamination surveys, as well as safety analyses and assessments, should be used to gauge the expected and actual degree of safety associated with decommissioning activities.

Consideration should be given to optimizing waste management and minimizing cross-contamination and secondary waste generation. The radiation exposure to workers and the public may vary according to the waste minimization strategy. An integrated approach should be used to balance waste minimization goals with the objective of keeping radiation exposures as low as reasonably achievable. The different categories of waste should be managed through pathways that are proven to be adapted to their characteristics and toxicities (radiological and non-radiological). Reuse and recycle strategies have the potential of reducing the amounts of wastes to be managed. Similarly, the release of low activity material from regulatory control (clearance) as ordinary waste or for reuse and recycle can also substantially reduce the amount of material which has to be considered for disposal as radioactive waste. The removal of regulatory controls should be done in compliance with criteria established by the national regulatory authority. Guidance on criteria for the removal of regulatory controls and on the management of the regulatory process for removal of controls is being developed by the IAEA. The decommissioning plan should include provisions for responding to all emergencies which might occur during decommissioning. Some of these might be different from those considered during operation.

An appropriate quality assurance programme should be planned and initiated by the operating organization before the decommissioning of a research reactor commences. A description of the quality assurance programme, including a definition of its scope and extent, should be included as part of the decommissioning plan, and be put into effect before the start of decommissioning. All significant changes affecting systems, structures and components important to safety during the operation should be documented for use in the planning for decommissioning.

The acquisition and retention of records and information relevant to the reactor site should be emphasized in the development of a quality assurance programme for decommissioning. Records should be retained as appropriate to meet the needs of future decommissioning and as dictated by national requirements. Where long periods of storage are anticipated, records should be periodically checked. In the case of extended periods of safe enclosure, accurate and complete information relating to the locations, configurations, quantities and types of radioactive materials remaining at the reactor installation are essential and should be maintained. For deferred dismantling, the reports should specify the future maintenance and surveillance activities, as well as the need for the documentation of the results of these activities.

9. COMPLETION ON DECOMMISSIONING

On completion of decommissioning, it is necessary to verify that end state conditions have been met and appropriate records should be retained. In accordance with the national legal framework, these will be held and maintained for purposes such as confirmation of completion of the decommissioning activities in accordance with the approved plan; recording the disposal of waste, material and premises; and responding to possible liability claims. The records to be assembled should be commensurate with the complexity of the installation being decommissioned and the associated hazard potential. The site should be controlled until approval is received from the regulatory body to release it from regulatory control.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Predisposal Management of Radioactive Waste, Including Decommissioning, IAEA Safety Standards Series No. WS-R-2, IAEA, Vienna (2000).
[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Nuclear Power Plants and Research Reactors, IAEA Safety Standards Series No. WS-G-2.1, IAEA, Vienna (1999).
[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Safe Enclosure of Nuclear Facilities During Deferred Dismantling, Safety Reports Series No. 26, IAEA, Vienna (2003).

BLANK

IMPLEMENTATION OF STAGE 3 DECOMMISSIONING OF THE TRITON FACILITY

E. LOPES, L. PILLETTE-COUSIN
Section Assainissement du Site, Unités de Soutien Logistique et Technique,
Direction de la Recherche Technologique – Ile de France,
Commissariat a l'énergie atomique (CEA),
Fontenay-aux-Roses, France
Email: lucien.pillette@cea.fr

Abstract

An extensive decommissioning programme has been launched by the French Commissariat a l'énergie atomique (CEA), including decommissioning of nuclear facilities related to the front-end and the back-end of the nuclear fuel cycle, and experimental reactors. This programme includes the stage 3 decommissioning of the Triton facility, comprising two piles, Triton and Nereïde, and a hot cell, located at the CEA research centre of Fontenay-aux-Roses. This facility was built in the late 1950s, primarily for R&D activities related to neutron physics studies, and radiological shielding experiments. As of 1982, stage 2 decommissioning started, was completed in 1986 and since that date, no work has been done in the facility. Extensive studies have been carried out to acquire better knowledge of the facility's radiological status, in order to set up specific zoning for dismantling and to reduce amounts of generated radioactive waste. The selected dismantling scenario is provided, followed by some conclusions of the security assessment studies, taking into account a number of significant non-radiological hazards. The preliminary dismantling operations performed in 2002 on concrete structures and decontamination of the hot cell walls are described. Significant experience feedback has already been acquired.

1. INTRODUCTION

The French Commissariat a l'énergie atomique (CEA) research centre of Fontenay-aux-Roses was created in 1946 at the time that the French nuclear energy programme began. Two generations of facilities have been built and have operated. The first generation of facilities remained operational for 15 years and was dismantled in the late 1950s. It was replaced by a new generation of facilities, as part of the French electronuclear programme, and included the Triton and Nereïde research reactors (hereafter referred to as the Triton facility).

In accordance with the CEA strategy and taking into account its urban location, the CEA centre of Fontenay-aux-Roses decided in 1998 to launch an extensive cleanup programme to be implemented from 2010 onwards. This includes the stage 3 decommissioning of the Triton facility.

This decommissioning was decided at the earliest stage of the programme. The CEA centre of Fontenay-aux-Roses aims at a strong development of high technologies for the industry (integrated systems) and of life sciences, notably concerning the Prion disease research, neurological diseases (for example, Alzheimer's, Parkinson's) and studies to understand the effect of radiation on living matter.

These activities would soon require considerable construction to install laboratories and support activities.

2. TRITON AND NEREIDE DESIGN AND HISTORY

The Triton facility was built in the late 1950s. The Triton pile was first operated in June 1959. It was a 6 MW research reactor working with highly enriched uranium as fuel, and light water as moderator and coolant. The Nereïde pile was first operated in 1960. It was a 600 kW pile also using enriched uranium as fuel and light water as moderator and coolant.

The Triton pile was used mainly for radioelement production, radiological protection studies, neutron diffraction experiments (see Fig. 1) and

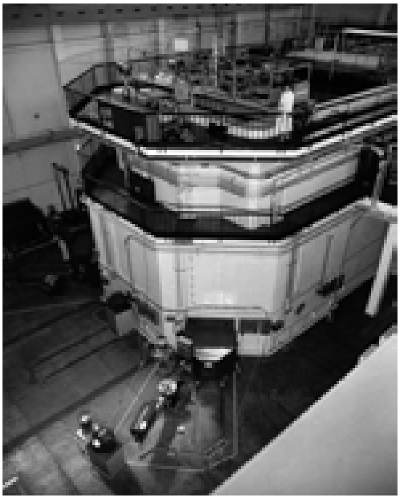


FIG. 1. Physics experiment at the Triton facility.

fundamental physics experiments. Five neutron beam ports were built in the concrete radiation shielding of the Triton pool to make neutron experiments possible (see Fig. 2).

The Nereïde pile was a mobile system that in remote position made the preparation possible of experiments in a dry pool called Naïade. In contact position, the Nereïde pile was used to irradiate experimental devices. The Nereïde pile and Naïade pool were mainly used for radiation protection experiments and for studies regarding graphite gas sub-critical assemblies.

A hot cell equipped with remote handling systems was built against the Triton pool. It was used to transfer spent fuel and activated materials and equipment from the pool, for conditioning and packaging, before ‘evacuation’.

The Triton and Nereïde piles were shut down in May 1982 and December 1981, respectively. From 1983 to 1986, stage 2 decommissioning operations of the Triton facility were performed.

The Triton facility, which had been registered in the records as a basic nuclear installation (BNI) no. 10, was removed from the BNI list in 1987. It was reclassified on the basis of environmental protection concerns. In 1993, it was decided to perform a complementary decontamination of the hot cell and to remove the spent radioactive sources present in the facility, in order to send them to an interim storage facility. As of 1994, the Triton facility was definitively closed and from that date, only a simplified radiation monitoring of the facility has been performed. In March 2001, the chief executive of the CEA centre of Fontenay-aux-Roses decided to launch the stage 3 decommissioning of Triton.

FIG. 2. Triton pile with the five neutron beam ports.

3. TRITON STAGE 3 DECOMMISSIONING PROJECT

3.1. Decommissioning plan

Following international recommendations, a decommissioning plan, i.e. a conceptual study including the technical and economic analysis of the decommissioning, was drawn up in 2001: it was called the preliminary project study. At this time, the plan was to dismantle the entire facility by cutting it into blocks to be sent to an interim storage facility as very low level waste. As a result, the stage 3 decommissioning of the Triton facility would have generated about 2000 m^3 (4000 metric t) of very low level waste, mostly concrete. No low or medium level waste was expected.

Based on the history of the facility, its operating conditions and its radiological inventory, it was decided in 2002 to change the decommissioning strategy. Thus, new studies were carried out, based on a comprehensive radiological characterization of the facility and on the implementation of zoning called specific zoning for dismantling. The zoning would aim at defining workshops generating conventional waste and others generating radioactive waste. A specific zoning for dismantling is composed of three parts:

(1) The first part, describing the risk points, their history and radiological status;
(2) The second more analytical part, precisely identifies the past events having consequences on the radiological status: at this point, a preliminary specific zoning can be established;
(3) The third part makes it possible to finalize the specific zoning by means of radiochemical analysis of samples, use of calculation codes, in situ measurements by gamma spectrometry.

3.2. History of facility

National and local archives and documentation were collected, including historical documents, safety documents, photographs, slides and video records. Documents (such as records, plans, etc.) cover all the life phases of the plant: design, construction, operation, shutdown, stage 2 decommissioning and modifications of the installation that have occurred since the end of stage 2 decommissioning.

The collection of evidence from workers was organized. Evidence was obtained from retired employees as well. The evidence collected has allowed the project team to bring consistency to the data collected.

3.3. Radiological characterization of the Triton facility

The artificial radioactivity present in the facility comes from two distinct sources:

(1) The first origin is surface contamination due to the contact of inner surfaces of the Triton pool with contaminated water, or contact of the inner walls of the hot cell with contaminated dust. The contamination is composed of fission products (^{137}Cs, ^{90}Sr) and activation products (^{60}Co and ^{63}Ni).
(2) The second type is neutron activation inside the concrete radiological shielding surrounding the Triton pile and particularly around and along the five neutron beam ports, due mainly to the neutron streaming effect. The main radionuclides are ^{60}Co, ^{63}Ni, ^{55}Fe, ^{152}Eu and ^{3}H.

The radiological status of the facility was based on:

— A set of measurements carried out at the end of the stage 2 decommissioning in 1987;
— A second set of measurements performed in 1998, which gave more consistency and credibility to the 1987 measurements.

Indeed, following complementary cleanup work in the hot cell, the dose rate into the cell dropped significantly, from 450 μGy/h in 1987 to values lower than 0.1 μGy/h in 1998.

To evaluate the surface contamination, the 1998 inventory was used to produce a radiological photograph of all parts and workpieces of the facility, in which hundreds of dose rate measurements, smear tests, direct surface contamination measurements, and more than 500 samplings and core drillings were performed. Moreover, an in situ gamma spectrometry method was developed that made it possible to measure 10 m^2 of wall surface at a time. It was successfully used to identify the ‘major’ gamma emitting nuclides present in the surface contamination and to confirm results obtained from direct surface contamination and sampling methods. Dozens of such gamma spectrometry measurements were carried out. Figure 3 shows the Lamas, a mobile laboratory unit ensuring in situ measurements outside or inside facilities. Figure 4 shows a gamma spectrometer, connected to the Lamas, measuring surface and shallow deep gamma activity of internal walls of the Triton pool, with models developed by CEA.

To evaluate the neutron activation in the radiation shielding around the Triton core, both calculations and measurements were performed.

Calculations were used to obtain a profile of neutron activation distribution inside the radiation shielding but not the absolute values of the specific activities. The calculations were made by experts of the Reactor Studies and Applied Mathematics Division of CEA (CEA/DEN/SERMA), using a model of the Triton pile concrete shielding and iron structure combined with a core model. The TRIPOLI-4 code was used to compute neutron transport and the DARWIN-PEPIN-2 code made it possible to calculate the activities and decay of the activation products.

Dozens of core drillings and samplings were performed. The activities of beta and beta-gamma emitters were determined by laboratory methods. Both in the calculation and in the measurements, the predominant activation

FIG. 3. Lamas, mobile laboratory unit for site characterization.

FIG. 4. Gamma spectrometry of inner Triton wall.

nuclides were found to be ^{52}Eu, ^{154}Eu, ^{60}Co, ^{63}Ni, ^{55}Fe, and ^{3}H. The calculated relative profile of the activity distribution within the concrete shielding was in good agreement with experimental results.

Finally, the radiological characterization was completed. Areas and workpieces where significant radioactivity remained were called 'risk points'. These 'risk points' were studied in detail, to prepare the dismantling operations, as per the ALARA approach, and in order to define 'specific zonings for dismantling'.

3.4. Specific zoning for dismantling

The radiological characterization described above made it possible to define the reference zoning of the Triton facility, with a classification of 'non contaminating area' with 19 risk points, including the Triton pool, the dry pool Naiade, the hot cell, the liquid waste tanks.

In accordance with the reference zoning, all the waste generated by the cleanup and decommissioning of the Triton pool and the hot cell would be considered as radioactive waste. Nethertheless, processing of measurement data showed that the reference zoning could be reviewed in order to reduce radioactive waste streams. So, a 'specific zoning for dismantling' of each part of the facility which contains one or several risk points has been defined.

As an example, specific zoning of the Triton pool revealed that:

— Water was responsible for contamination transport inside the Triton pool. As a result, the contaminated areas are located:

- On the internal walls of the pool, through the epoxy resin coating and to a depth of 2 cm inside the concrete;
- On the floor of the pool, including tiles, seals and cement coating;

— The radioactivity induced by neutron activation is located:

- Within 1 m radius around the five neutron beam ports set in the radiological shielding of the Triton pool;
- Around the irradiation window, between the Nereïde pile and the Naïade dry pool.

Finally, depending on the specific zonings of dismantling, waste generated by the dismantling work will be:

— 1200 m^3 of conventional waste, mainly concrete and scrap metal. This waste is to be disposed of in repositories, accepted by the CEA centre of Fontenay-aux-Roses.
— 250 m^3 of very low level waste (VLLW), mainly concrete and technological waste. This waste is to be stored at an interim storage facility before being disposed of at the Andra's repository for VLLW.

Low level radioactive waste production is not expected.

The implementation of the specific zonings will result in a considerable reduction (about 60%) of the amount of radioactive waste to be generated.

4. DECOMMISSIONING SCENARIO

The decommissioning scenario has been broken down into phases of conventional work and radioactive work.

In the first phase, all the floors, rooms and footbridges around the Triton pool and the hot cell have to be demolished in order to gain full access around them before demolition (see Fig. 5). This phase will generate conventional waste (concrete and scrap metal).

Workers will not wear any individual protection against ionizing radiation or individual dosimetry. Indeed, dose rates in the Triton facility are at the background level, except for some risk points.

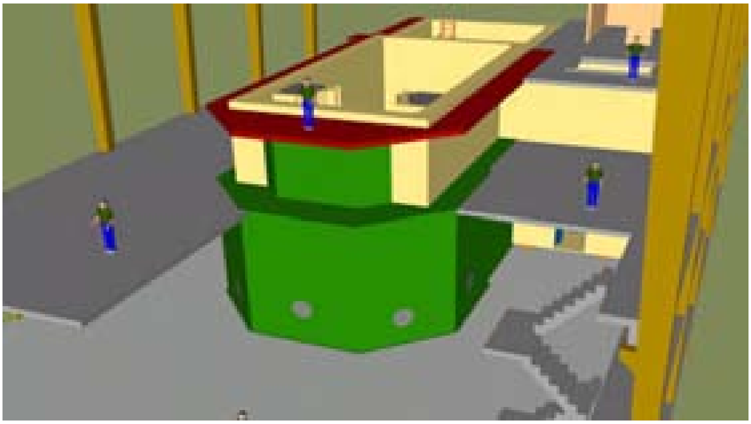


FIG. 5. Floors, rooms and footbridges around the Triton pool and the hot cell.

In the second phase, the hot cell will be cleaned up and partially dismantled. The concrete walls were already cleaned up in 2002 (see next section). Still to be cleaned are the metal inserts to be scraped (with a hand-held grinder) and some concrete areas to be cleaned up (with a chipping hammer). Finally, the metal inserts and lead glass will be removed by cutting the concrete blocks around them (wire-sawing). Waste (concrete rubble, metal dust and concrete blocks) will be conditioned as very low level waste and sent to an interim storage facility.

The next phase will consist of the cleanup of the inner walls of the Triton pool by scarifying and scraping them with hand-held tools. The bottom floor of the pool will be cleaned up by means of a demolition hammer to remove the tile floor and 5 cm of concrete below.

Then, the radiation shielding will be cut into about 120 blocks by the wire-sawing technique, which will be conditioned and sent to an interim storage facility.

The fourth phase will consist of final cleanup of three liquid waste tanks, which were already cleaned up in 2001 (removal of internal polymer coating), but which need to be scraped (hand-held chipping hammer) to remove the residual activity on a 2 cm thickness of concrete.

At the end of the cleanup and removal of all risk points (see Fig. 6), the demolition of remaining concrete structures will then be performed by conventional techniques.

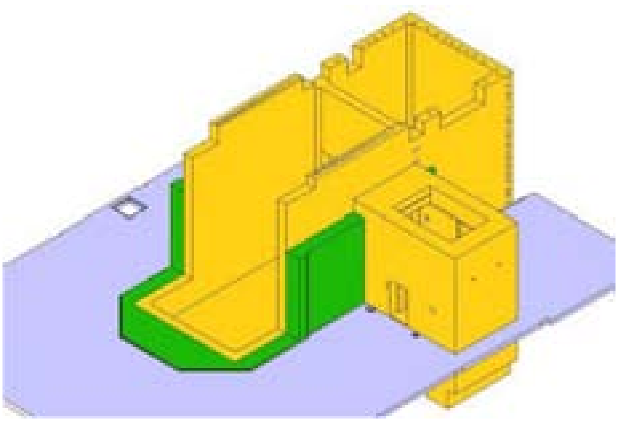


FIG. 6. Remaining civil engineering structures after cleanup and dismantling of all risk points.

Security and safety arrangements

During all phases related to cleanup and demolition of risk points, vacuum and shrouding attachments will be included in the process. During all the phases related to cleanup or demolition of radioactive parts of the facility (risk points), the work areas will be contained and equipped with absolute filtering systems, in order to avoid cross-contamination of the building.

An important requirement the project team imposed on the subcontractor is that under no circumstances will the demolition of a conventional area be carried out at the same time as cleanup of a radioactive risk point. All conventional waste has to have been evacuated from the facility before generation of radioactive waste and vice versa.

Moreover, the facility has to be kept as clean as possible, even during demolition of conventional civil engineering, even when heavy machines and tools are used. To comply with this, the subcontractor has planned many cleaning operations a week.

The project team has decided to avoid, when possible, techniques generating contaminated liquids that would need to be collected and processed. However, the wire-sawing technique needs water as a lubricant. Special precautions are to be taken by the subcontractor, to collect effluents during the sawing operations and to transfer them to a processing unit consisting notably of a decantation unit, filter presses, in state of the art containment conditions.

Special attention has been given to perform a risk analysis related to the civil engineering work to be carried out during stage 3 decommissioning of the Triton facility. The most important risks are:

- Lifting and handling heavy loads (e.g. equipment, waste containers, etc.) by means of a top running crane. This requires a precise circulation scheme in the working area and continuous surveillance;
- Using civil engineering demolition machines (e.g. 12 t mechanical shovel), requiring surveillance around the machine by an operator to prevent the circulation of workers;
- Use of hand-held tools for cleanup operations, requiring the use of individual protective equipment (special gloves, dust or filtration masks; safety glasses, ear-plugs).

The project team has also identified precautions to be taken during the work, concerning fire hazards, electric shock and collapse of the Triton pool structure during cutting and demolition.

It is worth mentioning that, in accordance with French decree no. 94-1159 of 26 December 1994, related to the coordination of security and the health protection of workers during major civil engineering works, a coordination mission has been planned, independent of both CEA and from the subcontractor, in order to have an outside view of the security conditions during the decommissioning work.

A goal fixed for the project team by the facility's owner is that decommissioning operations will be carried out under quality assurance and traceability from one end of the 'processing' line to the other.

5. PRELIMINARY DISMANTLING WORK

In 2002, it was decided to carry out preliminary cleanup and dismantling work, prior to the main dismantling work, planned in 2003. The main reason was these works would confirm the specific zoning for dismantling, being implemented. This involved:

— The demolition of a concrete backfilling poured in the place of the Triton core after the stage 2 decommissioning, probably in order to reduce the dose rate inside the Triton pool, close to the neutron beam ports (no justification report was found);

— The cleanup of the hot cell concrete walls, by scraping and scarifying. It was assumed that removal of wall paint would help to confirm historical data and operational data related to the hot cell, and confirm its radiological characterization (that was the case!).

These works lasted from October 2002 to January 2003.

At the end of the work, radiological expertise was performed on the cleaned surfaces to confirm that the cleanup goals had been reached; Results of the expertise showed that these goals were reached and that the specific zoning for dismantling was confirmed.

One of the most important items of experience feedback gained from these works is that the assessment of technical and engineering competence is of great concern when choosing a subcontractor for decommissioning. Technical audits have to be performed to be sure of these capabilities.

Another important point is that the owner of the facility, responsible for the decommissioning, should have people trained in the fields of radiation protection and nuclear measurements (for waste characterization). Moreover, the owner should have skill in civil engineering in order to remain in control of the operations and, consequently, of the final goals of the decommissioning.

6. CONCLUSIONS

Triton stage 2 decommissioning was completed in 1986 and additional decontamination work was performed in 1993.

The stage 3 decommissioning project started in March 2001. Comprehensive studies were carried out as well as preliminary dismantling work prior to decommissioning.

Considering the low radiological hazards and experience feedback gathered from similar decommissioning projects, due consideration has to be given to the industrial hazards associated with the use of the techniques selected and especially to the lifting and handling of heavy loads.

Calls for tenders have been issued, and a subcontractor chosen. The most important parameters for decision making have been waste minimization, security arrangements, technical efficiency and cost effectiveness.

The stage 3 decommissioning of the Triton facility is to be completed by September 2004, so that the building can be used for new non-nuclear activities.

The dismantling work, currently in progress, is promising in terms of experience feedback, especially concerning the specific zoning methodology and the cleanup and cutting techniques of contaminated concrete.

This experience feedback will be beneficial not only for the future decommissioning projects planned on the CEA promises of Fontenay-aux-Roses, for which the issue of alpha contamination is more crucial, but also on the other sites where facilities have to deal with radioactivity originating from both contamination and activation.

EXPERIENCE WITH PARTIAL DECOMMISSIONING OF PARR-1 (5 MW) FOR CORE CONVERSION AND POWER UPGRADE

M. ISRAR, Q.D. SHAMI, S. PERVEZ
Pakistan Institute of Nuclear Science and Technology,
Islamabad, Pakistan
Email: israr@pinstech.org.pk

Abstract

The Pakistan Research Reactor-1 (PARR-1) was operated at a power level of 5 MW for about 25 years using high enriched uranium (HEU). It was planned to convert the reactor core from HEU to low enriched uranium (LEU) fuel, installation of stainless steel lining on interior of the reactor pool and power upgradation from 5 MW to 10 MW. For this purpose, the reactor was partially decommissioned in 1990. After completing all the requirements, the reactor was re-commissioned with LEU fuel in October 1991.

1. INTRODUCTION

Partial decommissioning of the Pakistan Research Reactor-1 (PARR-1), operating at a power level of 5 MW, was required before taking up improvements in the facilities and installation of equipment for the core conversion and power upgradation. It was a major activity involving the reactor and its support systems. For power upgradation, this activity could be limited to cooling systems and process instrumentation. Lining of the reactor pool with stainless steel was also required for the elimination of seepage of the pool. Therefore, decommissioning activities were extended to make the pool available for unrestricted use and the installation of S. S lining on the interior of the reactor pool.

2. PREPARATION FOR DECOMMISSIONING ACTIVITIES

Preparations made before the decommissioning activities included the design, construction and commissioning of spent fuel storage bay; the fabrication of the fuel transfer cask; assessment of the activity of irradiated components; procurement/fabrication of equipment/tools; and decommissioning plan and procedures [1].

2.1. Spent fuel storage bay

In order to make the reactor pool area accessible, all the reactor components had to be removed and stored in a separate storage area. Therefore, construction of a spent fuel storage bay was started in the vicinity of PARR-2 and it was completed in November 1990.

2.2. Fuel transfer cask

A cask was fabricated for shifting the irradiated spent fuel elements, from the reactor pool to storage bay at a distance of about 265 m. It was designed on the basis that the dose rate outside the cask will not exceed the permissible level if four irradiated fuel elements with the maximum activity were loaded.

2.3. Assessment of activity in reactor pool

Before dismantling the reactor systems, some of the reactor components were surveyed for radiation/contamination. The dose rate from active components was estimated to make arrangements for handling of these components.

2.4. Analysis of pool water

Samples from pool water were analysed. The activity was measured and found to be 1.5 Bq/ml. It was due to ^{110m}Ag.

2.5. Analysis of pool ceramic tiles

Samples of ceramic tiles mounted on the pool walls and floor were analysed for contamination and radiation. Tiles which were not exposed directly were analysed and the radioactivity level was found to be less than 37 Bq/kg due to ^{110m}Ag. ^{214}Bi and ^{214}Pb were found to be slightly above the background level. However, the dose rate on the tiles of the stall end floor near the core outlet was up to about 1 mSv/h.

2.6. Dose assessment of reactor components

The dose rate was assessed on beam tubes, pneumatic rabbit tubes, extension of the thermal column, thermal shield, grid plate and graphite reflector elements ranging from 6 to 400 mSv/h.

3. DECOMMISSIONING

The decommissioning was started in November 1990 by removal of fuel and control rods from the core and shifting to an irradiated fuel storage bay, already constructed for this purpose. The dismantling consisted of the following steps [2]:

— Removal of experiments on beam tubes, dismantling of the reactor core and shifting of fuel elements to fuel storage bay, dismantling of control rods drive mechanism, removal of fission and ion chambers.
— Drainage of reactor pool water to waste storage area.
— Removal of through tube, beam tubes, thermal shields, field instrumentation and dismantling of partial primary and secondary cooling system piping.

The grid plate was removed along with the suspension tower. This was the most active component. The complete structure was unbolted from the pool bridge and was taken out of the pool with the help of a polar crane and stored in a special shielding designed and constructed on the beam port floor.

4. RADIATION PROTECTION

During the decommissioning, adequate measures were taken to protect the personnel working around the reactor, general public and environment against radiation exposure. All the workers were provided with TLDs, dosimeters and protective clothing, as required. The ALARA principle was followed, as far as possible. The personnel underwent compulsory monitoring for contamination and radiation dose. During the period of about 11 months from decommissioning to restartup of PARR-1, the maximum external dose received by a worker was 4.7 mSv (470 mR), which was much below the maximum permissible annual limit.

5. CONTAMINATION CONTROL AND WASTE DISPOSAL

During the decommissioning activities, measures were taken to control contamination. Liquid and solid wastes generated during these activities were treated and disposed of.

5.1. Liquid waste

A major source of liquid waste was primary water in the pool and hold-up tank. After a delay time of about three months, a sample from the pool water was analysed, which gave an activity of less than 1.5 Bq/mL, only due to ^{110m}Ag. This activity was in the low range; as such, the primary coolant was sent in two steps to the seepage pit after dilution. Total estimated quantity of liquid waste was 600 m^3.

5.2. Solid waste

Solid waste generated from PARR-1, was mainly due to the ceramic titles removed for stainless steel lining. The tiles were thoroughly flushed before removal. The debris was collected in steel drums. Direct dose from the tiles was negligible, however, tiles on the stall pool floor had a high dose because of the direct irradiation of these tiles, which were therefore isolated from other tiles. Analysis of these tiles showed the presence of these isotopes: ^{152}Eu, ^{154}Eu, ^{60}Zn, ^{47}Ca, ^{46}Sc, ^{108}Ag and ^{110}Ag. Total activity was about 2200 Bq/kg while other tiles, which were not exposed directly had only ^{110m}Ag with activity of the order of 37 Bq/kg. The waste was packed in about 66 drums having a total volume of about 14 m^3 and the dose rate varied from 10 to 360 μSv/h. The drums were disposed of according to the standard waste disposal procedures. Another source of solid waste was aluminium piping discarded from the primary cooling system. These were washed for the removal of loose contamination. The pipes were not active but had only negligible fixed contamination. These pipes were disposed of as solid waste.

REFERENCES

[1] ISRAR, M., SHAMI, Q.D., PERVEZ, S., Partial Decommissioning of PARR-1 (5 MW) for Core Conversion and Up-gradation, NED-302 (1993).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety in Decommissioning of Research Reactors, Safety Series No. 74, IAEA, Vienna (1986).

DECOMMISSIONING RUSSIAN RESEARCH FACILITIES

S.I. MOROZOV
Federal Nuclear and Radiation Safety Authority of Russia
(Gosatomnadzor of Russia),
Moscow, Russian Federation
Email: msi@gan.ru

Abstract

Gosatomnadzor of Russia is conducting the safety regulation and inspection activity related to nuclear and radiation safety of nuclear research facilities (RR), including research reactors, critical assemblies and sub-critical assemblies. Most of the Russian RR were built and put in operation more than 30 years ago. The problems of ageing equipment and strengthening of safety requirements in time, the lack of further experimental programmes and financial resources, have created a condition when some of the RR were forced to take decisions on their decommissioning. The result of these problems was reflected in reducing the number of RR from 113 in 1998 to 81 in the current year. At present, seven RR are already under decommissioning or pending it. Last year, the Ministry of Atomic Energy took the decision to finally shut down two remaining actual research reactors in the Physics and Power Engineering Institute in Obninsk: AM-1, the first reactor in the world built for peaceful purposes, graphite-type reactor, and the fast liquid metal reactor BR-10, and to start their preparation for decommissioning. It is not enough just to declare the decommissioning of a RR: it is also vital to find financial resources for that purpose. For this reason, due to lack of financing, the MR reactor at the Kurchatov Institute has been pending decommissioning since 1992 and still is. The other example of long-lasting decommissioning is TVR, a heavy water reactor at the Institute of Theoretical Physics in Moscow (ITEF). The reason is also poor financing. Another example discussed in the paper concerns on-site disposal of a RR located above the Arctic Pole Circle, owned by the Norilsk Mining Company. Furthermore, the experience of the plutonium reactor decommissioning at the Joint Institute of Nuclear Research is also discussed. As shown, the Russian Federation has had good experiences in the decommissioning of nuclear research facilities.

1. INTRODUCTION

When the majority of our research facilities were built and put in operation more than 30 years ago, there had been neither requirements nor regulations concerning their future decommissioning. Due to that fact, nobody thought of that in the initial designs of these facilities.

The situation had been drastically changed when in 1994 a top-level safety standard for RR was issued by Gosatomnadzor of Russia (Safety Provision for Safety of Research Reactors [1]) with a special chapter 7, devoted to decommissioning issues. Unfortunately, it was just one page explaining what to do in general terms and was not explicit enough.

Only in 2001, Gosatomnadzor of Russia has developed and issued two comprehensive standards: General Provisions for Safety of Nuclear Research Facilities [2], and Rules for Safe Decommissioning of Nuclear Research Facilities [3].

2. PREPARATORY PROCEDURES

The above standard on decommissioning explains in more or less detail the procedures to be accomplished before the actual decommissioning process can start. Basically, these are:

(1) Remove fuel from the reactor core to an interim spent fuel storage facility at the reactor;
(2) After a specified time of cooling, remove the fuel from the reactor building;
(3) Drain coolant by using the operational manual and remove the former from the reactor building;
(4) Conduct engineering and radiation monitoring of the facility to get initial data on radiation conditions;
(5) Develop a principal decontamination and decommissioning (D&D) programme;
(6) Develop D&D documentation;
(7) Develop a D&D safety analysis report (SAR).

All the above procedures can be carried out under a facility operational licence and when accomplished an operating organization may apply for a D&D licence to be issued by Gosatomnadzor of Russia.

3. CONTENT OF RULES FOR SAFETY DECOMMISSIONING OF NUCLEAR RESEARCH FACILITIES

The following is a list of main issues to be taken into account in a D&D project and D&D SAR:

(1) Designation and application;
(2) General safety objectives;
(3) Principles, criteria and requirements to ensure safety during decontamination and decommissioning of nuclear research facility;
(4) Safety analysis and measures to be implemented to ensure safety;
(5) Preparation for D&D;
(6) Work programme for implementing D&D;
(7) Radiation protection;
(8) Fire protection;
(9) Industrial safety;
(10) Emergency preparedness.

4. TAKING OUT OF REGULATORY CONTROL

When a D&D project is fully accomplished by an operating organization (or other), it shall be evidenced by a special commission, including a GAN representative. A final report on D&D with the radiation data achieved in the end of its implementation shall be written and submitted to Gosatomnadzor of Russia for review and decision making on taking a research facility out of regulatory control.

5. RESEARCH REACTORS UNDER DECOMMISSIONING OR PENDING IT

From the total number of 82 nuclear research facilities, including 34 research reactors, 33 critical assemblies and 14 sub-critical assemblies [4], we (except critical and sub-critical assemblies) have now five actual research reactors under decommissioning stage and one pending decommissioning.

At the end of 2002, the decision by the Ministry of Atomic Energy was made to permanently shut down two RR and to start preparation work for their decommissioning under the operational licence in the regime of final shutdown:

TABLE 1. NUCLEAR RESEARCH REACTORS UNDER DECOMMISSIONING, PENDING IT, OR DECOMMISSIONED

No.	Name	Owner	Location	Power (MWt)	Built	Reactor type/ fuel	Status
1	RG-1M	NGMK	Norilsk	0.1	1970	Pool,EK-10 enrich. 10%	Decom. in 2002 on-site disposal
2	TVR	ITEF	Moscow	2.5	1949	Tank heavy water	Under decom. since 1988
3	IBR-30	JINR	Dubna	0.025 impulse	1969	LM, Pu	Under decom. since 2001
4	Arbus (ACT-1)	NIIAR	Dimitrovgrad	12	1963	Tank organic	Under decom.
5	TIBR-1M	NIIP	Lutkarino Moscow region	6 MJ impulse	1974	U-Mo alloy 90%	Under decom.
6	MR	KI	Moscow	50	1964	Tank, 90% UO2	Under extended shutdown since 1992. Pending decom.

(1) AM (the first NPP in the world for peaceful purposes), channels in graphite, similar to RBMK type, 15 MW(t);
(2) BR-10, LMFR type, 10 MW(t).

All reactors mentioned above with historical and operational parameters and current status are shown in Table 1 and Table 2. It is necessary to point out

TABLE 2. RR OF PHYSICS AND POWER ENGINEERING INSTITUTE UNDER DECOMMISSIONING PREPARATORY WORK

No.	Name	Power (MWt)	Built	Reactor type /fuel	Status
1	AM-1	15	1954	Graphite/channel	Operational at the final shutdown
2	BR-10	10	1959	Fast liquid metal	Operational at the final shutdown

the variety of the reactor types in terms of construction, fuel, coolant, moderator, etc. All this creates a unique situation when a general decommissioning approach for safety must be elaborated on a reactor-by-reactor basis in the concrete D&D project for each RR and reflected in a specific SAR.

6. DIFFICULTIES ENCOUNTERED

6.1. TVR reactor

Handling of heavy water with tritium content and highly irradiated materials in the core and surroundings has created a problem, especially when it is located in the heart of Moscow. Careful, step by step procedures have been elaborated with safety precautions not to spread contamination. Heavy water was taken into a special hermetic volume and sent to a reprocessing plant. What is left to finish decommissioning is to get rid of highly irradiated materials embedded in the structural designs. The final goal is to clean up the premises, including the reactor hall, to construct a new facility—neutron generator.

6.2. PG-1M reactor

The PG-1M reactor is located above the Arctic Circle; the problem is the transfer of irradiated parts of the reactor during very short navigation period. The decision was made to bury them on the site, thus creating a shallow-land radioactive waste storage facility. The work has been done successfully. A new radioactive waste repository is pending an operational licence.

6.3. MR reactor

The MR reactor is a very complex reactor with many neutron beams, experimental channels and irradiation loops. During its operation, there were many radioactive spills, thus creating highly contaminated areas in congested spaces. This would require very expensive work and careful radiation work planning. A major problem is the lack of financial resources.

6.4. AM-1 reactor

There is no experience yet in the Russian Federation of the decommissioning of graphite-type reactors. The main problem is how to utilize the highly irradiated graphite of a large volume. The institute is planning to develop a new decommissioning technology to resolve this problem.

6.5. BR-10 reactor

This will be the first experience in the Russian Federation on the decommissioning of fast liquid metal reactors. The basic problem they are trying now to resolve is sodium coolant poisoned with other heavy metals (e.g. lead, bismuth), used at the facility in the past.

6.6. BR-30 reactor

The IBR-30 reactor was a unique, first of a kind, impulse reactor (with a rotating weal and embedded piece of Pu). The problem is that the fuel (Pu rods) cannot be removed from the reactor core prior to the start of the D&D project. An exclusive decision was made and approved by Gosatomnadzor of Russia to start the reactor dismantling with fuel in the reactor core (subcriticality is 9%).

7. CONCLUSION

As shown in the information discussed, the Russian operating organizations and Gosatomnadzor of Russia have a good practice, experience and safety assurance when dealing with issues of research reactor decommissioning.

REFERENCES

[1] GOSATOMNADZOR OF RUSSIA, Safety Provision for Safety of Research Reactors, Gosatomnadzor, Moscow (1994).
[2] GOSATOMNADZOR OF RUSSIA, General Provisions for Safety of Nuclear Research Facilities, Gosatomnadzor, Moscow (2001).
[3] GOSATOMNADZOR OF RUSSIA, Rules for Safety Decommissioning of Nuclear Research Facilities, Gosatomnadzor, Moscow (2001).
[4] GOSATOMNADZOR OF RUSSIA, Nuclear Research Facilities Database, Gosatomnadzor, Moscow (2003).

SAFARI-1: PLANNING FOR MTR DECOMMISSIONING ACCORDING TO IAEA GUIDELINES

C.S.B. PIANI, L.J. SHAYI
SAFARI-1 Research Reactor,
South African Nuclear Energy Corporation (Necsa),
Pretoria, South Africa
Email: csbpiani@necsa.co.za

Abstract

The SAFARI-1 research reactor has been operational since April 1965 under the auspices of the South African Nuclear Energy Corporation (Necsa), according to a licence granted by the National Nuclear Regulator (NNR) of South Africa. In terms of this licence, it is required that all nuclear facilities demonstrate their ability to ensure satisfactory decommissioning at the end of life. In the case of SAFARI-1, due to relatively low levels of utilization initially, the anticipated end of operational life is expected to be post-2020. The Decommissioning Plan (DP) has nevertheless been completed and submitted for regular evaluation. Although the initial prerequisites of the plan, as per regulatory licence, asked for evaluation in terms of feasibility to return the final decommissioned nuclear facility to a condition of safe shutdown, the DP for SAFARI-1 was approached according to the options specified in selective IAEA guidelines.

1. INTRODUCTION

The South African Fundamental Atomic Research Installation (SAFARI-1) research reactor is a 20 MW(t) MTR based on the ORR, i.e. a tank-in-pool type reactor. It is owned and operated by the South African Nuclear Energy Corporation (Necsa) on behalf of the Department of Minerals and Energy and is located at Pelindaba, about 40 km southwest of Pretoria.

The major utilization of the reactor, which first went critical on 18 March 1965, is for the production of radioisotopes for medical application (national and export) as well as for the production of Neutron Transmutation Doped (NTD) silicon in the pool-side facility. There are also pneumatic and fast pneumatic systems utilized for Neutron Activation Analysis (NAA). The utilization of beam-ports for institutional (academic) purposes is encouraged and neutron diffraction and neutron radiography facilities are operational.

SAFARI-1 has, since its initial operation, applied a management system which was primarily focused on the technical design and safe operation of the plant. Currently, the plant is operated under an integrated management system, incorporating quality, health, safety and environment (QHSE). The reactor is fully certified according to ISO 9001 (2000), with ISO 14001 (1996) certification currently being finalized [1].

2. LEVEL OF UTILIZATION AND ANTICIPATED END OF LIFE

The historic utilization of SAFARI-1 is reflected in Fig. 1 in terms of operational MWh/quarter and accumulated MWh experienced from 1965 to date. It is immediately evident from the integrated level of utilization and comparison with equivalent fluence levels that the low levels of exposure, particularly during the first few years of operation, as well as during the period 1977 to 1993, will have the benefit of an extended lifetime for the reactor.

Recent evaluations [2] have indicated that an expected operational lifetime beyond 2020 would be quite realistic for the reactor. The current licence of SAFARI-1, as authorized by the National Nuclear Regulator (NNR) endorses such an approach, but requires, however, that a full decommissioning plan (DP) be prepared for SAFARI-1.

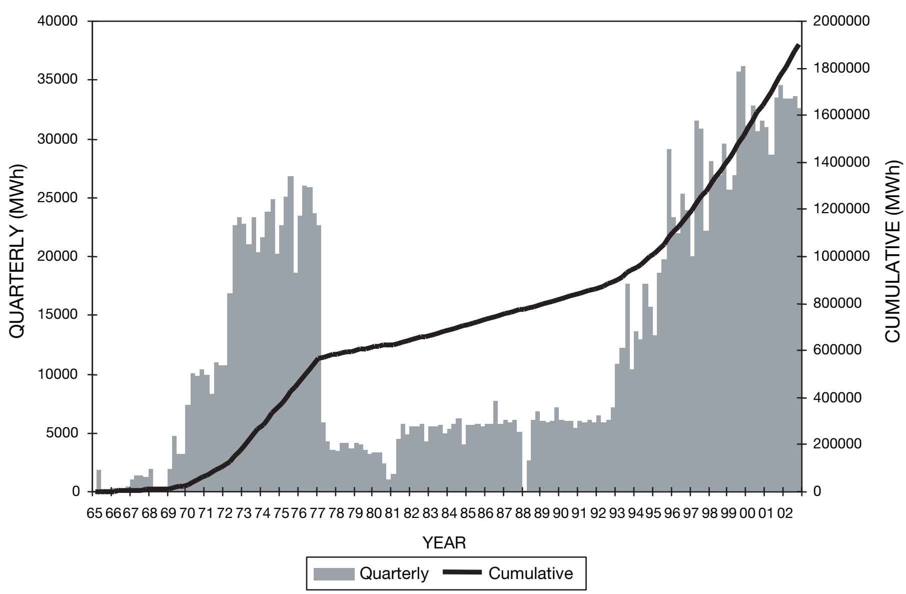


FIG. 1. SAFARI-1 power history.

3. SELECTION OF DECOMMISSIONING STRATEGY

Using the IAEA guidelines [3, 4] as reference, an initial evaluation was made as to the various alternatives available internationally for consideration in a DP:

3.1. Immediate dismantling

This is an alternative whereby the complete reactor assembly (including its ancillary equipment) and the portion of the facility and site containing radioactive materials are removed or decontaminated to a level that permits the property to be released for unrestricted use. This approach normally occurs shortly after the reactor shutdown.

Advantages:

- Some of SAFARI-1 operations personnel, with intimate knowledge of the plant, would still be available;
- Waste costs would be known;
- Immediate reuse of the property is possible;
- No maintenance or surveillance would be required after dismantling;
- There would be no liability for the reactor after decommissioning.

Disadvantages:

- Risk of high radiation doses to workers during dismantling is increased;
- Dismantling and waste management costs may be higher due to the activity levels;
- Funds must be provided immediately;
- Low and intermediate level wastes must be stored on-site in temporary storage until such time as a South African disposal site is identified and commissioned.

3.2. Entombment

The entombment of a reactor facility is the method by which radioactive material and the reactor structure are encased in a structurally long-lived material, such as concrete. The entombment structure is maintained and continued surveillance is carried out until the radioactivity decays to a level that permits site release.

This method is limited to facilities where the appropriate decay of the nuclides in the nuclide vector is shorter than the expected life structure. Currently, however, this method is not recommended by the IAEA.

Advantages:

- Waste disposal cost is low;
- Waste management (short term) is simplified;
- Overall decommissioning cost is low;
- Radiation hazards to operators during decommissioning are low;
- Special equipment or technology is not required.

Disadvantages:

- It is not recommended by the IAEA: it postpones the problem to a next generation;
- Long term surveillance is required;
- Long term liability and maintenance costs may cause a serious problem;
- The reactor site and SAFARI-1 buildings cannot be reused or demolished;
- Public relations may be a problem;
- More volume of waste (and cost) will be generated if a decision (maybe political) were made to dismantle the entombed SAFARI-1 before the waste has fully cooled down.

This alternative is not feasible for SAFARI-1 decommissioning because of the disadvantages outlined above and risks involved during its entombed period of potentially several centuries. Therefore, it was not considered as an alternative.

3.3. Long term storage

Long term storage is one of the most popular methods for large reactors in which the nuclear facility is placed and maintained in a condition that allows it to be safely stored for an appropriate decay period before it is substantially decontaminated or dismantled to levels that permit release for unrestricted use.

If the facility or the reactor site is to be reused for future nuclear activities, restricted release may be a better option than unrestricted release. Certain areas may be decontaminated to unrestricted levels while others decontaminated to a specific release level. Alternatively, other conditions may be placed on the facility to limit worker exposure to the radioactive material.

Advantages:

- — Newer technology for dismantling will probably be developed;
- — Activity level of SAFARI-1 components will be reduced after sufficient decay period;
- — Radiation exposure to operators during dismantling can be lowered;
- — Demand for remote handling equipment will be much lower;
- — A national radioactive waste repository may become available by the end of the long term storage period;
- — Waste from dismantling can be handled safely without special equipment due to the reduced activity levels.

Disadvantages:

- — Personnel with first-hand knowledge of and experience with SAFARI-1 will be not available;
- — Maintenance and surveillance must be carried out on a regular basis to prevent deterioration;
- — Costs for care and maintenance will accumulate to a large sum;
- — Waste management and labour costs may increase;
- — Regulations may change;
- — The reactor site cannot be reused until dismantled;
- — It may hamper public relations, in particular, those associated with a replacement reactor.

Analysis of the various alternatives, as outlined, based on total costs, personnel dosage including continued surveillance and escalation and benefits in the extent of removal of radioactivity, showed that the alternative discussed in Section 3.1 (Immediate dismantling) would be the best option for SAFARI-1, as evaluated under current technology awareness and resource availability.

3.4. Decommissioning strategy

Necsa plans, at the appropriate time, to decommission SAFARI-1 (located in Building P 1800 in the western section of the corporate site) and to release the facility for clearance restricted use as defined by the NNR licensing document requirements (see Section 9.2).

The Corporation plans to remove and store some of the radioactive waste within an established area within the site, while the rest would be shipped off-site.

The Corporation's current intent is to cease the operation of SAFARI-1 reactor by 2020 should refurbishment not be feasible. No reuse of the reactor and its biological shielding is contemplated following shutdown. After a suitable cool-down period, fuel and neutron sources will be removed from the reactor. Spent fuel would be transported to site storage within the Corporation's site. After allowing a suitable decay period, all the other components will be dismantled and disposed of in accordance with ALARA principles.

The analytical results of the estimates of low level waste quantities and worker exposure given in this decommission plan will be confirmed by field measurement. In general, decontamination will proceed from the areas exhibiting maximum radiation levels to those areas with minimum levels.

4. MAJOR DECOMMISSIONING TASKS

After the removal of fuel elements from the core, nine major tasks, as given in Table 1, are planned for the decommission programme. As stated earlier, these tasks would commence after a predetermined cool-down period and after completion of the decommissioning licensing process.

5. RADIOLOGICAL STATUS (CURRENT AND ENVISAGED)

The current and envisaged radiological status of the facility has been estimated by a study of much of the bulk material in the vicinity of the reactor core that was irradiated at high neutron fluxes. The components included in this study are:

— Grid plate sections;
— Core boxes;
— Hold-down arms;
— Bearing plates.

Since the components are composed of various materials and are exposed to various fluences during their lifetime, estimates were made of the expected residual activity based on the historic utilization of the reactor and on a postulated operational requirement in future. The 3D code OSCAR-3 for fuel modelling, together with the 3D Monte Carlo transport computer code MCNP-4B and ORIGEN-S were used to substantiate the calculations for the above major components as well as for some smaller components in high flux positions. Typically, these include the thimbles, nozzles, hydraulic rabbit piping,

TABLE 1. IDENTIFIED TASKS FOR SAFARI-1 DECOMMISSIONING

Task:	Description
1	Removal of spent fuel and sources
2	Trained team move-in (mostly former SAFARI-1 employees)
3	Initial radiation survey
4	Installation of confinement barriers
5	Removal of reactor components and pool liner
6	Removal of material with potential surface contamination and other activated materials
7	Cleanup and removal of tools and equipment
8	Packaging and shipment of radioactive waste
9	Termination of radiation survey
10	Demolition of non-radioactive portion of reactor installation

etc., as well as the reflector elements. Postulated values (current in 2000, as well as at the time of decommissioning in 2020) are given in Table 2. Activities of SAFARI-1 spent fuel elements were dealt with separately.

The resultant dose rates at the surface of each of the components, as well as 1 m away in air were calculated with the QAD-CGGP computer code (part of the SCALE 4.4 System).

6. DECOMMISSIONING ORGANIZATION AND RESPONSIBILITIES

The manager/executive engineer of SAFARI-1 would be responsible for the decommissioning of the reactor. Reorganization and retraining of former employees of the reactor could be implemented to oversee the decommissioning programme. Outside assistance in the form of consultants could be considered for areas where specific expertise is required.

6.1. Decommissioning programme committee

A decommissioning programme committee consisting of six members, including representatives from Risk Management, Nuclear Liability Management, Quality Safeguards and Engineering, Project Manager

TABLE 2. ANTICIPATED ACTIVITY OF ACTIVATED MATERIAL

Component	Activity as at January 2000 (Ci)		Activity as at December 2020 (Ci)	
	1 month decay	3 months decay	1 month decay	3 months decay
Grid plate (Section 1)	476.3	263.4	652.2	408.6
Grid plate (Section 2)	173.1	93.3	257.4	149.6
Core box (east/west/south sides)	1977	1116	2593	1713
Core box (north—not adjacent to fuel)	403.6	225.3	540.6	347.5
Core box (north—adjacent to fuel)	646.2	370.4	824.5	564.0
Hold down arm (at row 5)	6424	4038	1045	7187
Hold down arm (at row 7)	4223	2646	6945	4729
Bearing plate	668.3	417.2	1114	751.5

Operational, Operational Support Manager and a Project Engineer would be established. The committee's responsibilities would include:

- Advise the manager/executive engineer on all matters pertaining to the decommissioning activities;
- Provide direct oversight and planning;
- Review and approve all decommissioning planning and operation documentation.

6.2. Operational support groups

These groups would consist of the following subgroups: radiation protection officers (RPOs), safety officers and quality assurance (QA) personnel to ensure that support is forthcoming for:

- Review of industrial safety, fire and industrial hygiene programs;
- Review of decommissioning operations with respect to their impact on the personnel and the environment;
- Review of radiation monitoring results to ensure that decommissioning activities meet the requirements of the NNR licence;
- Establishing and implementing the QA plan for decommissioning of SAFARI-1.

The corporate QA supervisor will prepare and carry out a comprehensive system of planned and periodic audits/surveillance using documented checklists to verify compliance with all aspects of the QA programme and to determine its effectiveness.

7. TRAINING AND QUALIFICATIONS

7.1. Specific training

A special training programme designed to provide orientation to personnel and meet the requirements of the applicable in-house regulatory and licence documents (NL27) will be implemented. General site training according to established QMS standards will be required for all personnel assigned to the decommissioning project.

Training topics will depend upon health and environmental impacts of planned operations as well as upon applicable regulations, standards and guidelines pertinent to operations involving radiological or chemically hazardous materials or waste.

The anticipated training programmes will follow actions to include:

— Radiation worker: personnel involved with radioactive materials;
— Respiratory protection: to meet project requirements in compliance with a respiratory protection programme;
— Hazard communications: conducted as applicable;
— Technical training: job activity simulations or briefings to ensure issues and ALARA considerations.

7.2. Administration and record keeping

The RPOs shall be responsible for the training programme and maintenance of personnel training, qualification and exposure records. If the Necsa corporate functions that carry out these responsibilities are still in place, they will execute the responsibilities.

Complete up-to-date training, qualification and exposure records will be maintained on all personnel. The records will include the following:

— Bioassay analysis;
— Personnel exposure records;
— Individual dosimeter readings as related to daily tasks and work procedures;

— Respiratory protection qualifications (medical clearance and fitness test);
— Audiogram results;
— Training records;
— Visitor logs and exposure information.

8. OCCUPATIONAL RADIATION PROTECTION PROGRAMME

The Occupational Radiation Protection Programme (ORPP) of a SAFARI-1 research reactor decommissioning project consists of a set of policies, procedures and instructions enacted to protect workers, the general public and the environment.

Objectives of the ORPPs include:

— Ensuring the health and safety of personnel by providing protection programmes which include a commitment to the principles of maintaining exposures as low as reasonably achievable (ALARA);
— Minimizing the exposure of the general public and the environment to the radioactive and/or hazardous chemical effluents that may be released during the decommissioning activities;
— Identifying and separating contaminated structures, surfaces, systems and components from those that are not contaminated;
— Disposing of contaminated and non-contaminated components and materials properly and safely;
— Ensuring that the facility meets all radiological decommissioning requirements and is ready to be released for unrestricted use;
— Decontamination of those components and structures to be recycled or fit for other applications.

The ORPPs provide integrated occupational health, health physics, industrial hygiene and safety elements. In order to meet IAEA Safety Guide recommendations (Safety Series No. RS-G-1.2), the following elements were examined: radiation protection programme, industrial safety and hygiene programme.

In particular, the radiation protection programme for the decommissioning estimates requirements to monitor radiation and radioactive materials, to control distribution and releases of radioactive materials and to keep collective radiation exposure for individuals within limits at ALARA levels. The programmes shall meet with the requirements of the RM-LIS-8xxx series of documents listed in Section 9.2.

9. LICENCE DOCUMENTS AND CORPORATE STANDARDS

This section identifies typical licence and corporate documents and standards applicable to the decommissioning of the SAFARI-1 reactor and applies directly to the NNR and Necsa requirements in terms of a nuclear facility.

9.1. NNR licence documents

LD-1001:Notification Requirements of Occurrences Associated with Nuclear Installation or activities on the Pelindaba site of Necsa.

LD-1049:Radiological Surveillance Program for Workplaces.

LD-1079:Controlled Program for Instrumentation used in Radiological Surveillance.

LD-1091:Requirements on Licences of Nuclear Installations Regarding Risk Assessment and Compliance with the Safety Criteria of the NNR.

9.2. Necsa licence documents

The following list includes Necsa licence documents:

- NECSA-PS01:General Security Measures Applicable at the NNR.
- RM-LIS-8002:Radiation Protection Organization.
- RM-LIS-8003:System Classification and Demarcation of Controlled Radiological Areas.
- RM-LIS-8004:Removal of Material from Controlled Radiological Areas.
- RM-LIS-8005:Radiological Surveillance Program.
- RM-LIS-8006:Controlled Program for Instrumentation used in the Radiological Surveillance Program.
- RM-LIS-8007:Radiation Dosimetry Program.
- RM-LIS-8008:Radiation Dose Limitation.
- RM-LIS-8012:Radiation Protection Work Permit.
- RM-LIS-8035:Meteorological Program at the Pelindaba site of Necsa.
- RM-LIS-8036:Solid Radioactive Waste Management Program at Pelindaba.

10. HEALTH, SAFETY AND ENVIRONMENT PROGRAMME

The Health, Safety and Environmental Programme (HSEP) for the decommissioning project ensures the protection of all personnel occupying the

SAFARI-1 building from detrimental non-radioactive exposures and hazards. The HSEP will be administered in accordance with corporate regulations, except that in the absence of a particular regulation, guidance will be obtained from ISO 14001. Detailed specifications of the health and safety limits and their implementation for workers and the public were evaluated and included in the DP.

The HSEP deals in particular with the aspects relating to:

— Personnel certification, including medical fitness for the tasks, training;
— Specific training, e.g. fire protection and prevention, special material handling, etc.;
— Exposure limits according to an ALARA programme;
— Surveillance, audit and inspection programmes.

11. OCCUPATIONAL HAZARDS

The monitoring of specific hazards and proposed mitigation of such actions were evaluated and typical examples identified as indicated in Table 2.

The occupational doses indicated in Table 3 are typical of values for research reactors and give an indication of the possible impact on the workforce. These generic estimates will be refined at some stage in future by conducting a hazard assessment for the preferred decommissioning option of SAFARI-1 specifically.

The radiological hazards to be encountered during decommissioning will probably differ significantly from those encountered during normal operation in scope and nature. This will require detail task-by-task dose estimates and ALARA planning during decommissioning activities and may require additional resources in terms of RP personnel and instrumentation.

There will also be increased exposure to other occupational hazards during certain decommissioning, e.g. noise and hazardous chemicals. This will require additional training, proper supervision and personal protective equipment (PPE).

12. COST ESTIMATES

Based on selective information available at the time of preparation of the DP, an overall estimate of $50 million was reached, including the disposal of the last core but excluding the cost of demolition of non-contaminated structures.

TABLE 3. MITIGATION AND MONITORING DURING DECOMMISSIONING

Hazard	Mitigation	Monitoring
Airborne dust or radio nuclides	Water fogger Respirators HEPA filtration units	Whole-body counts Continuous air sampling Grab air sampling
Falling and flying debris	Safety glasses Limited access, safety shoes Hard hats	Incident reports
High sound levels	Ear protectors	Physical exams dB measurements
Beta-gamma exposure rates	Limited access	Personnel dosimetry Daily surveys
High heat and humidity	Ventilation Work breaks	Temperature measurements Stay time limits
Loose surface contamination	Anti-contamination clothing Cleanup/decontamination	Contamination monitoring
Airborne hazardous and chemical vapours	Respirators Cleanup/decontamination	Air sampling

TABLE 4. OCCUPATIONAL EXPOSURE FOR DECOMMISSIONING A RESEARCH REACTOR (MAN-MSV)[a]

Decommissioning element	Immediate dismantling	Safe storage		
		10 years	30 years	100 years
Preparation of safe storage	na[b]	1120	1120	1120
Continuing care	na	Neg[b]	Neg	Neg
Dismantling	3320	860	60	10
Truck shipments during preparation of safe storage	na	120	120	120
Truck shipments during dismantlement	220	20	Neg	Neg
Total	3340	2120	1300	1250

[a] Adapted from the IAEA's Technical Reports Series, see Ref. [4].
[b] na = not applicable; neg = negligible.

This is in agreement with typical estimates provided by selective IAEA literature, for example, see Ref. [4].

13. DISMANTLING AND DECONTAMINATION OF SAFARI-1

Details of these various activities are accommodated in the applicable DP but are too exhaustive to be discussed at this stage.

REFERENCES

[1] DU BRUYN, J.F., PIANI, C.S.B., "ISO 9001 and ISO 14001: An Integrated Quality Management System for an MTR Facility: SAFARI-1 Research Reactor", Research Reactor Utilization, Safety, Decommissioning, Fuel and Waste Management (Proc. Int. Conf. Santiago, 2003), IAEA, Vienna (2005).

[2] SOUTH AFRICAN NUCLEAR ENERGY CORPORATION (NECSA), The SAFARI-1 Research Reactor Safety Analysis Report, RR-SAR-01 to 21 (reflecting Chapter 01 to 21 of licensing submissions to the national nuclear regulator), Necsa, Pretoria.

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Planning and Management for the Decommissioning of Research Reactors and Other Small Nuclear Facilities, Technical Reports Series No. 351, IAEA, Vienna (1993).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning Techniques for Research Reactors, Technical Reports Series No. 373, IAEA, Vienna (1994).

INTERNATIONAL DECOMMISSIONING STRATEGIES

L. JOVA SED, D. REISENWEAVER
Division of Radiation and Waste Safety,
International Atomic Energy Agency,
Vienna
Email: L.Jova-Sed@iaea.org

Abstract

The IAEA has been developing guidance and technical information relating to the decommissioning and decommissioning strategies of nuclear facilities for over 20 years. During this time, the international concept of decommissioning strategies, and its importance, has changed. Three basic decommissioning strategies are envisaged as possibilities for nuclear installations: immediate dismantling, deferred dismantling and entombment. All have advantages and disadvantages, but the International Conference on Safe Decommissioning for Nuclear Activities demonstrated that immediate dismantling is the generally preferred option. However, there are a number of factors that might lead operators to choose one of the other strategies, and each situation has to be examined individually to identify the optimal strategy for that situation. The basic approach of these three strategies is discussed in the paper.

1. INTRODUCTION

Hundreds of nuclear installations, including research reactors, will end their operational lifetimes during the next 50 years. While there is considerable regulatory experience at the “front end” of the regulatory system for the design, construction, commissioning, and operation of the nuclear installation; the experience at the back end is, at present, limited, since comparatively few installations have actually been decommissioned.

Nevertheless, a recognisable international strategy for decommissioning is emerging. It can be seen in the relevant Safety Standards of the IAEA, which have been developed and approved by committees of national regulators and most recently in the findings of the International Conference on Safe Decommissioning for Nuclear Activities, held in Berlin, Germany in October 2002 [1]. Dismantling of nuclear facilities is basically not a technical problem but a challenge to project management and logistics once the legal and economical boundary conditions have been clarified [2]. This paper presents the main elements of the emerging international strategies.

2. PLANNING

Many regulatory authorities and facility operators believe that decommissioning begins when they permanently stop operating the plant. At this point many of them do not know what to do and do not have any funds to perform the necessary safety activities. Even when this situation confronts them for one facility, they only consider the current facility and do not start the planning for other facilities that use nuclear material in their country or under their control. The decommissioning planning activities, which begin at design or as soon as possible when it is recognized that resources will be needed, is critical for ensuring safety during the entire life of the facility. Decommissioning has always been the orphan in the nuclear process and only dealt with when it would not go away.

A successful and safe decommissioning project requires sound planning and a framework for the continuous regulatory oversight of the project. Ideally, planning for decommissioning should start early, preferably when the facility is being designed, as required in the IAEA Standards [3, 4] and the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management [5]. The planning should address establishment of mechanisms for the funding of decommissioning and should anticipate that nuclear facilities may cease operations prematurely for technical, economic or political reasons. However, in practice that has not always been the case. It has been noted that many facilities have less than adequate decommissioning plans or funding mechanisms. For existing research reactors, the aim should be to start the planning for decommissioning well before the end of operation. For new research reactors, a preliminary decommissioning plan should be prepared and approved prior to the issuance of the construction permit.

It is important to obtain a thorough knowledge of the condition of the facility at the end of operations, including knowledge of all the waste streams to be expected during the subsequent decommissioning. The decommissioning plan for the facility should include a description of the intended management approach for each of these waste streams. This in turn requires that the State should have national plans in place for the safe management of these wastes.

3. STRATEGIES

A specific decommissioning strategy will, among other things, define the timing and the sequencing of decommissioning activities. Decommissioning strategy can range from immediate dismantling and removal of all radioactive materials from the site, allowing unrestricted release, to an option of in situ disposal involving encapsulation of the reactor and subsequent restriction of access.

An evaluation of the various decommissioning options should be performed by considering a wide range of issues, with special emphasis on the balance between the safety requirements and the resources available at the time of implementing decommissioning. Cost benefit or multi-attribute type analyses provide systematic means for such an evaluation. These analyses should utilize realistic estimates of both costs and radiation doses. It should be ensured that the selected option meets all the applicable safety requirements. The selection of a preferred decommissioning strategy should be made by analysing components such as [4]:

(a) Compliance with laws, regulations and standards which should be applied during decommissioning;
(b) Characterization of the installation, including the design and operational history as well as the radiological inventory after final shutdown and how this changes with time;
(c) Safety assessment of the radiological and non-radiological hazards;
(d) The physical status of the nuclear installation and its evolution with time, including, if applicable, an assessment of the integrity of buildings, structures and systems for the anticipated duration of the deferred dismantling;
(e) Adequate arrangements for waste management, such as storage and disposal;
(f) Adequacy and availability of financial resources required for the safe implementation of the decommissioning option;
(g) Availability of experienced personnel, especially staff of the former operating organization, and proven techniques, including decontamination, cutting and dismantling, as well as remote operating capabilities;
(h) Lessons learned from previous, similar decommissioning projects;
(i) The environmental and socio-economic impact, including public concerns about the-proposed decommissioning activities; and
(j) The anticipated development and use of the installation and the area adjacent to the site.

This list contains many issues that have greater or lesser significance, depending on the specific circumstances of decommissioning in each country. To assist the development of options, a number of these components are further developed below.

The transition from operation to decommissioning will usually be accompanied by organizational changes, particularly reductions in staff. Such reductions may be inevitable, but the operator must manage the change so as to retain the expertise needed and to guard against a degradation of safety culture due to demotivation of remaining staff. The regulator also needs to be particularly vigilant in relation to the possible effects of such changes.

The safety situation at a facility will typically change much more often during decommissioning than during operation, and the safety case may therefore need to be updated more frequently. Although the general trend will be to the risks to reduce, there may be short term increases risk, particularly to workers, for example during decontamination or dismantling of a normally inaccessible part of a facility. Unexpected conditions may also be encountered, and decommissioning plans and regulatory attitudes need to be flexible to deal with such situations.

Three basic decommissioning strategies are envisaged as possibilities for nuclear installations: immediate dismantling; deferred dismantling; and entombment. All have advantages and disadvantages, but immediate dismantling is the generally preferred option. However, there are a number of factors that might lead operators to choose one of the other strategies, and each situation has to be examined individually to identify the optimal strategy for that situation.

3.1. Immediate dismantling

This term means a decommissioning strategy in which the equipment, structures, and portions of a facility and site containing radioactive contaminants are removed or decontaminated to a level that permits the property to be released for unrestricted use shortly after cessation of operations. This is complete and prompt decommissioning enabling the facility to be released for generally unrestricted access, for most research reactors within 3 years after permanent shutdown. It involves the decontamination, dismantling and removal of all equipment, structures and other parts of the facility that had become radioactively contaminated.

Immediate dismantling typically has the fewest uncertainties, eliminates the risk associated with the facility promptly, will normally cost less than delaying dismantling, and allows the use of operational staff who know the facility and its history (such staff can contribute their expertise and experience during the decommissioning process).

Under this option the principle advantage is that the responsibility for the decommissioning is not transferred to future generations, because the site is available for unrestricted use promptly. But immediate decontamination may lead to generally higher worker doses, because there is little opportunity for the radioactive decay of short-lived radionuclides.

An impediment to immediate dismantling is the absence of a waste disposal route. However, the absence of a repository should not be reason to prevent immediate dismantling. If repositories are not available, regulators should provide guidance to operators on the appropriate arrangements for the safe conditioning and storage of waste.

The magnitude of the decommissioning task facing the nuclear industry is not fully realized. Because of this, many countries are not ready to accomplish decommissioning in a safe manner. This is illustrated with the lack of disposal facilities for nuclear waste in many countries, even though they have decided to begin decontamination and dismantling their facilities. This means that waste must be placed in storage until a permanent solution is found. This is also the case with spent nuclear fuel from nuclear power plants and research reactors. This normally requires additional resources and means that another facility will require eventual decommissioning. This is a process that can feed on itself and grow as time continues.

3.2. Deferred dismantling or safe enclosure

The deferred dismantling decommissioning strategy requires that a limited number of activities be taken immediately after shutdown and the facility is placed and maintained in such condition that the nuclear facility can be safely stored for an extended period of time and subsequently decontaminated and/or dismantled to levels that permit release for unrestricted use. As the name implies, this usually involves placing the facility in a safe, stable, and monitored condition and keeping it in that state until a decision is made to dismantle. Usually removal from regulatory control has to take place within 50 years after shutdown [6].

To allow this storage period to occur, in the operational phase all of the liquids should be drained from the systems, any operational waste that has been collected during the operational period should be removed and areas not normally in need of access during the storage period should be secured. This option does allow for the decay of radionuclides, but this is not normally the primary reason this strategy is chosen.

There are many advantages to this option. Some minor decontamination may occur and allow the boundary or footprint of the controlled area to be significantly reduced, which will save money and other resources over the 50 years period. Portions of the facility or site may be used for other purposes.

Spent fuel may also be an issue. It is preferred that all spent fuel is removed from the site before the long-term storage period begins. This reduces the safeguards and security concerns and allows for a large reduction in the overall risk of the facility. It also reduces the number of systems that must be maintained to ensure safety during the 50 years period.

Deferred dismantling minimizes the initial commitments of time, funds, radiation exposure, and waste disposal capacity, while complying with requirements for protection of the public health. Another advantage is in the case where there are other operational nuclear facilities at the same site or where there is a shortage of radioactive waste disposal capacity.

Deferred dismantling may have benefits for facilities, which contain short-lived radionuclides that represent an important source of risk. It may provide ‘breathing space’ in cases where sufficient funding is not yet available, or may be convenient where there are multiple facilities on the same site. Deferral of dismantling and demolition may reduce the quantities of radioactive waste produced and reduce radiation exposure to site personnel. In addition, this delay in dismantling may permit technological improvements in the future to be incorporated into the process when decommissioning activities are resumed. However, such benefits should be considered in the context of the additional costs associated with providing long term surveillance and maintenance, the problem of ensuring that sufficient expertise and knowledge will be available for dismantling, and the additional uncertainties introduced by delay. This may be particularly difficult if the operator’s or the national nuclear programme has come to an end. For example, financing may be more difficult to guarantee, there may be unforeseen changes in regulatory requirements, or the condition of the facility may deteriorate despite care and maintenance programmes. The containment of radioactive material and integrity of structures may become key safety issues in the context of protecting both the workforce and public. The deferred dismantling concept implies that the operator would need to operate a surveillance and monitoring programme and have contingency plans in place to remediate in the event of unforeseen occurrences or to decommission early, if problems occur.

There may be additional disadvantages in delaying dismantling and demolition. If deferred dismantling is being considered for a prolonged period of time, due regard should be given to gradual deterioration of the structures, systems and components designed to act as barriers between the radionuclide inventory and the environment. This deterioration may also apply to systems that could be necessary during plant dismantling. The safety assessment should consider the requirement for maintenance, tests of requalification or replacement of these systems (mechanical handling systems, ventilation, power supply and waste handling systems) and the implications of deterioration for safety should be evaluated. To implement safe enclosure, new systems and structures may have to be installed or existing systems and structures modified. The integrity of these new systems and structures should be assessed over the prolonged period of safe enclosure (deferred dismantling). Delays in decommissioning may also lead to an increased liability resulting from possible exposures or from releases and migration of residual radionuclides.

This strategy is not normally applicable for facilities that contain very long lived half live radionuclides (i.e. reprocessing plants).

If deferred dismantling is instituted, the safety documentation and decommissioning plans should be periodically reviewed to ensure that they represent current installation conditions.

The absence of an available disposal route has been used as an argument for choosing the safe enclosure strategy; the idea being that dismantling is delayed until a repository becomes available.

3.3. Entombment

This is a decommissioning strategy is similar to immediate dismantling, except that not all of the radioactive material is removed from the site and the radioactive contaminants are encased in a structurally long-lived material, such as concrete. The entombment structure is appropriately maintained, and continued surveillance is carried out until the radioactivity decays to a level permitting unrestricted release of the property. This strategy eliminates the need for total decontamination by proceeding directly from permanent shutdown to the encasing of radioactive contamination in a structurally sound material such as concrete. Then it must be appropriately maintained and monitored until the radioactivity decays to a level permitting release of the property.

As a general principle, entombed facilities becomes a near surface waste repository and must comply with radiological requirements for waste disposal facilities, but more specific international guidance is needed on the long term safety conditions which should govern the entombment strategy. Entombment may be an option for States needing to decommission a single facility, for example, one research reactor, and not having the resources to develop or obtain the infrastructure needed for dismantling and waste disposal.

In the normal lifecycle of a nuclear facility, the facility is planned, designed, constructed, operated and decommissioned. Once a facility begins operation, it must meet the operating safety conditions until such time as it begins to implement the decommissioning strategy that has been chosen.

4. STRATEGY IMPLEMENTATION

The overall decommissioning strategy to be adopted should be identified as early as possible in the planning process. The presentations and discussions at the Berlin Conference [1] indicated a distinct shift in recent years towards immediate dismantling as a preferred strategy. This preference seems to be based on a range of considerations, notably the availability of know-how and experienced staff from the operational phase, and certainty of funding. Nevertheless, there will still be cases in which one of the other strategies – deferred dismantling or entombment may be appropriate.

There are a large number of research reactors that have “shutdown”. This is a very ambiguous term and it is not clearly defined what the current status is of these facilities. It could be that these facilities are shutdown for maintenance, refuelling, and equipment replacement or have permanently ceased operations. This has an impact on operations and future planning activities. There have been cases where a facility has been shutdown for 10 – 20 years, sitting in a limbo state. During this time, the facility is not producing any income, but funding and resources must be made available to maintain the facility in a safe manner. The longer the facility is shutdown, the smaller the funding gets until the point where the facility becomes a safety concern.

It is common to meet the establishment and recognition of a new phase in the use of a facility called: “Extended Shutdown” as a category. In our opinion, this phase is part of the “Operational Phase” and nothing else. There is no need for another phase in the life of a research reactor. In particular, concern has been expressed about research reactors that are shut down for extended periods without plans for either restarting or decommissioning.

A research reactor should either be maintained in an operational state or the implementation of the decommissioning strategy should begin. If there are reasons that the facility cannot be immediately dismantled (i.e., lack of funding, lack of waste disposal or storage route), and then the facility should be placed into a safe storage condition. This means that as a minimum, the fuel should be removed from the reactor and preferably from the site, if possible; all systems drained of liquids and the liquids processed; operational waste processed and either sent for permanent disposal or placed into approved storage; and a long term surveillance and maintenance plan be developed and implemented.

In any project there are associated uncertainties and managing these is an integral part of project management. A strategy that involves intensive care and maintenance far into the future will be subject to more uncertainty than one that does not and this in itself can be a powerful driver for choosing the immediate decommissioning option. For example, cost estimates would need to contain appropriate risk margins to accommodate these uncertainties.

Sharing of practical experience is very valuable. Building on the experience of others should lead to future decommissioning projects being completed more easily at lower cost with less waste produced and lower personnel doses.

5. REMOVAL OF REGULATORY CONTROL

The ultimate aim of decommissioning is to allow the removal of some or all regulatory controls from a site. Release of the site for uncontrolled use (i.e. any use) is the generally preferred option, but this may not always be practicable, and controls on the future use of some sites or parts of sites may need to be maintained. The recycling or re-use of material from decommissioning can greatly reduce the amount of waste that needs to be disposed of in a repository. This can preserve resources and repository capacity.

There has been extensive international discussion on the radiological criteria appropriate for the release of material. There is a reasonable degree of consensus on the use of an individual dose criterion of around 10 Sv/y as a basis for determining the activity concentrations of artificial radionuclides below which unrestricted release can be allowed. For naturally occurring radionuclides, criteria based on the world-wide average levels of natural radionuclides in the environment are being proposed (around 0.5 Bq/g) as the level below which unrestricted release can be permitted.

Radiological criteria for the release of sites and buildings are not yet well established internationally and the Berlin Conference showed that a range of radiological criteria is currently being used in countries.

The release of materials and sites from regulatory control is a subject which concerns the general public, for obvious reasons. Thus, decision making in this area must take due account of the opinion of those who may be affected.

REFERENCES

[1] RENNEBERG, W., "Presentation of findings", Safe Decommissioning for Nuclear Activities (Proc. Int. Conf. Berlin, 2002), IAEA, Vienna (2003) 543–545.

[2] RITTSCHER, D., "Decommissioning of nuclear power plants: project planning and implementation in Greifswald and Rheinsberg", Safe Decommissioning for Nuclear Activities (Proc. Int. Conf. Berlin, 2002), IAEA, Vienna (2003) 325–340.

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Predisposal Management of Radioactive Waste, Including Decommissioning, IAEA Safety Standards Series No. WS-R-2, IAEA, Vienna (1999).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Nuclear Power Plants and Research Reactors, IAEA Safety Standards Series No. WS-G-2.1, IAEA, Vienna (1999).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, INFCIRC/546, IAEA, Vienna (1997).

[6] REISENWEAVER, D., "The IAEA's decommissioning concept", Safe Decommissioning for Nuclear Activities (Proc. Int. Conf. Berlin, 2002), IAEA, Vienna (2003) 123–130.

'GREEN VINCA'—VINCA INSTITUTE NUCLEAR DECOMMISSIONING PROGRAMME

M. PEŠIC, K. SUBOTIC, V. LJUBENOV, O. ŠOTIC
Centre for Nuclear Technologies and Research,
The Vinca Institute of Nuclear Sciences,
Belgrade, Serbia and Montenegro
Email: mpesic@vin.bg.ac.yu

Abstract

Current conditions related to the nuclear and radiation safety in the Vinca Institute of Nuclear Sciences, Belgrade, Serbia and Montenegro, are the result of the previous nuclear programmes in the former Yugoslavia and the strong economic crisis during the previous decade. These conditions have to be improved as soon as possible. The process of establishment and initialization of the Vinca Institute Nuclear Decommissioning (VIND) programme, known also as the 'Green Vinca' programme supported by the Government of the Republic of Serbia in 2002, is described in the paper. It is supposed to solve all problems related to the accumulated spent nuclear fuel, radioactive waste and decommissioning of RA research reactor. Particularly, materials associated with the RA reactor facility and radioactive wastes from the research, industrial, medical and other applications, generated in the previous period which are stored in the Vinca Institute, are supposed to be properly repackaged and removed from the Vinca site to some other disposal site, yet to be decided. Besides that, a research and development programme in the modern nuclear technologies is proposed with the aim to preserve experts, 'manpower' and to establish a solid ground for new researchers in the field of nuclear research and development.

1. INTRODUCTION

As a result of some ambitious nuclear programmes in the former Yugoslavia and a strong economic crisis during the previous decade, nuclear and radiation safety in the Vinca Institute of Nuclear Sciences, Belgrade, Serbia and Montenegro, has to be improved as soon as possible [1]. The main safety problems that have to be solved are:

(1) Spent nuclear fuel from the operation of the RA research reactor [2], that is stored in the water pools within the reactor building;
(2) Inadequate storage facilities for the low and intermediate radioactive wastes at the Vinca site.

Besides that, a complex decommissioning project of the RA facility [3], whose extended shutdown stage took almost 18 years, is initiated, too.

To solve the problems mentioned above, a new Vinca Institute Nuclear Decommissioning (VIND) programme was initiated and established in the Vinca Institute during 2002 [4, 5]. The programme team is assembled from about 60 experts from the Institute and relevant organizations. The programme is known also as the 'Green Vinca' programme, due to the adopted general idea that the Vinca site will be cleaned from the radioactive waste and a new one will not be generated in the future. The Government of the Republic of Serbia accepted this programme in July 2002. It is expected that the programme will be supported, besides the government funding and expected donation from foreign institutions, by experts' help from the IAEA. The necessary equipment will be obtained through the assistance from the IAEA's technical cooperation programme. Close cooperation of the team members with experts and relevant companies from the nuclear developed countries is expected.

2. RA RESEARCH REACTOR AND SPENT NUCLEAR FUEL

Heavy water research reactor RA (Fig. 1) was operated at a power of 6.5 MW in the Vinca Institute of Nuclear Sciences from 1959 to 1984, using 2%

FIG. 1. Heavy water 6.5 MW research reactor RA.

enriched and 80% enriched uranium fuel elements. Spent nuclear fuel elements comprise about 2.5 t of metal uranium (initial enrichment 2%) and about 16 kg of uranium dioxide (dispersed in aluminium matrix, initial uranium enrichment 80%). Both types of fuel elements, known as the TVR-S type, are ex-USSR in origin and have the same shape and dimensions, and approximately the same initial mass of ^{235}U nuclide (7.5 g and 7.7 g). The total of 8030 spent fuel elements are stored at the RA research reactor premises, almost all in the spent fuel pool filled by ordinary water (Fig. 2). The 480 highly enriched uranium spent fuel elements, the ones used during the last operation of the reactor, are kept in the drained RA reactor core since 1984.

Aluminium cladding, 1 mm thick, covers at both sides, a 2 mm thick and 10 cm long ring-type fuel layer of the TVR-S fuel element. Due to inadequate chemical parameters of ordinary water in the spent storage pool (pH, very high conductivity and chloride ions contents), the corrosion processes penetrated aluminium cladding as well as the aluminium walls of 30 storage barrels, during a storage period of between 20 and 40 years. The activity of the fission products (^{137}Cs nuclide) is detected in the water samples during the storage pool inspection in 1996, and experts from the IAEA, the Russian Federation and the

FIG. 2. Spent nuclear fuel pool of the RA research reactor.

United States of America were invited to help. Works on the remediation of the water transparency of the storage pool were made by end of 2001, including the removal of debris and about 3 m^3 of the radioactive sludge from the pool bottom. Inspections of the water samples taken out, by underwater drilling, from the aluminium storage barrels with the spent fuel elements were carried out, by the staff of the Vinca Institute and with the help of the experts from the Russian Federation and the IAEA. The increased specific activity of ^{137}Cs in the water samples, taken out from a few of the 30 aluminium barrels and from about 150 stainless steel containers (tubes) with the spent fuel elements, was measured between 1999 and July 2003. The measurements of the activity and the chemical parameters of the water samples will be continued by the examination of the last remaining 150 stainless steel storage containers with the spent nuclear fuel by the end of 2003.

For a number of reasons, both technical and political, the reactor has not been restarted after a long period of extended shutdown [2]. All plans for the reactor RA refurbishment, initiated with various intensities during the last 19 years, are abandoned. The proposal for the decommissioning of the reactor RA in the very near future, based on economical, technical and legislation reasons, was submitted to the Governments of the Federal Republics of Yugoslavia and to the Republic of Serbia, in 2001 [3]. During the summer of 2002, both Governments had made decisions on the final shutdown of the RA research reactor and its decommissioning.

Following the new initiatives on the international perspective on the nuclear spent fuel management, a proposal was sent to the Serbian Government to ship the spent fuel elements of the RA research reactor to the Mayak, the spent nuclear fuel reprocessing plant in the Russian Federation [4]. This proposal is in line with, and generally supported, the tripartite initiative established by the IAEA and the Governments of the USA and the Russian Federation. According to the decision of the Government of the Republic of Serbia, all unused, fresh highly enriched uranium fuel elements from the RA and RB reactors in Vinca have been shipped to the country of origin (the Russian Federation) in August 2002 [6]. For future experimental research in the field of nuclear reactors, only the operation of the RB critical assembly [7] in the Vinca Institute of Nuclear Sciences will be supported.

3. RADIOACTIVE WASTE IN THE VINCA INSTITUTE

Materials generated as radioactive waste, declared as low level waste (LLW) and medium level waste (MLW), during the operation of the RA and RB reactor facilities, radioactive LLW and MLW from research, industrial,

FIG. 3. The old hangar: the interim LLW and MLW storage at the Vinca site.

medical and other applications, generated in the previous period throughout the former Yugoslavia, are stored in the Vinca Institute in two interim storage hangars. Beside hangars, three underground tanks with liquid LLW and MLW exist, too. The older hangar (Fig. 3) is completely filled with radioactive waste. It is in very bad condition: the metal walls and roof are corroded. A newer hangar is almost half full with the RAW materials packaged in 200 l drums and 30 l barrels. The radioactive waste packages from the oldest hangar are supposed to be properly repackaged and stored temporally in the third interim storage (hangar), to be built near the other two. The third hangar should be designed is such a way to have enough space to accept all radioactive waste generated during forthcoming activities on the spent fuel shipment and decommissioning of the RA reactor, too. The oldest hangar will be dismantled after the management and removal of the radioactive waste packages to the third hangar. The liquid waste will be treated and properly stored ready for a future shipment, too. It is planned that all LLW and MLW will be removed from the Vinca site, in the future, to some long term national disposal site, yet to be decided. The basic idea of the Green Vinca is that new radioactive waste will not be created or stored at the Vinca site after that shipment.

4. GREEN VINCA: VINCA INSTITUTE NUCLEAR DECOMMISSIONING PROGRAMME

In order to solve the problems mentioned previously, the Green Vinca: Vinca Institute Nuclear Decommissioning Programme was initiated in the Vinca Institute in 2002. The structure of the programme is coordinated with IAEA experts on spent fuel, decommissioning and radioactive waste issues. The programme, run by the programme manager, consists of three inter-related projects and three supporting activities. Each project has its own project leader that closely cooperates with the programme manager and the Director General of the Vinca Institute, within frame of the Executive Program Board. The projects are:

- Safe Removal of Spent Fuel of the RA Research Reactor;
- Safe Management of Waste in the Vinca Institute;
- Decommissioning of the RA Research Reactor.

Supporting activities for all three projects include coordinated works on nuclear and radiation safety evaluation, health and medical protection, operational dosimetry and waste management and administration management.

The VIND programme's complete team is assembled from about 60 experts, half of whom are engineers, and the other half are technicians and administration staff. The staff members are from the Institute and other country's relevant organizations. The new organizational unit within the Vinca Institute, the Centre for Nuclear Technologies and Research (Centre NTI), was established in mid-2002 to carry on the main tasks with respect to spent fuel and decommissioning issues. Also, the Radiation Protection Laboratory of the Vinca Institute is engaged in the Green Vinca project with the aim to prepare and master appropriate dismantling and decontamination techniques for waste management, to obtain all licences, and to establish the appropriate waste management facilities and the temporary radioactive waste storage for low level and intermediate level waste at the Vinca site.

The project manager and the project leaders are supposed to work with the Vinca Institute's Director General with the aim to coordinate the project tasks within the programme, and also to coordinate the programme activities ofo the IAEA and the Ministry of Science, Technologies and Development of the Republic of Serbia. The Ministry of Science, Technologies And Development has also established the Green Vinca Committee in the autumn of 2002, as its expert advisory body with the aim to survey the programme activities and to support the programme in case of various difficulties.

Almost at the same time, starting from 2003, the IAEA established three projects within the Technical Cooperation Programme (TCP) to support the VIND programme in the next five to six years with a total funding of about $8.5 million, of which $5 million was agreed (with officials from the Governments of the former Yugoslavia and Serbia, as well as the USA and a non-governmental organization) as the financial support of the non-governmental organization Nuclear Threat Initiative (NTI). Each project of the IAEA has a technical officer in the IAEA that closely cooperates with the corresponding project leader of the Vinca's VIND programme. The first activities started in December 2002 and include (a) verification of spent fuel containers; (b) preparation of decommissioning plan; and (c) a feasibility study for consolidation and upgrading waste management at the Vinca site. Initial equipment, necessary to fulfil these tasks, was selected and purchased.

Besides the government funding (that is in an initial phase) and the expected donation from foreign institutions, the experts from the IAEA will support the VIND programme team through expert missions and training through workshops. Also, it is planned that the necessary special equipment will be obtained through the technical assistance of the IAEA. Close cooperation of the team members with various experts and relevant companies from the nuclear developed countries is expected.

The long term solution for the storage of aluminium cladded spent fuel is to ship the fuel elements to the supplier for reprocessing and the successive, adequate long term storage. An alternative, adopted by some countries (e.g. the USA) is to provide their own dry storage. According to a few IAEA expert missions visiting the Vinca Institute in 2001 and 2002, transportation of the fuel to the Russian Federation (or another site for reprocessing and final disposition) is under serious examination due to the existing technical difficulties affecting the cost of the operation. For these reasons, intermediate dry storage of the Vinca's spent nuclear fuel, for no less than 50 years, should also be considered, even if this option is not economically reasonable for a small amount of the spent nuclear fuel in the Vinca Institute.

If the Russian side accepts the proposal for receiving back the spent fuel elements of the RA research reactor for reprocessing and long term storage, some of the foreseen difficulties, related to the shipment and affecting contract conditions, are:

- Leaking fuel elements due to corrosion process;
- Inadequate conditions of existing storage containers;
- Inadequate dimensions of stainless steel containers and aluminium storage barrels in respect to available transport containers;

— Legal matters related to the licensing of existing Russian made transport containers according to the laws in Europe;
— Legal matters related to transit permissions to be issued by countries crossed in the transportation of the spent fuel should be carried on.

Activities related to the spent nuclear fuel project management, besides evaluation of the fuel shipment options and procedures, the continuation of the regular monitoring and maintenance of the water quality in the spent fuel storage pool and the working plans to improve the conditions in the existing temporary RA reactor spent fuel storage pool, in the near future are:

— Washing up corrosion deposits from all surfaces in contact with the pool water using the technology and equipment already provided by a Russian company;
— Design and production of special equipment for underwater cutting and removal (conditioning and storage at the temporary waste storage) of the corroded carbon steel structure in the basin 4 of the storage pool;
— Final removal of sludge from the spent fuel storage pool. Physical purification of pool water by mechanical filtering and chemical purification of the water using the ion exchange resins.

Activities related to the decommissioning project of the reactor RA in 2003, within the VIND project and the IAEA's TCP should establish, in the next 4 to 6 years, a detailed decommissioning plan, including the site characterization, and preparing and mastering the appropriate dismantling and decontamination techniques for the waste management. Simultaneously, it is necessary to obtain all necessary licences. Also, it is supposed that, within that time interval, the spent nuclear fuel will be removed from the site and that appropriate waste management facilities and temporary storage will be established in the Institute with the aim to allow an uninterrupted decommissioning progress.

It is also expected that national authorities will make the decision, as soon as possible, about the site selection and construction of the long term low level and intermediate radioactive waste storage in the country and establish the appropriate regulation organization structure, laws, acts, rules and directives. On the other hand, the government requires that the RA reactor be decommissioned and that the spent fuel be removed from its location so that the reactor building could be used for other purposes. These requirements are essential to the government support expected by the Green Vinca, i.e. the VIND programme. Besides that, a research and development programme in modern nuclear technologies is proposed to the Government with the aim to preserve experts and to establish a solid ground for new research activities in the nuclear power field, generally.

5. CONCLUSION

Solving the problem of the safe disposal of research reactor irradiated fuel and the decommissioning of a research reactor, including the provision of an adequate low level and intermediate level radioactive waste disposal site (even a temporary one) is a difficult task for a country with no long term nuclear power programme and with limited potential and resources. This paper describes the process of establishment and initialization of the Vinca Institute Nuclear Decommissioning Programme, known also as the Green Vinca Programme, with the aim to solve such problems. The first activities began in December 2002 and included verification of spent fuel containers, preparation of a decommissioning plan and feasibility study for the consolidation and upgrading of waste management at the Vinca site. Initial equipment necessary to fulfil these tasks was selected and purchased. Factors that may cause delays or prevent implementation of the above projects are lack of 'manpower', lack of material resources and necessary equipment, as well as general economic difficulties in the country.

ACKNOWLEDGEMENT

This paper was supported in part by the Ministry of Science, Technologies and Development of the Republic of Serbia through project no. 1958 "Transport processes of particles in fission and fusion systems", the Green Vinca Programme, and through the IAEA's TCP projects YUG/4/029, YUG/4/030 and YUG/4/031..

REFERENCES

[1] PLEĆAŠ, I., PEŠIĆ, M., PAVLOVIĆ, R., NEŠKOVIĆ, N., MATAUŠEK, M., "Arguments for New Yugoslav National Nuclear Scientific Program", SIEN 2001: Nuclear Energy Development in the South-East Europe—Nuclear Energy, NPP Operation and Maintenance (Proc. Int. Symp. Bucharest, 2001), Politehnika University, Bucharest (2002?).

[2] PEŠIĆ, M., KOLUNDŽIJA, V., LJUBENOV, V., CUPAĆ, S., "Causes of Extended Shutdown State of 'RA' Research Reactor in Vinca Institute", Extended Shutdown and Decommissioning of Research Reactors (Proc. Regional Workshop, Riga, 2001), IAEA Technical Co-operation Project RER/9/058: Safety Review of Research Reactor Facilities, IAEA, Vienna (2002).

[3] PEŠIĆ, M., SUBOTIĆ, K., ŠOTIĆ, O., PLEĆAŠ, I., LJUBENOV, V., PERIĆ, A., MILOŠEVIĆ, M., "Yugoslav Spent Nuclear Fuel Management Program and International Perspectives", Irradiated Nuclear Fuel Management 2002, INFM2002: Russian New Initiatives (Proc. Int. Conf. Moscow, 2002), JSC "Technabexport", Moscow (2002).

[4] PEŠIĆ, M., SUBOTIĆ, K., ŠOTIĆ, O., PLEĆAŠ, I., LJUBENOV, V., PERIĆ, A., Vinèa Nuclear Decommissioning Program, ETRAN (Proc. XLVI Yugoslav Conf. Teslić, Republika Srpska, 2002) Vol. IV, Banja Vrućica (2003?).

[5] LJUBENOV, V., PEŠIĆ, M., KOLUND_IJA, V., RA Research Reactor in Vinèa Institute—Decommissioning Project, USDOE/CNCAN Joint Decommissioning (Workshop, Bucharest, 2003), unpublished presentation.

[6] PEŠIĆ, M., ŠOTIĆ, O., HOPWOOD, JR., W. H., "Transport of HEU Fresh Fuel from Yugoslavia to Russian Federation", Nuclear Technology & Radiation Protection XVII 1-2 (2002) 71–76.

[7] PEŠIĆ, M., PLEĆAŠ, I., PAVLOVIĆ, R., "Safety Assessment of Various Operation Modes of RB Research Reactor", Topical Issues in Nuclear Safety (Proc. Int. Conf. Vienna, 2001), IAEA Vienna (2002).

INNOVATION IN RADIOACTIVE WASTEWATER STREAM MANAGEMENT

A. KARAMELDIN
Atomic Energy Authority,
Cairo, Egypt
Email: alykarameldin@hotmail.com

Abstract

Treatment of radioactive wastewater streams is receiving considerable attention in most countries having nuclear reactors. Among these countries is Egypt, which has two nuclear research reactors. The first research reactor, ETRR-1, has been in operation over 40 years, resulting in the accumulation of large quantities of wastewater collected in special drainage tanks (SDTs). Previous attempts were aimed at the volumetric reduction of streams present in SDTs, by a reverse osmoses (RO) system. The proposed RO system previously succeeded in reducing the water volume present in SDTs from 450 m^3 to 50 m^3 (during the period 1998 to 2000). The main drawbacks of the RO system are the additional amount of secondary wastes (turbidity and emulsion filters media replacement, and the excessive amounts of chemicals for cleaning, flushing and storing of the membranes), and a limited contaminant release in the SDTs area, resulting in the decommissioning of the RO system. Meanwhile, the SDTs waste contents reached 500 m^3, according to the inspection project of ETRR-1. Recently, an invention of a system for volume reduction of the wastewater streams present in SDTs has been achieved. This system substantially utilized the air conditioning and ventilation techniques in water transfer from the wastewater to air. This process is promoted by mutual heating and humidification of compressed dry air introduced through SDTs (or in another tank). From the point of view of the probable release of radioactive nuclides, the analysis of the evaporation of waste streams present in SDTs has indicated that the proposed optimal evaporating temperature is around 75^o C. The design curve of the daily volumetric reduction of the wastewater streams versus the necessary volumetric airflow rates at different operating temperature has been achieved. The evaporating temperature varied from 40^o C to 95^o C with a step of 5^o C. The obtained curve illustrates that the required volumetric airflow rate utilized to evaporate 1 m^3/d (when maintaining SDTs at a temperature of 75^o C) is less than 90 m^3/h. Assessments of the curve obtained have indicated that this system is feasible, viable and economic and has no secondary waste residuals. Recently, an experimental facility was proposed to be constructed to obtain the optimum operating parameters of the system, regarding probable emissions of the radioactive nuclides within the permissible release limits.

1. GENERAL

The management of radioactive wastes has become a major concern particularly with regard to the release of radioactive material to the environment and possible risk of contamination. The development of rational and acceptable options for radioactive waste disposal requires a clear understanding of radiators protection objectives and their application in planning, regulation and licensing. Considerable progress has been made over the past three decades within many countries utilizing nuclear reactors to develop strategies for the management of nuclear wastes. All wastes should be managed in such a way that high standards of conditioning are maintained and that potential hazards originating from their disposal are reduced to levels that are as low as reasonable and well below admissible levels [1].

2. RADIOACTIVE WASTEWATER VOLUME REDUCTION

At present, the research development work for the treatment of low and intermediate level radioactive wastes is concentrated on saving energy and raising efficiency. Evaporation, ion exchange and filtration have been widely used in the treatment of liquid wastes [2].

The evaporative techniques include:

- Submerged combustion evaporator technology that was modified for treatment of low level radioactive liquid wastes [3];
- Thermal evaporator where the water vapour from the evaporation process is condensed, filtered and can be pumped through an ion exchange bed before transfer to a retention basin. Meanwhile, the non-condensable portion of the vapour is filtered [4] and continuously monitored before venting to the atmosphere [5];
- Infrared heated method, which is ideal for the concentration of small amounts and low or medium radioactive liquid wastes [6];
- Solar evaporation basins for low level radioactive wastes[7];
- Solar evaporating ponds for low level radioactive wastes [8].

The ion exchange method is a costly process; therefore, it is used for purification of low radioactive waste where the loss of radioactive materials is less than in the evaporation methods [9]. The ion exchange can be used as a polisher and for caesium removal up to 99.83% without interference from other species [10]. Meanwhile, precipitation and flocculation processes give a low degree of purification. Membrane processes offer intermediate purification at reasonable cost [11]. Meanwhile, reverse osmosis and filtration is a novel process of low level radioactive waste management [12], and has been widely used using different types of membranes. Freeze-drying is one of the new methods [13].

3. PROBLEM IDENTIFICATION

The treatment of radioactive waste streams is receiving considerable attention in most countries containing nuclear reactors. Among these countries is Egypt, which has two research nuclear reactors. The first research reactor ETRR-1 has been in operation for 40 years, resulting in an accumulation of large quantities of wastewater in the special drainage tanks (SDTs). Previous attempts were aimed at the volumetric reduction of waste streams present in SDTs, by a reverse osmoses (RO) system. The proposed RO system previously succeeded in reducing the water volume present in SDTs from 450 m^3 to 50 m^3 (during the period 1998 to 2000). The main drawbacks of the RO system are the additional amount of secondary wastes (turbidity and emulsion filters media replacement, and the excessive amounts of chemicals for the cleaning, flushing and storing of the membranes), and a limited contaminant release in the SDTs area, resulting in the decommissioning of the RO system. Meanwhile, the SDTs wastewater contents recently reached 500 m^3, resulting from the inspection project of the ETRR-1, which used an excessive amount of inorganic detergents including EDTA, ammonia and citric acid, which drained ultimately to SDTs.

Figure 1 illustrates the ETRR-1 SDTs. Two identical tanks each have a storage capacity of 300 m^3. As shown in Fig. 1, each tank has a concrete cylindrical shape with an upper and lower bonnet structure and a steel lining at the basin and walls. The inner diameter of the tanks is 10 m. The heights of the upper and lower bonnets each reach about 1.5 m, while the cylindrical wall reaches about 3.8 m in height. A cylindrical concrete hatch of 1 m diameter and 2 m height is located at the top of the upper bonnet. The tanks are equipped with a set of drain and air vent piping, and manholes.

A recent analysis of ETRR-1 SDTs wastewater is shown in Table 1, where both the identified radioactive nuclides and corresponding activity concentration are given.

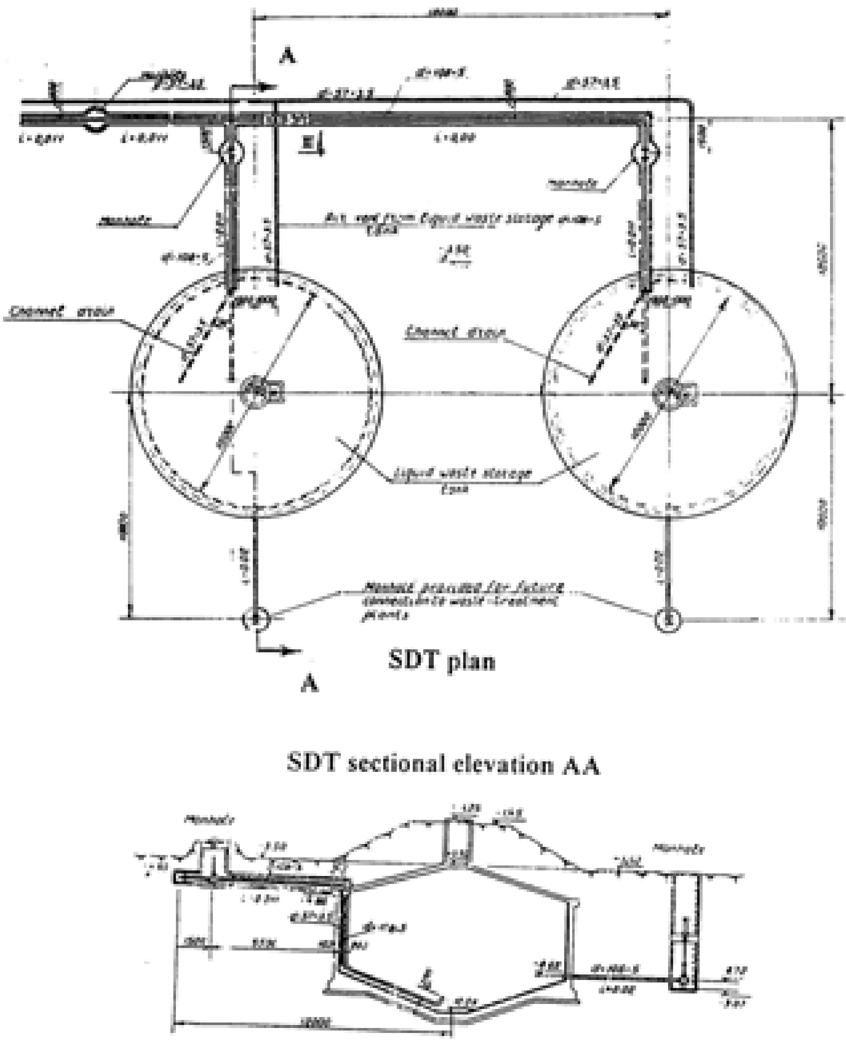


FIG. 1. Special drainage tanks and piping drawings.

TABLE 1. ETRR-1 SPECIAL DRAINAGE TANKS WASTEWATER ANALYSIS

Activity analysis		
Identified radioactive nuclides	Activity concentration, Bq/L	Mass concentration
Na-22	3.9 ± 0.15	—
Co-60	49.3 ± 1.7	1.2 pg/L
Sb-125	76.6 ± 6.2	1.95 pg/L
Cs-134	44.6 ± 2.3	0.93 pg/L
Cs-137	2683.6 ± 15.6	0.83 ng/L
Ce-144	127.2 ± 10	< 0.01 pg/L
Organic analysis		
Element to be analysed	Result	Units
Sulphate	84.4	ppm
Chloride	53	ppm
Nitrite	Not detected	—
Nitrate	350	μ g/L
Ammonia	0.55	ppm
Phosphate	1532.7	μ g/L
TS	404	ppm
TDS	388	ppm
TSS	16	ppm
COD	6	ppm
pH	7.94	
Conductivity	577	μ S/cm

4. RECENTLY INNOVATED METHOD FOR RADIOACTIVE WASTEWATERS BUBBLING VOLUME REDUCTION (RWBVR)

4.1. Physical bases

The vapour emitted from the wastewater (or any substance) exerts pressure known as vapour pressure, and as the temperature of the wastewater is raised, the vapour pressure increases because of the increase of evaporation.

When the local vapour pressure above the wastewater becomes equal to the total pressure because of heat, boiling will occur. For most solids at ordinary temperatures and pressures, the vapour pressure is small or negligible, noting that the vapour pressure can become important even for metals at an elevated temperature and reduced pressures. Conversely, when a solution of two or more volatile substances is heated, the resulting vapour will contain all substances, although generally in properties different from the original solution. A higher percentage of the more volatile vapour is normally evaporated first; this principle forms the basis of the process of desalination [14]

The air above the wastewater contains water vapour (moisture), and is known as humid air. The maximum amount of moisture content is dependent on the air temperature. It increases by the increase of the air temperature until the saturation conditions, at which no more water vapour is released from the wastewater. The weight of water vapour contained in a volume of air is defined as the absolute humidity. Meanwhile, the ratio between the actual water vapour content and the saturated water vapour content, at the same temperature, is defined as relative humidity. If the temperature of the air rises and no change occurs in the water vapour content of the air, the absolute humidity remains the same but the relative humidity is lowered. Conversely, a fall in temperature increases the relative humidity. However, a further decrease in temperature ultimately terminated the air to the dew point (100% relative humidity).

Table 2 gives the average monthly meteorological conditions based on the observed data of the Helwan meteorological station for the period 1904–1954. It is worth mentioning that these data were used for the Inshas site in the design of ETRR-1's cooling tower [15].

4.2. Thermodynamic bases of gas-vapour mixtures

The water vapour can be treated as an ideal gas with negligible error (under 0.2%), even when it is a saturated vapour. So, the atmospheric pressure, p, can be determined through the vapour pressure, p_v, and air pressure, p_a [16].

$$w = \frac{m_v}{m_a} = \frac{\mathrm{p_v} V/R_v T}{\mathrm{p_a} V/R_a T} = 0.622\frac{\mathrm{p_v}}{\mathrm{p_a}} = \frac{0.622 \cdot \mathrm{p_v}}{\mathrm{p\text{-}p_v}} \tag{1}$$

The absolute humidity or specific humidity (humidity ratio), w, is the mass fraction of water vapour, m_v, to dry air, m_a.

$$\varphi = \frac{m_v}{m_g} = \frac{\mathrm{p_v} V/R_v T}{\mathrm{p_g} V/R_v T} = \frac{\mathrm{p_v}}{\mathrm{p_g}} \tag{2}$$

TABLE 2. AVERAGE METEOROLOGICAL CONDITIONS AT THE INSHAS SITE

Average year (month)	Air temperature, T_a °C			Relative humidity φ (%)	Wet-bulb air temperature (T_{aw} °C)	Calculated air properties		
	Average monthly	Absolute maximum	Absolute minimum			w kg_v/kg_a	m_v kg/m^3	M_a kg/m^3
January	12.2	29.5	1.6	55	8.3	0.0049	0.0094	1.212
February	13.3	33.4	1.6	47	8.4	0.0045	0.0054	1.209
March	16.5	38.5	3.4	44	10.7	0.0052	0.0062	1.193
April	20.3	44.9	5.7	37	12.8	0.0056	0.0065	1.177
May	24	46	10.7	34	15.2	0.0064	0.0074	1.161
June	26.7	47.5	13	36	17.5	0.0080	0.0091	1.147
July	27.5	42.9	16	42	19.2	0.0098	0.0111	1.141
August	27.3	42	17.2	45	19.2	0.0103	0.0118	1.141
September	25.3	42.4	14.6	49	18.5	0.0100	0.0115	1.150
October	23.1	40.3	10.4	50	16.8	0.0089	0.0100	1.160
November	19	37.7	5.7	53	13.8	0.0073	0.0087	1.179
December	14.1	30.4	1.3	55	9.9	0.0056	0.0067	1.202

The relative humidity, φ, as defined as the air moisture content, m_v, relative to the maximum amount of moisture, that air can hold, m_g.

$$\varphi = \frac{m_v}{m_g} = \frac{p_v V / R_v T}{p_g V / R_v T} = \frac{p_v}{p_g} \tag{3}$$

Combining equations (2) and (3), therefore,

$$\varphi = \frac{w \cdot p}{(0.622 + w) \cdot p_g} \quad , \quad \text{and} \quad w = \frac{0.622 \cdot \varphi \cdot p_g}{p - \varphi \cdot p_g} \tag{4}$$

The enthalpy of the moist air, H, is the sum of the enthalpies of the dry air, $m_a.h_a$, and water vapour, $m_v.h_v$,

$$h = h_a + w \cdot h_g \tag{5}$$

Different processes can done on an air stream, such as heating, cooling, humidification, dehumidification, etc. As discussed, the heating process resulting in lowering the relative humidity. In addition, humidification of an air stream results in an increase of the relative and absolute humidity. The mutual heating–humidification process is accomplished by spraying water in the air stream, a part of the latent heat of vaporization will come from air, and therefore, heating of the air is performed. This process results in increasing the absolute humidity.

4.3. Radioactive wastewaters bubbling volume reduction (RWBVR) theory

The innovative method for volume reduction of radioactive wastewater shown in the present paper is to evaporate the wastewater at a relatively low temperature below the boiling temperature, in accordance with the radioactive nuclides, analysis, volatility and boiling points, as required. This method makes use of the classical methods of heating–humidification used in air conditioning and similar to the heat and mass transferred in the cooling towers. It can be moved down by forcing a dry airflow to be bubbled through heated wastewater to transfer a part of its mass to the bubbled air. The wastewater tank pool is heated to a controlled elevated temperature below the boiling point. Then compressed air introduced through a set of nozzles at the bottom of the wastewater tank, directing it to bubble through the wastewater. This allows the process of heat and mass transfer from the wastewater to the bubbling air to be performed. The required heat for vapourization of the transferred mass from the wastewater to the bubbling air (latent heat) can be supplied from radioactive nuclides residual heats or/and additional heating.

5. RADIOACTIVE WASTEWATERS BUBBLING VOLUME REDUCTION (RWBVR) CALCULATIONS AND RESULTS

Using equations (1) though (5), for the data listed in Table 2 at the Inshas site, one can calculate the average monthly air vapour content (moisture). This is being varied between about 5 g/m^3 up to 12 g/m^3 of water vapour, with a yearly average of 8.4 g/m^3 of water vapour.

When dry air is forced to bubble through the heated wastewater in SDTs, the mass and the heat transferred to the air is maximized. Therefore, the air at the outlet of the SDTs can be assumed to be saturated air, and its temperature is the same as the wastewater temperature [17]. Noting that at each wastewater temperature, the water vapour content presences at the saturation conditions are different, and increases with the increase of the SDTs controlled elevated operating temperature. Assuming that the SDTs are subjected to a continuous controlled heating up to a temperature that varies from 40° C up to 95° C, with a temperature step of 5° C. Reusing equations (1) though (5), one can calculate the specific humidity of the outlet air from the SDTs. Consequently, the masses of the air and water vapour per 1 kg of the saturated air can be calculated. In addition, the required air mass to evaporate 1 cubic m of water from the wastewater can be calculated and its hourly volumetric flow rate. The results of these calculations are obtained by running a simple computer program. The obtained results are presented in Table 3, and illustrated in Fig. 2.

It is noted from Table 3 and Fig. 2 that as the temperature of the wastewater SDTs increases, the required air volumetric flow rate decreases. At the wastewater temperature of about 40° C, the required volumetric airflow rate dramatically decreases tells the wastewater temperature of 60° C, wherein slowly decrease the rest of the curve. The required air volumetric flow rate at the wastewater temperature of 60° C and 75° C are about 220 and 88 m^3/h. The latent heat required for vapourization of 1 kg of water is of the order of 2500 kj/kg, while 1 kW.hr contains 3600 kj. Considering the heat losses and re-heating the SDTs bubbling air to pass through a filter before release to the atmosphere, one can satisfactory take the electric power consumption of 1 kW.hr/kg of

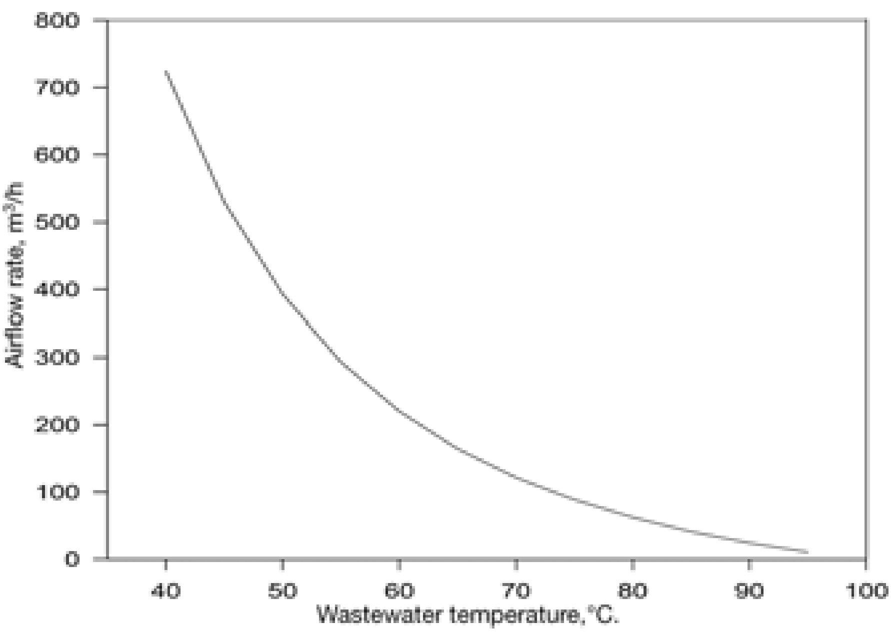


FIG. 2. Relation between airflow rate and evaporating temperature to evaporate 1 m^3/d.

TABLE 3. CALCULATED AIR QUANTITY TO EVAPORATE ONE CUBIC METER OF WATER FROM WASTEWATER SDTs

Temperature (°C)	w kg_v/kg_a	m_v kg/m^3	m_a kg/m^3	m_a kg /1 m^3 H_2O	m_a kg/h /1 m^3 H_2O/24h	m_a m^3/h /1 m^3 H_2O/24h
40	0.0495	0.0510	1.0306	24169	1007.04	724.05
45	0.0660	0.0653	0.9901	17388	724.5	529.23
50	0.0875	0.0827	0.9452	12714	529.76	393.06
55	0.1163	0.1041	0.8945	9343	389.29	293.3
60	0.1547	0.1296	0.8375	6908	287.82	220.16
65	0.2077	0.1604	0.7725	5081	211.70	164.36
70	0.2815	0.1968	0.6990	3709	154.56	121.77
75	0.3906	0.2401	0.6147	2652	110.52	88.34
80	0.5596	0.2866	0.5121	1841	76.69	62.16
85	0.8533	0.3500	0.4102	1201	50.03	41.14
90	1.4590	0.4184	0.2868	699	29.14	24.30
95	3.4104	0.4979	0.1460	298	12.43	10.51

wastewater volume reduction. Taking a price of 6 cents/kW.hr, can lead to a cost round $60/m^3 of wastewater volume reduction.

6. DISCUSSION

From the above analysis, the design curve of the proposed innovated method for radioactive wastewater volume reduction of SDTs is viable and feasible. This method among the surveyed methods is distinguished by the simplicity, costless and almost secondary 'waste-less' process. From the emission of radioactive nuclides point of view, a condensed study on the analysis of the wastewater radioactive nuclides present in the SDTs has been carried out. Moreover, the most probable future radioactive nuclides that may accidentally occur are also studied.

Generally, the most important parameter in the thermal methods is the probable release of the radioactive nuclides. Actually, the radioactive nuclides may exist as chemical compounds. These compounds may evaporate at a temperature lower than the original nuclides. Therefore, a study of the existing nuclides is considered. This study handled the classified isotopes concerning the melting and boiling temperatures, number of existing isotopes, and their minimum and maximum half-life, the number of their chemical compounds and the lowest temperature at which the compounds decompose. Moreover, a list of the radioactive nuclides of ETTR-1 and ETRR-2, which may release and their analysis [18] are listed in Table 4.

The analysis of Table 4 from the existing nuclides isotopes and half-life indicates that the boiling point of the Antimony penta-Chloride is 79^{o} C, noting that the nuclide compounds decomposition may result in higher boiling points. Therefore, the operating temperatures must be carefully chosen below this temperature. It is worth mentioning that the exit temperature of the ventilating air system reached about 78.9^{o} C, which present in a recent test results by the Westinghouse Hanford company using a simulated small-scale waste tank [17]. From the site of Inshas characteristics point of view and the structure of SDTs (the upper bonnet is made of concrete without steel lining), a proposed operating temperature of 65^{o} C may be suitable. This temperature may be suitable for the expected events on the temperature variations in this site.

This recent innovated method may be the best of all methods that deals with the wastewater tanks that generate a high quantity of residual heats [17, 19, 20]. In the Hanford site, air lift circulating system to mix the waste during storage, where air passes through it as bubbles [17]. By the recent method, a mutual cooling and volume reduction can be performed by the same process. Moreover, in this case, the airflow is the governing parameter that can control the wastewater tanks temperature. In addition, it is distinguished by the secondary ‘waste-less’ and low operating cost.

7. FUTURE WORK

In the second part of this paper, an experimental setup proposed to be constructed to determine the main optimal operating parameters. These parameters include wastewater level in the wastewater tanks level, airflow rate, exact heat requirement, and optimal operating conditions. Moreover, the release of radioactive nuclides can be determined in accordance with the operating temperature so, if it occurred, the required treatment system can be assigned.

TABLE 4. IDENTIFIED RADIONUCLIDES AT INSHAS SITE SPECIAL DRAINAGE

Present radio-nuclides	Melting/ boiling points	No. of isotopes	Isotopes minimum/ maximum H.L.	No. of compounds	Compounds minimum boiling points °C	Radio nuclides		Half life (H.L.)
						ETRR-2	ETRR-1 SDT	
Sodium	97.8/ 882.9	17 ($_{11}Na^{19}$ - $_{11}Na^{35}$)	1.5 ms ($_{11}Na^{35}$) 2.605 y ($_{11}Na^{22}$)	264	100 $NaH_2PO_3.2.5H_2O$ (orthophosphite dihydrogen)	-----	$_{11}Na^{22}$	2.605 y
Chromium	1857/ 2672	19 ($_{24}Cr^{44}$ - $_{24}Cr^{62}$)	0.05 s ($_{24}Cr^{44}$) 27.7 d ($_{24}Cr^{51}$)	48	117 CrO_2Cl_2 (oxychloride)	$_{24}Cr^{51}$	-----	27.7 d
Cobalt	1495/ 2927	21 ($_{27}Co^{50}$ - $_{27}Co^{70}$)	0.15 s ($_{27}Co^{68}$) 5.27 y ($_{27}Co^{60}$)	107	48, de-composes 55 $Co(NO)(CO)_3$ (nitrosylcarbonyl)	-----	$_{27}Co^{60}$	5.27 y
Rubidium	38.89/ 686	29 ($_{37}Rb^{74}$ - $_{37}Rb^{102}$)	53 ms ($_{37}Rb^{100}$) 4.88E10 y ($_{37}Rb^{87}$)	67	60 , - $4H_2O$ Rb $NO_3.Nd(NO_3)_3.4H_2O$ (neodymium nitrate)	$_{37}Rb^{88}$	-----	17.8m
Strontium	769/ 1384	26 ($_{38}Sr^{77}$ - $_{38}Sr^{102}$)	66 ms ($_{38}Sr^{102}$) 29.1 y ($_{38}Sr^{90}$)	62	100, - 6 H_2O $SrCl_2.6H_2O$ (chloride hexahydrate)	-----	$_{38}Sr^{90}$	29.1 y
Zirconium	1853/ 4377	25 ($_{40}Zr^{80}$ - $_{40}Zr^{104}$)	0.81 s ($_{40}Zr^{90}$ m) 1.5 E6 y ($_{40}Zr^{93}$)	32	210, $-8H_2O$ $ZrOCl_2.8H_2O$ (chloride)	$_{40}Zr^{95}$	-----	64.02d

TABLE 4. IDENTIFIED RADIONUCLIDES AT INSHAS SITE SPECIAL DRAINAGE (cont.)

Niobium	2477/ 5127	16 ($_{41}Nb^{103}$ - $_{41}Nb^{118}$)	1 s ($_{41}Nb^{106}$) 3.7 E7 ($_{41}Nb^{92}$)	16	254 $NbCl_5$ (chloride penta)	$_{41}Nb^{95}$	-----	34.97 d
Ruthenium	2310/ 3900	23 ($_{44}Ru^{91}$ - $_{44}Ru^{113}$)	1.5 s ($_{44}Ru^{111}$) 1.02 y ($_{44}Ru^{106}$)	10	De-composes at 108 RuO_4 (oxide tetragonal)	$_{44}Ru^{103}$	-----	39.27 d
Antimony	630.63/ 1750	31 ($_{51}Sb^{106}$ - $_{51}Sb^{136}$)	0.82 s ($_{51}Sb^{136}$) 2.758 y ($_{51}Sb^{135}$)	28	79 $SbCl_5$ (chloride penta)	$_{51}Sb^{122}$ $_{51}Sb^{124}$ -----	----- ----- $_{51}Sb^{125}$	2.72 d 60.2 d 2.76 y
Iodine	113.7/ 184.4	34 ($_{53}I^{109}$ - $_{53}I^{142}$)	0.11 ms ($_{53}I^{109}$) 1.7 E7 y ($_{53}I^{129}$)	14	98 IF_6 (fluoride, hexa)	$_{53}I^{131}$ $_{53}I^{133}$ $_{53}I^{135}$	----- ----- -----	8.04 d 20.8 h 6.57 h
Cesium	28.44/ 671	36 ($_{55}Cs^{113}$ - $_{55}Cs^{148}$)	33 ms ($_{55}Cs^{113}$) 2.3 E6 ($_{55}Cs^{135}$)	81	150, - Br_2 $CsBr_2Cl$ (dibromochloride)	----- $_{55}Cs^{137}$ $_{55}Cs^{138}$	$_{55}Cs^{134}$ $_{55}Cs^{137}$ -----	2.07 y 30.3 y 32.2m
Cerium	798/ 3443	30 ($_{58}Ce^{123}$ - $_{58}Ce^{152}$)	1 s ($_{58}Ce^{151}$) 284.6 d ($_{58}Ce^{144}$)	44	De-composes at 200 $Ce(NO_3)_3.6H_2O$ (III nitrate)	$_{58}Ce^{141}$ $_{58}Ce^{144}$	----- $_{58}Ce^{144}$	32.5 d 284.6d

REFERENCES

[1] MERZ, E.P., “Multi barrier effectiveness as the expedient measure for selecting the appropriate stabilization and immobilization procedure for the various waste categories”, Advanced Workshop on Defence Nuclear Waste Disposal in Russia: Implications for the Environment, Krasnoyarsk, Russia Federation, 1996, NATO, Brussels (1996?).

[2] WANG-XIANDE, SUN-MINGSHENG, Research and development on the technology for the treatment of low- and intermediate-level liquid radioactive wastes in the last ten years, J. Nucl. Radiochem. **V 12** 2 (Mar. 1990).

[3] PETERS, G.A., in Waste Processing of Chemical Cleaning Solutions, 53. Annual American Power Conference, Chicago, IL, 1991.

[4] JOHNSON, R.E., “Determination of the radio nuclide release factor for an evaporator process using non-destructive assay”, Proc. Conf. on Nuclear Air Cleaning and Treatment, Minneapolis, MN, 1998, DOE/NRC, Washington, DC (1998?).

[5] JENSEN, C.E., 1998 242-A interim evaporator tank system integrity assessment plan, Funding Organization: USDOE Office of Environmental Restoration and Waste Management, Washington, DC (1998).

[6] PENG-XUEMING, CHEN-LIANG, LU-ANHUA, ZHOU-GUANGKUI, WEN-DEQUAN, CHEN-LI, Study on infrared heated method for concentration of radioactive liquid, China Nuclear Power Inst., Nuclear power engineering, China, V18(6), Dec, 1997.

[7] HEINE, W.F., “Action taken to remediate the 183-H solar evaporation basins at the Hanford, Washington site”, Spectrum'88 Proc. Int. Topical Mtg on Nuclear and Hazardous Waste Management, Pasco, WA, 1988.

[8] LIU-KEQIANG, ZOU-CHANGGUI, Study on radio nuclides transport from natural evaporating ponds to the atmosphere, China Nuclear Information Centre, Beijing (1997).

[9] PYUNG-GOOK, J., CHONG-MOOK, P., SEUNG-GY, R., Comparison of ion exchange method and evaporation method for the detection of radioactivity in water, J. Korean Assoc. Radiat. Prot. **13** 2 (1988).

[10] HERBST, R.S., LAW, J.D., TODD, T.A., WOOD, D.J., WADE, E.L., Integrated AMP-PAN, TRUEX, and SSREX Flow sheet test to remove caesium surrogate actinide elements, and strontium from INEEL tank waste using sorbent columns and centrifugal contactors, USDOE Office of Environmental Management, DOE, Washington, DC (2000).

[11] ANTONESCU, M., “Treatment of radioactive waste liquids by membrane separation techniques”, SIEN ’97, Nuclear Energy: Radioactive Waste Management (Int. Symp. Bucharest, 1997) Energy Information Documentation Centre, Bucharest (1997).

[12] SHEKHAR, P., SUDESH-NATH, GANDHI, P.M., MISHRA, S.D., “Design features a reverse osmosis demonstration plant for treatment of low level radioactive waste”, ACE-94 advances in chemical engineering in nuclear and process industries, Mumbai, June 1994.

[13] COPPA, N.V., STEWART, P., RENZI, E., Freezing drying method, US patent document 8-716,008 (1999).
[14] Encarta® 97 Encyclopaedia. © 1993-1996 Microsoft Corporation.
[15] Two-cell induced-draft cooling tower work drawing, Catalogue number 869/793, Moscow (1960).
[16] CENGEL, Y.A., BOLES, M.A., Thermodynamics and an engineering approach, 2nd ed., McGraw-Hill, New York (1994).
[17] LEE, Y.J., AND DALPIAZ, E.L., "Development of a computer code to predict a ventilation requirement for an underground radioactive waste storage tank", 24. Nuclear Air Cleaning and Treatment Conference, Portland, OR, 1996.
[18] LIDE, D.R., FREDERKSE, H.P.R., CRC Handbook of Chemistry and Physic, a Ready-Reference Book of Chemical and Physical Data, CRC Press, Boca Raton, FL (1994).
[19] TURVEY, F., HONE, C., Storage of liquid high-level radioactive waste at Sellafield. An examination of safety documentation, Radiological protection 1st. of Ireland, Dublin, 2000.
[20] SATHYANARAYANA, K., Evaluation of waste temperature in AWF tanks for bypass mode operation of 702-Az ventilation system (Project W-030), Fluor Daniel Hanford Inc., Richland, WA, 1997.

FUEL MANAGEMENT

(Session 4)

Chairperson

J. KLEIN
Chile

Co-Chairperson

I. RITCHIE
IAEA

BLANK

RUSSIAN RESEARCH REACTOR FUEL RETURN PROGRAMME

T. DEDIK, I. BOLSHINSKY
Department of Energy/National Nuclear Security Administration,
Email: trisha.dedik@hq.doe.gov

A. KRASS
Department of State

Washington, D.C.,
United States of America

Abstract

For almost four years the United States of America, the Russian Federation and the IAEA have been discussing an initiative to return research reactor fuel of former Soviet Union or Russian origin to the Russian Federation. In a series of bilateral and trilateral meetings in Vienna and Moscow, considerable progress has been made towards defining the Russian Research Reactor Fuel Return Programme, as well as obtaining the necessary technical data to facilitate the return. More than 20 research reactors in 17 countries that have Soviet- or Russian-supplied fuel have identified. Most of these reactors have stocks of both fresh and irradiated HEU fuel that must be carefully stored and managed for many years to come. On 21 September 2003, the Russian Research Reactor Fuel Return Programme shipped 14 kg of fresh Russian-origin HEU fuel from Romania to the nuclear fuel fabrication facility in the Russian Federation, which represented the beginning of the practical implementation of the programme.

1. BACKGROUND AND HISTORY

Beginning December 1999, and continuing to the present, representatives from the United States of America, the Russian Federation and the IAEA have been discussing a programme to return to the Russian Federation Soviet- or Russian-supplied HEU fuel currently stored at foreign research reactors. The primary goal of this Russian Research Reactor Fuel Return (RRRFR) Programme is to advance nuclear non-proliferation objectives by eliminating stockpiles of highly enriched uranium (HEU) and encouraging eligible countries to convert their research reactors from HEU to low enriched uranium (LEU) fuel upon availability, qualification and licensing of suitable LEU fuel.

The goal of minimizing international commerce in HEU has been a pillar of US non-proliferation policy since 1978. In that year, the Reduced Enrichment for Research and Test Reactors (RERTR) Programme was initiated, in order to:

(1) Develop and qualify new LEU fuels that could replace HEU used in reactors of US design;
(2) Aid reactor operators with the analyses required to optimize performance of LEU fuels;
(3) Convert to LEU fuels.

The Russian Federation has its own RERTR Programme, under which it has significantly reduced enrichments on exported research reactor fuel and is working in cooperation with the US programme to develop new LEU fuels suitable for use in Russian-designed research reactors.

To complement the RERTR Programme, the Department of Energy (DOE) established the Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) Acceptance Programme in 1996. Under this programme, the USA accepts specified types of US-supplied spent and unused fresh fuel for management and disposition in the USA, on the condition that operators agree to convert their reactors to LEU as soon as practicable and, in any event, to not use HEU fuel in the reactor after the programme's scheduled end in 2006.

As the Acceptance Programme draws closer to its termination date, an increased number of requests for programme extension have been received. Currently, there are no plans to extend the policy beyond its current expiration date, therefore, eligible reactor operators interested in participating in this programme are strongly encouraged to evaluate their inventory, plan for future shipments and to contact the USA as soon as possible.

2. RESEARCH REACTOR FUEL RETURN PROGRAMME

Based on the success of the US Acceptance Programme, the DOE, supported by the Department of State, is working to bring about a similar effort in the Russian Federation. Trilateral discussions among the USA, the Russian Federation and the IAEA in Vienna have identified more than

20 research reactors in 17 countries that have Soviet- or Russian-supplied fuel.[1]

Most of these reactors use at least some HEU fuel, and many have stocks of both fresh and irradiated fuel that must be carefully stored and managed for many years to come. The goal of the DOE's National Nuclear Security Administration (DOE/NNSA) is to assist the Russian Federation to develop a broad based HEU minimization policy under which it would accept the return of spent and fresh HEU fuel from Soviet- or Russian-supplied foreign research reactors and develop new fuels that will allow conversion of such reactors to LEU. DOE/NNSA officials have led discussions with representatives from the Russian Federation's Ministry of Atomic Energy (MinAtom) and the IAEA on this issue, with the IAEA providing both technical and organizational support to the initiative.

President Bush has committed the US Government to strong, effective cooperation with the Russian Federation and the other States of the former Soviet Union to reduce weapons of mass destruction and to prevent their proliferation. To ensure that the promise of these programmes is fully realized, the Administration has undertaken a detailed review of US non-proliferation and threat reduction assistance to the Russian Federation. One conclusion of this review was the endorsement of the Takeback Programme as an important non-proliferation initiative that should continue.

The RRRFR Programme is an important aspect of the Administration's commitment to cooperate with the other nations to prevent the proliferation of nuclear weapons and weapons-usable/proliferation-attractive nuclear materials.

The USA provides funding to the RRRFR Programme based on the following criteria:

- The fuel return programme will include only existing former Soviet Union or Russian Federation research/test reactors in eligible countries that possess nuclear fuel supplied by the Former Soviet Union or the Russian Federation.
- Any country desiring to return fuel to the Russian Federation must agree to either convert its operating research or test reactor(s) using Soviet- or Russian-supplied nuclear fuel to LEU as soon as (i) suitable LEU, licensed by the country's national regulatory authority, is available, and (ii) the reactor's existing inventory of HEU is exhausted; or permanently shut down the reactor(s).

---

[1] Belarus, Bulgaria, China, Czech Republic, Democratic People's Republic of Korea, Egypt, Germany, Hungary, Kazakhstan, Latvia, Libya, Poland, Romania, Ukraine, Uzbekistan, Vietnam and Serbia.

— Whenever possible, all available HEU must be made available for return to the Russian Federation before any LEU is returned.
— All nuclear fuel to be delivered to the Russian Federation under the programme must be handled in accordance with IAEA INFCIRC/225/REV.4, INFCIRC/153 (corrected) and subsequent revisions thereto.

The Governments of the USA and the Russian Federation, and the IAEA will seek to encourage financial support from other IAEA Member States, where required, for the fuel return programme to supplement any US Government financial contributions.

In February 2003, USDOE tabled with MinAtom of the Russian Federation, the Government-to-Government Agreement between the Governments of the USA and the Russian Federation concerning the return of Russian research reactor nuclear fuel to the Russian Federation. This Agreement defines the terms and conditions for importation of the Russian-designed research reactor fuel to Russia and provides the legal framework for the RRRFR Programme. The Agreement is in the last stages of approval, having been under Russian interagency review since February. We are most anxious to conclude the Agreement and expeditiously implement the RRRFR Programme.

In spite of the absence of a signed formalized framework agreement thus far, we have made some progress to date. I am extremely pleased that last Friday, 7 November 2003, DOE Secretary of Energy Abraham, along with MinAtom Minister Rumyantsev, signed a joint statement reaffirming our common objective of reducing, and to the extent possible, ultimately eliminating the use of HEU in civil nuclear activity. Minister Rumyantsev unequivocally stated that the Russian Federation would complete all necessary preparations, which would allow signature of the Government-to-Government Agreement in the very near future.

3. FRESH HEU FUEL SHIPMENT FROM ROMANIA

On 21 September 21, 14 kgs of fresh Russian-origin highly enriched uranium (HEU) were returned from Romania to the Russian Federation. It was the first shipment of Russian-origin research reactor fuel under the RRRFR Programme. The HEU was airlifted from Bucharest, Romania to Novosibirsk, Russian Federation, where it will be down-blended and used in a nuclear power plant.

The shipment was a joint effort of the US DOE, the Russian Federation, the Romanian Government and the IAEA. The IAEA played an instrumental role in arranging the shipment from Romania and providing the contractual vehicle that made the shipment possible. The USA is deeply grateful to the IAEA for its role in this process.

The highly enriched nuclear fuel assemblies were originally supplied to Romania by the former Soviet Union for the Russian-designed 2 MW research reactor, located near Bucharest, Romania. The reactor was shutdown in December 1997, and is being decommissioned. The fresh nuclear fuel was loaded in 8 fresh fuel transportation canisters provided by the Russian Federation. IAEA safeguards inspectors and US DOE technical experts monitored the process of loading the fuel in the canisters. An IL-76 Russian cargo plane was used to complete the air shipment of the HEU fuel from Romania. We are still awaiting completion of all the legal documentation necessary before we carry out the first shipment of spent fuel under the RRRFR; however, Russian Federation Minister Rumyantsev and US Secretary of Energy Abraham jointly announced that they are nearing completion of preparations for another fresh fuel shipment to take place in the very near future.

4. PILOT SHIPMENT OF SPENT FUEL

The first candidate for the pilot shipment of spent fuel to the Russian Federation is Uzbekistan, whose Government has expressed a strong interest in participation in the RRRFR Programme. Uzbekistan possesses a VVR-SM research reactor at the Institute of Nuclear Physics, Uzbekistan Academy of Sciences, located in Ulugbek, about 30 km northeast of Tashkent. It is a heavily used 10 MW reactor of Soviet design that carries out an active programme of research and isotope production. From its first criticality in 1959, it used 90% enriched HEU fuel, but was converted to 36% fuel in 1989. Over its lifetime, the reactor has generated a large amount of spent fuel and made a number of spent fuel shipments to the reprocessing facility at Mayak in the past. Personnel who participated in the early shipments are still at the facility, so experience has been maintained. The facility has the necessary room and hardware to accommodate the transportation casks that will be utilized in the future shipments.

DOE has provided assistance to improve the physical protection system of the VVR-SM reactor, but it is located in a politically volatile region of Central Asia. All parties agree that the spent HEU and any remaining fresh HEU should be relocated to a more secure environment, thus removing it as a potential proliferation risk. Similar non-proliferation, physical security and safety concerns apply to other research reactors.

5. IN ANTICIPATION OF THE SHIPMENT

On 12 March 2002, DOE and Uzbekistan's Ministry of Foreign Affairs signed an Agreement to facilitate cooperation between the parties for the return of Uzbekistan's Russian-supplied nuclear fuel to the Russian Federation. Last summer, the DOE and the Institute of Nuclear Physics of Uzbekistan completed the facility preparation for the spent fuel shipment to the Russian Federation. Facility personnel have completed the training necessary to carry out the shipment. However, according to the new Russian environmental law, an ecological expertise needs to be conducted before importation of spent nuclear fuel to the Russian Federation. INP, Uzbekistan has signed a contract with the Mayak facility in the Russian Federation to provide an ecological expertise. An optimistic prognosis is that it will be completed in about three months. Simultaneously with the ecological expertise, TENEX of Russia is working with KATEP in Kazakhstan on preparation of all necessary approvals for spent fuel transit through Kazakhstan. We are hopeful that this first spent fuel shipment will move in the very near future and that this will provide the impetus to begin regular shipments that will allow us to totally eliminate the stores of HEU at these sites over the next decade or so.

EFFECT OF COUPON ORIENTATION ON CORROSION BEHAVIOUR OF ALUMINIUM ALLOY COUPONS IN THE SPENT FUEL STORAGE SECTION OF THE IEA-R1 RESEARCH REACTOR

O.V. CORREA, R.M. LOBO, S.M.C. FERNANDES,
G. MARCONDES, L.V. RAMANATHAN
Materials Science and Technology Centre,
Instituto de Pesquisas Energéticas e Nucleares (IPEN),
São Paulo, Brazil
Email: lalgudi@ipen.br

Abstract

Surveillance programmes to monitor the corrosion of aluminium clad spent research reactor fuels have used test racks containing horizontal metallic coupons. Spent MTR-type fuel elements are usually stored vertically, with their fuel plates, also vertical. Hence, the influence of coupon orientation on the corrosion behaviour of aluminium alloy coupons exposed to the spent fuel storage section of the IEA-R1 research reactor in São Paulo, Brazil, has been studied. Circular coupons of aluminium alloys AA 1050 and AA 6061, oriented both vertically and horizontally, were exposed to the storage section water for a year. Individual and coupled coupons were exposed to simulate general, crevice and galvanic corrosion. The storage section water parameters were periodically measured. Pitting was the main form of corrosion and coupon orientation had a marked effect on the extent of pitting. Vertically oriented coupons pitted less than horizontally oriented coupons.

1. INTRODUCTION

In Latin America, many research reactors (RRs) have been in operation since the late 1950s, and a significant amount of spent fuel has accumulated. Most of the spent fuel elements (SFE) were returned to the United States of America and the Latin American countries with concerns related to spent fuel storage are Argentina, Brazil, Chile, Mexico and Peru. The concerns are based on the fact that in May 2006, the option to send SFE to the USA could cease, and national solutions in countries without nuclear power programmes will be very difficult to implement. These concerns were the driving force for the initiation of the IAEA sponsored Regional Technical Cooperation Project for Latin America. The objectives of this Project are to provide the basic conditions to define a regional strategy for managing spent fuel and to provide solutions, taking into consideration the economic and technological realities of the countries involved. In particular, to determine the basic conditions for managing RR spent fuel during operation and interim storage as well as final disposal, and to establish forms of regional cooperation for spent fuel characterization, safety, regulation and public communication.

This Project is divided into four subprojects: (1) spent fuel characterization; (2) safety and regulation; (3) options for spent fuel storage and disposal; and (4) public information and communication. Corrosion surveillance is one of the activities of the subproject entitled Spent Fuel Characterization.

The dominant fuel type used in the Latin American (LA) RRs is plate-type (MTR), LEU, oxide fuel (U_3O_8-Al) clad in Al, followed by TRIGA-type (U-Zr-H) rods. Almost all the spent fuels from LA RRs are stored in racks within the reactor pool, in decay pools or in away-from-reactor wet basins. The in-reactor storage facilities consist of aluminium or stainless steel storage racks.

The main objective of the corrosion surveillance activities of the Regional Project is to evaluate the effect of Latin American spent fuel basin parameters on the corrosion behaviour of research reactor fuel cladding. This evaluation is being done by exposing racks containing aluminium and stainless steel coupons at the different spent fuel basins for different periods followed by their examination. In all corrosion monitoring programmes carried out so far, test racks containing horizontally oriented coupons have been exposed. It is well known that all MTR-type fuel elements are stored vertically in spent fuel basins and the fuel plates are also vertical. Hence, the influence of coupon orientation on the corrosion behaviour of coupons exposed to the spent fuel section of the IEA-R1 research reactor in IPEN, São Paulo, Brazil, is being evaluated. The tests were initiated in 2002 and one set of racks was withdrawn in June 2003 after one year of exposure. The other racks are expected to be withdrawn after two and three years of exposure. This paper presents the influence of coupon orientation on the corrosion behaviour of coupons withdrawn after one year of exposure.

2. METHODS AND MATERIALS

Aluminium alloys AA 1050 and AA 6061 (compositions shown in Table 1) are used in the manufacture of MTR-type fuel plates and elements. Circular test coupons 100 mm in diameter, 3 mm thick and with a central hole 35 mm in diameter were prepared from these alloys under conditions similar to that used to prepare fuel plates. The state of the surface of the coupons was also similar to that of the fuel plates. One machined and polished Al 1050 coupon per rack was rinsed, degreased and passivated in water at 95° C for 24 hours. One of the surfaces of this coupon was scratched with a 0.5 mm wide scriber to simulate a damaged fuel plate surface. The test rack consists of a stand on which the coupons and ceramic disks are stacked. The coupons are stacked as individual coupons or as coupled coupons. Ceramic disks (non-porous alumina) were used to separate the coupons and the coupled coupons, one from the other. The coupons were stacked in the following order, from top to bottom: Al 1050; Al 6061; Al 1050 (pre-oxidized and scratched); Al 1050 - Al 1050 (couple); Al 1050 - Al 6061(couple); Al 6061 - Al 6061(couple); Al 1050 - SS 304 (couple) and Al 6061 - SS 304 (couple).

Six racks were immersed in June 2002 in the spent fuel section of the IAE-R1 RR, of which three were suspended with its coupons oriented horizontally and the other three racks were suspended with its coupons oriented vertically. One rack (no. 44) with vertically oriented coupons and another (no. 47) with horizontally oriented coupons were withdrawn after 12 months of exposure (see Fig. 1). Similar withdrawals will be carried out after two and three years of exposure.

Immediately after withdrawal, the two racks were disassembled and the coupons examined and photographed. The pH of the water in the crevice of the couples was measured. After decontamination, the coupons from the two racks were examined in an optical microscope coupled to an image analysis system. The distribution of corrosion pits on the coupon surfaces was determined. Histograms of number of pits (counts), as a function of pit diameter, were plotted.

FIG. 1. Rack with vertically oriented coupons being withdrawn from the storage section of the reactor pool.

TABLE 1. Nominal composition of aluminium alloys AA 1050 and AA 6061

Alloy	Cu	Mg	Mn	Si	Fe	Ti	Zn	Cr	V	Al
AA 1050	0.05	0.05	0.05	0.25	0.05	0.03	0.05	0.05	0.05	99.5 min
AA 6061	0.15–0.40	0.08–1.20	0.15	0.40–0.80	0.7	0.15	0.25	0.04–0.35		Balance

3. RESULTS AND DISCUSSION

3.1. Basin water parameters

Basin water parameters, such as conductivity, pH and chloride ion concentration, were monitored periodically and radiometric analysis of the water was also carried out. The conductivity was determined directly (but intermittently) by the probe in the de-ionizing circuit. Conductivity was maintained at <2.0 μS/cm, the pH was always in the range of 5.5–6.5 and the chloride ion concentration was <0.2 ppm. When the conductivity came close to 2.0μS/cm or the chloride content came close to 0.2 ppm, the de-ionization resins were regenerated. The water specimens for radiological analysis were collected once a week after the reactor was switched off. Gamma spectroscopy was carried out to determine the nuclides ^{140}Ba, ^{58}Co, ^{60}Co, ^{51}Cr, ^{137}Cs, ^{131}I, ^{133}I, ^{99}Mo, ^{24}Na, ^{239}Np, ^{132}Te and ^{187}W.

3.2. Observations made immediately after removal of the racks

During the disassembly of the racks, it was observed that the coupled coupons were difficult to separate. Also during disassembly of the rack, the pH of the water in the crevice between the various couples was measured. In all cases, independent of coupon orientation, the pH of the water in the crevice was 5.5, one point below that of the bulk water pH (6.5).

3.3. Microscopic examination of coupon surfaces

The exposed surfaces of the two aluminium alloys revealed pits, independent of the orientation of the coupon. However, many features were specific to the alloy, the position of the coupon in the rack and the orientation of the coupon.

3.3.1. Horizontally oriented coupons (Rack 47)

Figures 2(a) and 2(b) reveal the histograms of pit count versus pit size on the upward facing and the downward facing surfaces of the individual AA 1050 coupon respectively. The upward facing surface revealed a large number of pits, ~90, in the size range 40–50 μm while the downwards facing surface revealed only 6–8 pits in the same size range. The shape of the pits on this coupon varied from irregular to round (see Fig. 3).

Most of the pits revealed a bright region around the pit, characteristic of a cathode region around a localized anode region. The shape of this bright region also varied from circular to elliptical (see Fig. 3). On the exposed surfaces of the other AA 1050 coupons, similar round and irregular shaped pits were observed. On the AA 1050 coupon surface, in contact with the AISI 304 stainless steel coupon, large pits were observed, revealing the deleterious effect of a galvanic junction. The two surfaces of the pre-oxidized and scratched AA 1050 coupon revealed few small pits and no bright regions around the pits. No pits were observed along the scratch.

3.3.2. Vertically oriented coupons (Rack 44)

On the surfaces of the individual AA 1050 coupon, round and irregularly shaped pits were observed. Many pit clusters were also observed. The bright cathode areas associated with pits were shaped like a comet with a tail giving a clear indication of the top and bottom of the vertically oriented coupons (see Fig. 4)

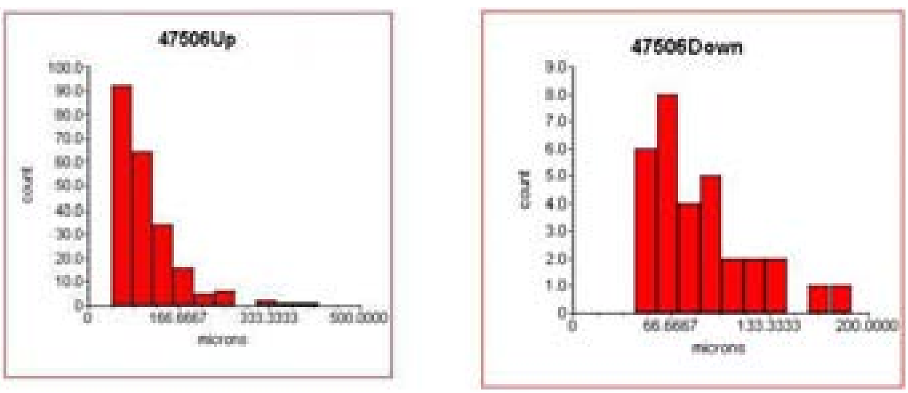


FIG. 2. Histograms of pit count versus pit size (a) on the upward facing (47506Up) and (b) downward facing (47506Down) surfaces on horizontally oriented AA 1050 coupon.

Almost all the pit features and the comet shaped bright areas associated with the circular pits, on AA 1050 were also observed on the exposed surfaces of AA 6061 coupons. Bright regions or comet tails were not observed around irregular shaped pits indicating that the mechanism associated with the formation of round and irregular shaped pits are different. The facing surfaces of the crevice couple coupons, AA 1050-AA 1050, AA 1050-AA 6061 and AA 6061-AA 6061 were stained and did not reveal any pits. The stains on the surfaces of the two alloys were distinct and characteristic of the alloy. The surfaces of the pre-oxidized and scratched AA 1050 coupon revealed a few small pits and no bright regions around the pits. No pits were observed along the scratch..

3.4. Pitting behaviour as a function of coupon orientation

Comparison of pit histograms obtained for the horizontally oriented top surface of AA 1050 with that obtained for one of the surfaces of the same alloy oriented vertically (see Fig. 5) reveals that twice as many pits (size range 40–50 μm) form on the horizontal coupon as compared with that on the vertical coupon.

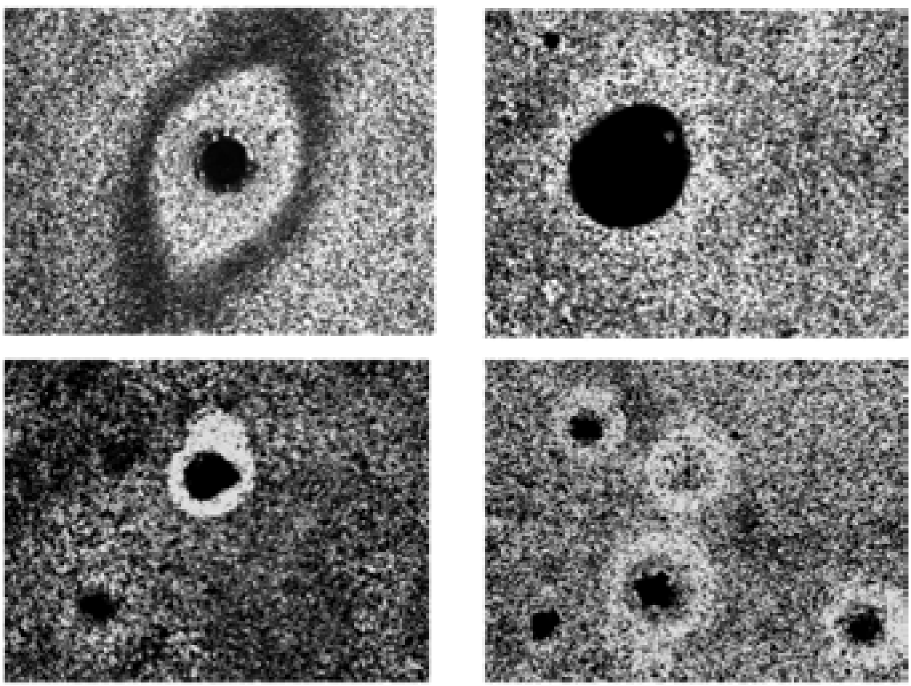


FIG. 3. Micrographs showing pits and bright regions on horizontal AA 1050 surfaces.

Comparison of histograms of the pitted surfaces of vertically and horizontally oriented AA 6061 alloy also revealed similar behaviour. This indicated that among the many parameters that control pit formation, such as alloy composition, metallurgical state and water parameters, settled solids contribute to pit initiation and formation.

4. CONCLUSIONS

(1) The top surface of the horizontally oriented coupons pitted more than the bottom facing surface. The extent of pitting on the top surface of horizontal coupons decreased with change in position of the coupon from top to bottom in the rack.

(2) The two sides of vertically oriented coupons of both alloys pitted to the same extent.

(3) The extent of pitting of vertically oriented coupons was considerably less than that of the horizontally oriented coupons indicating that pit formation is influenced by, among other factors, settled solid particles on the coupon surface.

(4) The pre-oxidized AA 1050 coupon revealed very few pits and these were less than 30 μm in diameter. Both the horizontally and vertically oriented pre-oxidized coupons pitted to a lesser extent than the corresponding un-oxidized alloy coupons.

(5) Coupon orientation had no noticeable effect on crevice or galvanic corrosion. The contact surfaces of AA 1050/AA1050, AA1050/AA6061 and AA 6061/AA 6061 couples and the surfaces of the aluminium alloys

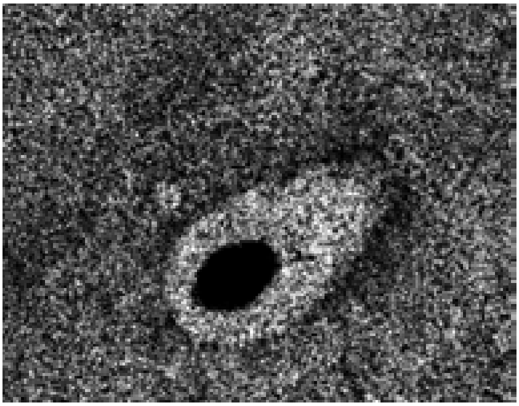


FIG. 4. Optical micrograph of vertically oriented AA 1050 surface revealing comet shaped bright region around a pit.

in contact with AISI 304 stainless steel, were stained with white to grey aluminium oxide.

(6) Overall, the results of this ongoing investigation have indicated that coupon orientation has a marked effect on the corrosion behaviour of aluminium alloy coupons. Since fuel plates of spent MTR-type fuel elements are usually oriented vertically, information that can be obtained from the use of vertically oriented coupons in a surveillance test would be more representative of actual corrosion processes taking place on fuel plates.

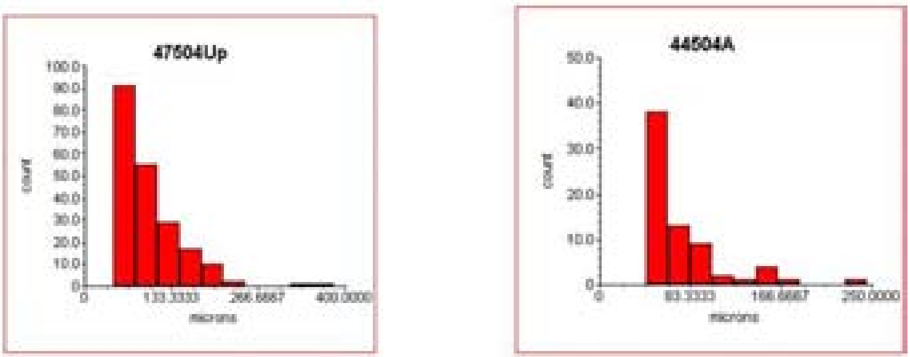


FIG. 5. Histograms of pit count versus pit size on the upward facing surface (47504Up) and vertical surface (44504A) of AA 1050 coupon.

STUDY OF THE EFFECT OF SEDIMENTED PARTICLES ON THE CORROSION BEHAVIOUR OF ALUMINIUM CLAD SPENT FUEL DURING STORAGE IN WATER

S. RODRÍGUEZ, L. LANZANI, A. QUIROGA, R. HADDAD
Comisión Nacional de Energía Atómica (CNEA),
Buenos Aires, Argentina
Email: haddad@cnea.gov.ar

E. SILVA,
Comisión Chilena de Energía Nuclear (CCHEN),
Santiago, Chile

Abstract

Localized attack induced by dust or other particles deposited on aluminium surfaces under water has been addressed as a matter of concern after completion of an IAEA coordinated research project (CRP) on the corrosion of aluminium clad spent fuel during storage in water basins. As the problem can seriously affect the fission product containing the capacity of stored spent fuel even in well maintained high quality water, it is important to learn about the involved mechanisms of attack and find out about the influence of particle composition, in order to establish the convenience or disapproval of the use of materials and procedures in storage basins. With this objective, an experimental approach has been developed to study the mechanism of corrosion attack linked with the deposition of particles of different composition on aluminium surfaces; this includes two kinds of iron flakes and concrete powder. Immersion tests of up to 40 days have been conducted in nuclear grade demineralized water; the position of the sediments was marked and followed through the process and the final state of the aluminium surfaces was assessed by optical and electron microscopy and other microanalysis techniques. Complementary activities were carried on in relation with this work: thorough characterization of intermetallic particles in Al 6061 (appearance, composition, etc.) and study of their electrochemical behaviour. Mg_2Si particles perform very actively, dissolving even in highly pure water at open circuit potential, leaving a small hole on the surface, whereas iron containing intermetallics tends to act as cathodes, producing a local alkaline environment, which gives place to attack on the surrounding oxide film. In the potential range where a hydrogen reaction takes place, a strong cathodic pitting-like corrosion is produced, proportional to the reduction current; at potentials where only the oxygen reduction reaction occurs, this attack is reduced to the environs of the precipitates. A parallel explanation between the microscopic behaviour of the intermetallic precipitates and the effect of sediments is proposed.

1. INTRODUCTION

As a result of a surveillance programme performed in the frame of an IAEA coordinated research project from 1996 to 2000, it was concluded that one of the main factors affecting aluminium clad spent fuel corrosion behaviour during storage in water basins is the deposition of particles of dust or other materials on metal surfaces [1]. As shown in Figs 1 and 2, pits sizing up to 1 mm in length and a few hundred microns in depth were encountered underneath aluminium hydroxide blisters formed on top and/or around these particles. Although their chemical composition is not well known, the two main sources of dirtiness are basin carbon steel components that corrode producing falling iron oxide flakes and masonry powder generated in construction-type work being performed in the surroundings, which includes concrete fine particles, lime, etc. Since a normal cladding thickness of MTR fuel ranges around 150 microns, this phenomena jeopardizes fuel integrity during long term interim storage in water.

There is a strong correlation between the intensity of the attack and the level of water parameters, such as conductivity, ions content, total soluble solids, etc. However, this fact has been verified even in sites with very high quality water, provided it stays stagnant. Since aluminium alloys are known to be very resistant to corrosion in pure water, a thorough study of this degradation process is needed to establish the right conservation conditions.

2. EXPERIMENTAL

The mechanisms of corrosion of aluminium alloy AA 6061 in water was studied by means of electrochemical tests, immersion tests and microscopic

FIG. 1. Pitting on coupon surface after extraction from basin (diameter 10 cm).

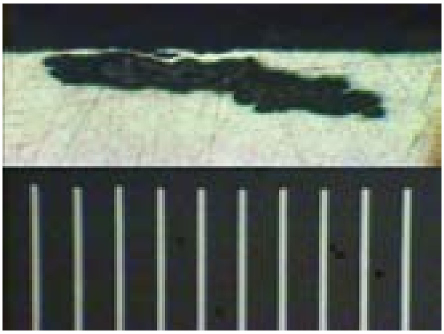


FIG. 2. Cross-section metallography of one pit from coupon of Fig. 1 1DIV=100 μm.

analysis. In order to reproduce some of the results encountered in the basins, the effect of different sediments was considered: two kinds of iron oxide flakes and concrete powder. The role of intermetallic precipitates was assessed based on acomprehensive characterization work followed by a meticulous tracking procedure.

2.1. Aluminium samples

Small rectangles (15 × 25 × 2.5 mm) were cut from lateral positions of un-irradiated fuel plates, out of the meat zone. These plates had been manufactured in the fuel element plant at the Constituyentes Atomic Centre of CNEA as part of the standard production for the RA3 reactor. The chemical composition of the alloy is listed in Table 1. Surface grinding up to grade 2500 was performed, followed by diamond paste of 3 and 1 μm successively. The use of water was avoided during this process, to prevent surface alteration prior to the tests. A thorough precipitate characterization was carried out by means of optical microscopy (OM), scanning electron microscopy (SEM) and energy dispersion spectroscopy (EDS), which permitted to establish their distribution and composition. Tiny marks were produced in some specimens by indentation with a micro-hardness measuring device in order to help locate specific areas on the metal surface and identify individual precipitates along the course of immersion experiments.

2.2. Electrochemical tests

Potentiostatic polarization was performed at different potentials in order to typify the electrochemical behaviour of the different precipitates. In all cases, a 0.5 M sodium citrate was used as supporting electrolyte. The following values were used: -1900, -1700, -1500, -1100, -800 y -300 mV $SO_4^{=}$; also, potential-time measurements were made at open circuit with and without air injection. All potential values are referred to the Hg/Hg_2SO_4 saturated electrode.

2.3. Immersion tests

Seven samples were tested in demineralized water (conductivity = 18.2 $[M.cm]^{-1}$ and pH = 5.5). In all cases, the cells were kept at room temperature (RT). Indentation marks were produced on the surface of samples dedicated to precipitate study, in order to allow for the tracking of individual particles. The specimens were taken out for observation at periodical intervals, totalling an

TABLE 1. CHEMICAL COMPOSITION OF ALUMINIUM ALLOY 6061 USED IN THE TESTS

Element	Content (% wt.)
Mg	0.95–1.10
Si	0.55–0.65
Fe	0.15–0.45
Cu	0.2–0.4
Cr	0.10–0.2
Mn	0.1
Zn	0.25
Ti	0.03–0.07
Al	rest

exposure time of 20 days for the precipitates study in pure water, and 60 days for the sediments study in all solutions.

2.4. Sediments

Particles of different materials were produced in order to deposit them on the samples surfaces. Two kinds of iron oxides were made: oxide A, just by scratching the rusted surface of an iron piece, covered by the common brownish ferric oxide film; oxide B, instead, was generated burning pure iron wire on a flame, which gave place to the formation of a black ferrous/ferric magnetite mixture. Also, commercial cement powder was used in the tests. The cement was let to cure for more than 20 days before applying it onto the specimens. In all cases the particles were suspended on the surface of solutions and immersed using a glass bar, in order to let them fall on the metal surfaces.

2.5. Observation

To assess the surface state of the specimens after each exposure interval, OM, SEM and EDS techniques were used. XY scanning mode of SEM was also useful to determine the topography of attack.

3. RESULTS

3.1. Second phase particles characterization

The following intermetallic precipitates have been identified through OM, SEM and EDS: Mg_2Si, Si, Si-Al-Fe, $FeAl_3$, (Fe,Cr,Mn)Mg3Si_6Al_8(), (Fe,Cr,Mn)$_3SiAl_{12}$() and (Fe,Cr,Mn)$_2Si_2Al_9$(). Also, some phases containing Ti have been found. As an example, Figs 3 and 4 show two types of particles under the light of the OM; the first one bears the typical Mg_2Si compound morphology [2], whereas the second corresponds to any of those containing Fe, mostly and phases, from now on to be referred to as Al-Si-Fe precipitates. Some of those of Mg_2Si have very small Fe containing precipitates inside.

Figure 5 shows the appearance of intermetallics at the SEM, together with the chemical composition inferred from the EDS quantitative analysis, showing the presence of the above mentioned elements.

3.2. Electrochemical performance of precipitates

Figure 6 shows the variation of the Corrosion Potential of the Al 6061 alloy with time, in a neutral 0.5 M sodium citrate solution. After an initial increment, probably due to the natural oxide tendency to grow, which increases passivation, it follows a steep decrease towards negative potentials, indicating that some active dissolution process is taking place.

In potentiostatic tests at -1900 mV the metal surface suffered strong pit-like attack, as shown in Fig. 7, with massive production of H_2 bubbles. However, the second phase particles remained undissolved, even when completely separated from the metal matrix, as seen in Fig. 8. At potentials higher than -1500 mV, the general attack progressively diminishes and Mg_2Si particles start to dissolve, completely disappearing in a matter of hours.

FIG. 3. Mg_2Si precipitates (large grey) 50X.

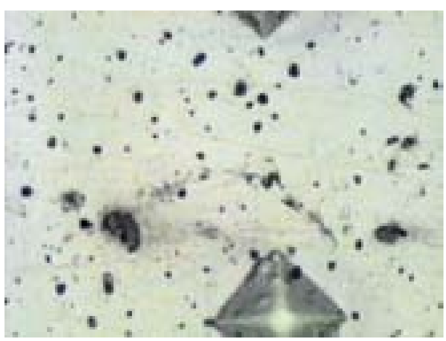


FIG. 4. Particles with Fe (small dark) 50X.

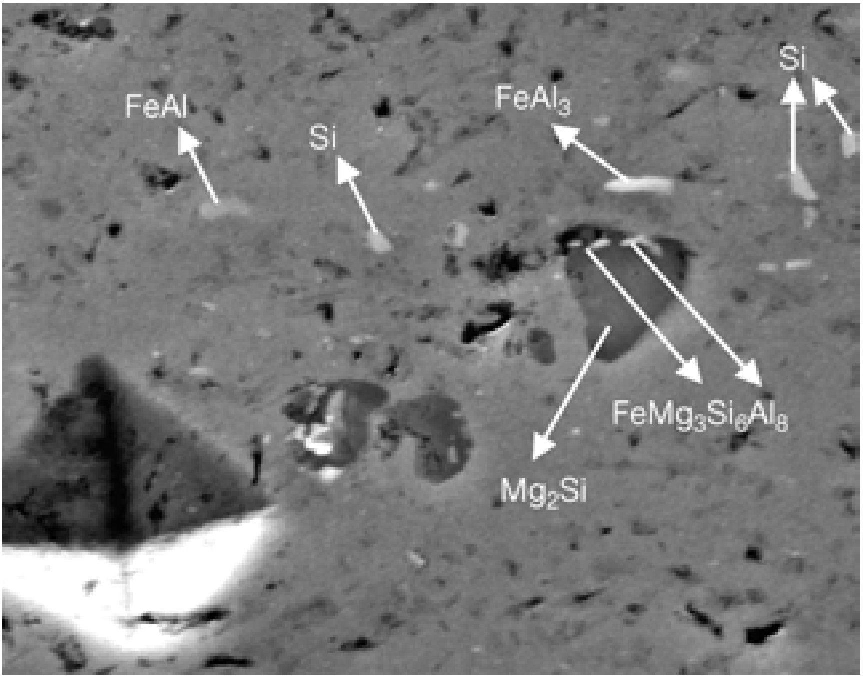


FIG. 5. Composition of some of the intermetallic compounds encountered, as deducted from the EDS quantitative analysis (SEM image X 2080).

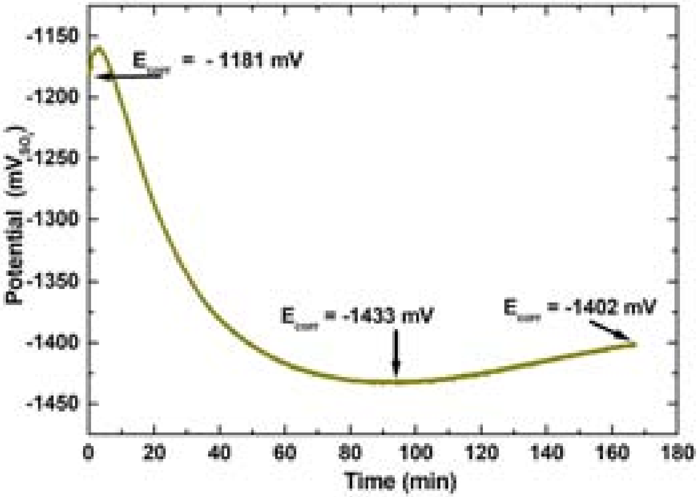


FIG. 6. Evolution of corrosion potential (E_{CORR}) with time.

Instead, attack around Fe containing particles is still observed until the oxygen redox potential is reached; above this value, no further attack have been found.

3.3. Behaviour of precipitates during immersion tests

Figures 9 to 12 show the aspect of four Mg_2Si second phase particles before and after 20 days of immersion in distilled water. The attack began during the first hours; those bearing the small Fe containing precipitate in their interior lasted more (A, C and D), whereas the pure ones (B) finally disappeared. Table 2 shows the measured composition of Mg and Si at the beginning and ending times. It can be seen that a selective dissolution of Mg has taken place. The small precipitates remained unchanged throughout the testing period.

TABLE 2. CHEMICAL COMPOSITION VARIATION OF Mg_2Si PARTICLES DUE TO IMMERSION IN DISTILLED WATER

PARTICLE ID	BEFORE IMMERSION		AFTER 20 IMMERSION DAYS	
	% at. Mg	% at. Si	% at. Mg	% at. Si
A	16.4	18	13.1	14.3
B	11.2	12.73	0	14
C	25.3	20.37	19.5	15.7
D	20.9	13.9	5.1	14.3

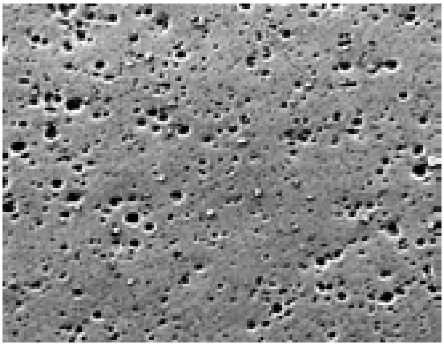


FIG. 7. Surface aspect after constant polarization at -1900mV (SEM X360).

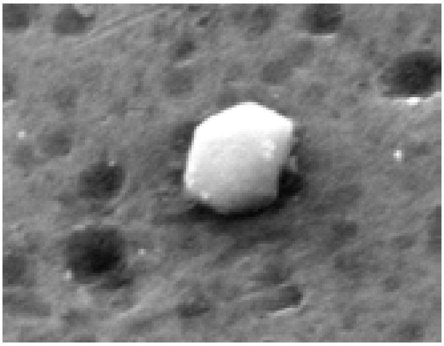


FIG. 8. Mg_2Si particle on the surface (magnification of Fig. 7) (SEM X5746).

FIG. 9. OM picture of Mg_2Si precipitates before immersion.

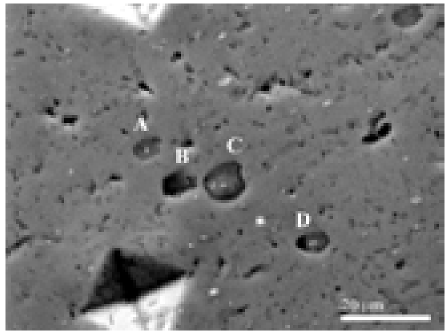


FIG. 10. Particles of Fig. 9 viewed with SEM. The white dots in the middle are Al-Fe-Si phase.

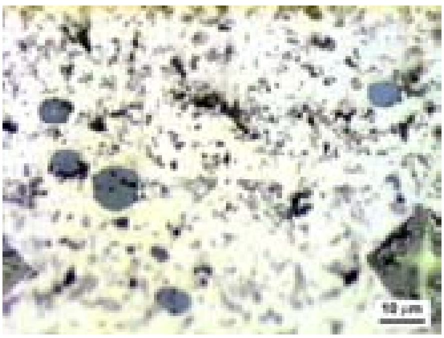


FIG. 11. Aspect of Mg_2Si particles after 469 immersion hours (end of test) (OM).

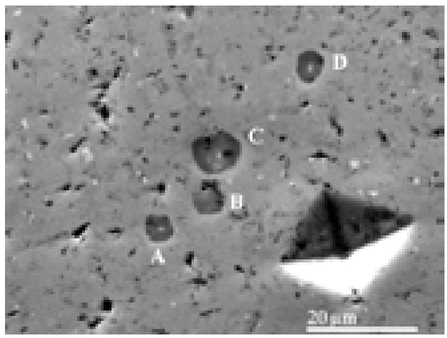


FIG. 12. SEM image corresponding to Fig. 11. Particle B disappeared.

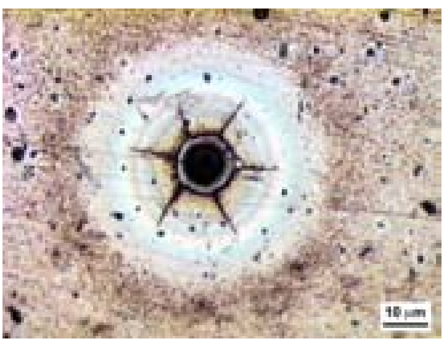


FIG. 13. Oxide growth around an Al-Si-Fe particle after four immersion hours.

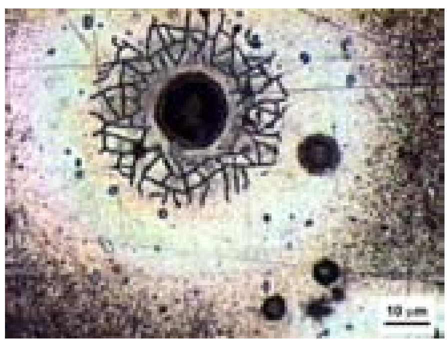


FIG. 14. Aspect of Al-Si-Fe precipitate after 68 hours of exposure to pure water.

Figures 13 to 16 show the change in appearance of one selected Al-Si-Fe particle. After four hours of immersion, a hemispheric attack is produced around it (Fig. 17) and a circular oxide halo is developed. As the process follows, the oxide layer grows thicker and starts to crack, taking a ‘volcano’ appearance, where the crater in the middle holds the precipitate and the corroded region as can be seen in Fig. 18.

3.4. Effect of sediments

The sediment type employed in the tests has strong influence on the corrosion process associated with them.; pH measurements performed before

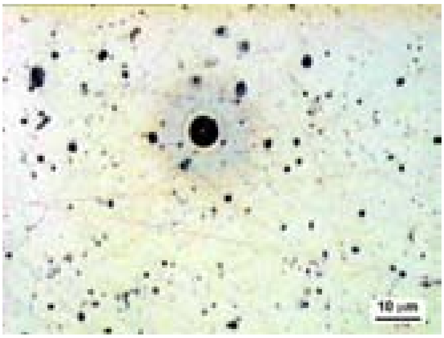


FIG. 15. Al-Si-Fe precipitate after 190 hours of immersion.

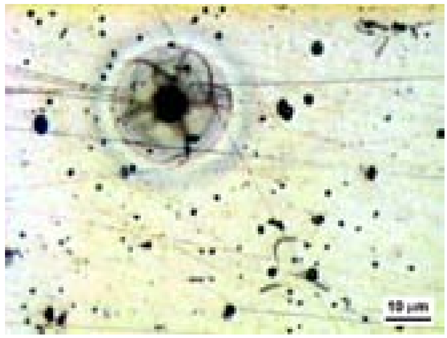


FIG. 16. Final appearance of Al-Si-Fe phase after 15 days of exposure.

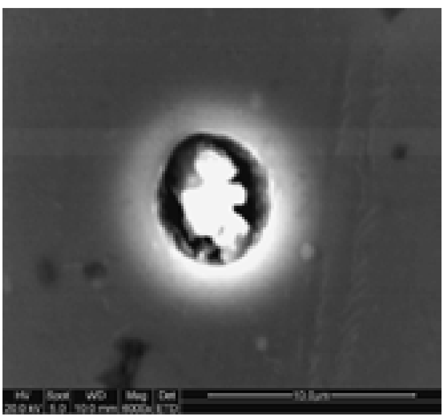


FIG. 17. Sem image corresponding to Fig. 13: attack around the particle, and oxide growth.

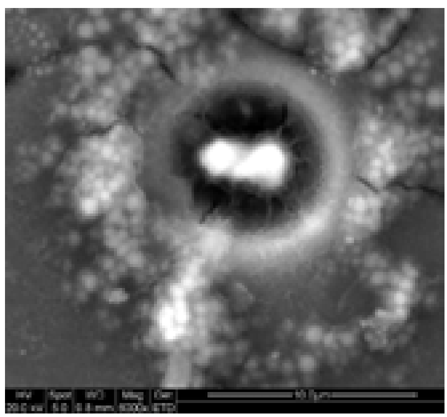


FIG. 18. SEM image corresponding to Fig. 16: enhanced oxide growth.

and after the experiments revealed an increase no greater than 1.8 over the initial value in all cases. This extreme shift corresponds to the addition of cement powder.

Corrosion attack produced by cement particles was evident after two days of immersion. Figure 19 shows the metal surface with seven specks, after 40 exposure days in high purity water. A white corrosion ring surrounds all sediments, some of them turning into brown after some time; a thinner white corrosion product layer covers the remaining surface. At a higher magnification (Fig. 20), it can be seen that the white ring corresponds to a thicker oxide formation, while in the region adjacent to the particles the oxide layer is thinner.

Seven out of 11 flakes of oxide A have behaved in an active manner, forming a white crown around (Fig. 21); the remaining sample surface was covered by a grey-brownish film. Eight days were needed for this corrosion activity to become evident. At high magnification with the SEM, Fig. 22, the fragments of this ferric oxide look like the Fe precipitates of Fig. 18: a surrounding groove and a circular zone of thick cracked oxide.

Unlike the previously described behaviour, only 1 out of 11 chips of oxide B has developed some action on the substrate (Fig. 23). In this case, the oxide grows normally around the particle, which remains embedded, with no separation from the matrix (Fig. 24).

4. DISCUSSION

Mg_2Si intermetallic precipitates dissolve electrochemically at potentials above –1500mV (with respect to the $SO_4^{=}$ reference electrode); as the

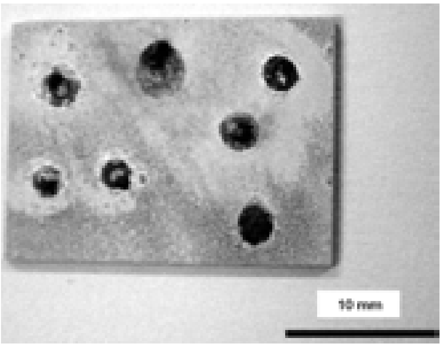


FIG. 19. Cement deposits on the aluminium surface: 40 days of immersion in pure water.

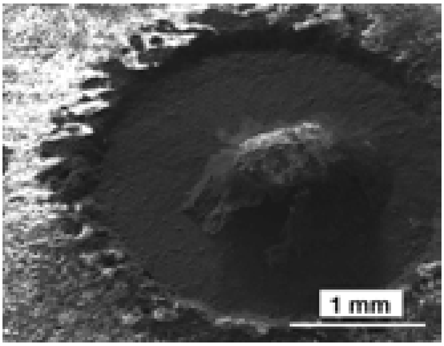


FIG. 20. SEM image showing oxide formation around a small cement bit.

corrosion potential varies in the range of –1500 to –1000 mV, it is quite a consistent result that in the immersion tests at open circuit these particles tend to corrode selectively in pure water, leaving either a hole or a Si rich remain.

Conversely, the iron containing precipitates are stable even at potentials near those where oxygen evolution occurs. These are the only possible sites for cathodic reactions to take place [3–6], when this happens, either hydrogen is produced or oxygen is reduced; in both cases, pH is increased (by proton consumption or OH^- production) in the environs of the precipitate. At high pH values, the aluminium oxide is unstable and dissolves, leaving the metal unprotected. This is the reason why when cathodic reactions are produced by electrochemical means, the surface is attacked around precipitates, producing the observed pit-like attack.

At open circuit potential, the only cathodic reaction is that of oxygen; it is not so strong to produce massive attack on the surface and the attack is limited to the environs of the precipitate. Any time a localized effect (as local

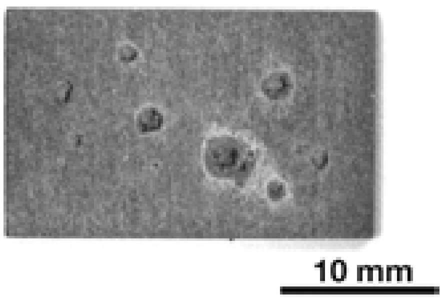


FIG. 21. Oxide A deposits on a sample: after 40 immersion days in pure water.

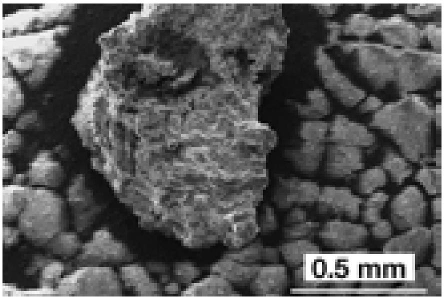


FIG. 22. SEM image corresponding to a small particle of Fig. 21.

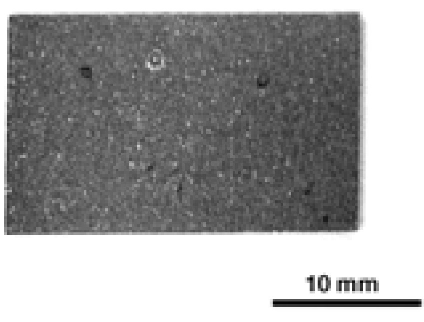


FIG. 23. Oxide B deposits on a sample: after 40 immersion days in pure water.

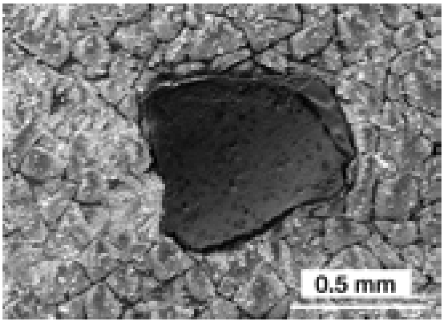


FIG. 24. SEM image corresponding to a small particle of Fig. 23.

incremented pH) generates aluminium dissolution, a crown of thick oxide is formed at some distance from the particle, where the given effect is no longer sustained. This gives place to the 'volcano' look depicted in Figs 15 and 16. It is not uncommon to find Al-Si-Fe precipitates embedded in Mg_2Si particles; they tend to inhibit the dissolution of the latter, probably by increasing the pH, thus stabilizing the protecting Mg oxide. The corrosion effects associated with sediments are strongly dependent of the type of material deposited. The order of aggressiveness encountered is, in decreasing order, cement particles, oxide A and oxide B.

The concrete additions produce two different actions: on one side, they contribute to increase the water pH, which promotes the aluminium corrosion by unstabilizing the protective aluminium oxide film; on the other, a localized attack is produced around the deposited fragments; this latter action could be originated in a mechanism similar to the one described for iron containing precipitates, or it may only be related with the alkaline nature of the product. The influence of ferric oxide additions (oxide A), instead, seems to become manifest in that same way, as revealed by the thick oxide that grows around these particles and the apparent attack on the base metal. Oxide B particles, in contrast, seem to neither affect the oxidation, nor promote metal dissolution. This could be due to a lower capacity of the magnetite of serving as a cathodic reaction site or to a lesser probability of electric contact between the particle and the base metal.

Besides the described sediments related mechanism of attack, it is quite probable that the water trapped underneath the deposits turn into a conductive solution, after the sure dissolution of the Mg_2Si precipitates present in the region covered by them. In such an environment, all the corrosion processes are expected to be more intense. These results may explain some of the features encountered in coupons immersed in basins for different times [1]. Corrosion on aluminium surfaces can occur even in high purity water. It is necessary to avoid the precipitation of dust or other particles onto the fuel elements stored for a prolonged period of time. To completely assess the detrimental consequences, it would be necessary to perform determinations of depth of attack, through cross-section metallography or other suitable techniques. Also, the impact of conductivity and ion content should be studied; these parameters take different values in the various storage sites surveyed; they might strongly influence the corrosion behaviour; the higher the conductivity, the faster all electrochemical reactions; also, chloride ions are known to produce pitting of aluminium alloys.

ACKNOWLEDGEMENT

The authors wish to acknowledge the IAEA for partially supporting this work.

REFERENCES

[1] BENDERSKAYA, O.S., et al., "Corrosion of Research Reactor Al-Clad Spent Fuel in Water" RERTR 2002, Bariloche, Argentina, 4-8 November 2002, paper 10-1.
[2] "Aluminium Alloys", in Metals Handbook, vol. 9 Metallography and Microstructures, ASM, Ohio (1985), pp. 351–388.
[3] SZKLARSKA-SMIALOWSKA, Z., Corrosion Sci. **41** (1999) 1743–1767.
[4] ABALLE, A., BETHENCOURT, M., BOTANA, F., CANO, M., MARCOS, M., Corrosion Sci. **43** (2001) 1657-1674.
[5] BUCHHEIT, R.G., J. Electrochem. Soc. **142** (1995) 3994–3996.
[6] NISANCIOGLU, K., DAVANGER, K., STRANDMYR, O., J. Electrochem. Soc. **137** (1990) 69–77.

BLANK

ENCAPSULATION OF NUCLEAR SPENT FUEL FOR SEMI-DRY STORAGE AT THE BUDAPEST RESEARCH REACTOR

S. TÖZSÉR, T. HARGITAI, I. VIDOVSZKY
KFKI Atomic Energy Research Institute (AEKI),
Budapest, Hungary
Email: tozser@sunserv.kfki.hu

Abstract

In order to change the temporary underwater storage mode of nuclear spent fuel (NSF) from wet to semi-dry as a means of slowing down or even stopping the corrosion effects of the cladding and thereby ensuring safe storage conditions for further temporary storage, AEKI's experts elaborated a canning technology and canning equipment. Manufacturing of the canning equipment and the installation work at the 'away from reactor' (AFR) pond of the Budapest research reactor (BRR) were completed, and the regulatory licence for NSF encapsulation was granted in March 2002. The canning has been running since May 2002. As a first step, a verification programme was performed during which both the technology and the canning equipment were verified and adjusted. Then, during the regular canning programme in 2002, we completed the encapsulation of all our EK-10 fuel (82 fuel assemblies were encapsulated) and at present we are canning the VVR-SM and -M2 type NSF. Until the end of September 2003, 135 single VVR NSF assemblies and 115 triple VVR ones had been encapsulated together with the 82 EK-10 assemblies mentioned. The focus of the paper is less on the wet storage problems of NSF and more on the canning technology and the canning equipment itself. Furthermore, a summary is provided of the experience gained during system commissioning, validation and during the regular canning procedure at the AFR pond of the BRR.

1. INTRODUCTION

The nuclear spent fuel (NSF) assemblies of the Budapest research reactor (BRR) have been stored under water on site in 'at reactor' (AR) and 'away from reactor' (AFR) ponds (the BRR was first commissioned with the AR pond in 1959, the AFR pond was constructed in 1967). Because corrosion signs appeared on the fuel cladding of a few NSF assemblies, in 1998 AEKI's management decided to change the storage mode from wet to semi-dry (or semi-wet), to slow down or even stop the corrosion effects or, in the best cases, avoid their appearance. The encapsulated fuel assemblies (packages) will be kept in the same pool under water but the fuel is maintained in dry conditions, namely, in an inert gas atmosphere thereby ensuring a safe extended temporary storage until the assemblies are transported to their final disposal place. The encapsulation programme relates to all NSF assemblies that have decayed over the last five years.

2. NUCLEAR SPENT FUEL CANNING

The manufacture of the canning equipment was finished in 2001 with a successful factory test[1] and during the last year the installation work and system commissioning was completed at the AFR pond of the BRR. In March 2002, the regulatory body issued the licence to change to semi-dry storage mode [1].

2.1. Canning technology

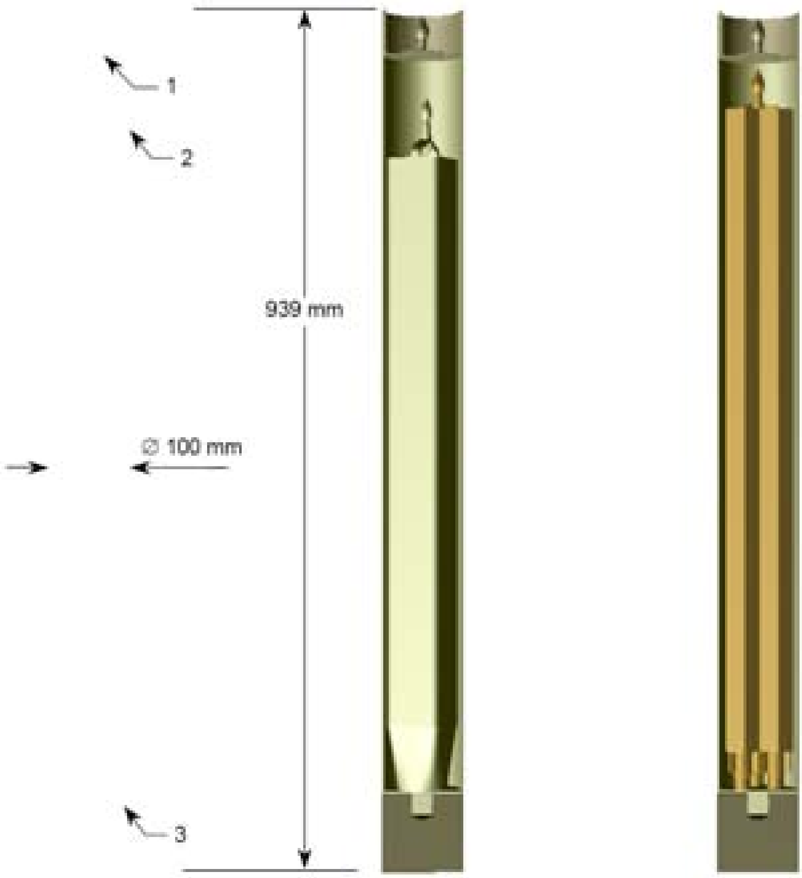
The canning technology (for details see Ref [1]) uses a tube-type capsule as shown in Fig. 1. The capsule is made of aluminium alloy, with a wall thickness of 3 mm. A bottom weight, consisting of an aluminium-clad iron disk is screwed to the bottom of the capsule to ensure sinking of the encapsulated NSF assembly, as well as to provide sub-criticality. The capsule is capable of accommodating one EK-10 or one triple VVR type assembly or three single VVR assemblies (see Fig. 1). The EK-10 assemblies are packaged 'as they are', apart from a small part of the aluminium leg of a VVR assembly (≈73 mm) being cropped off before canning. The overall dimensions of the canning tube are also indicated in the drawing.

Encapsulation utilizes a closed technology during which the capsule undergoes a powerful drying procedure (heated by an eddy current) and it is then filled with 2.5 bar nitrogen gas (the over-pressure is needed for leak detection purposes) after which it is closed by a capsule head with shrink and welded sealing. The main features of the canning technology are the following:

— It ensures intermediate storage for at least a further 50 years;
— It facilitates the canning of EK-10 and both (single and triple) VVR type NSF assemblies;

---

[1] The detailed system design and manufacturing was carried out by Ganz Co. Ltd., Hungary.



(a) EK-10 fuel *(b) VVR fuel.*
(1) Capsule head; (2) tube capsule; (3) bottom weight.

FIG. 1. 3D drawing of the tube construction.

— It ensures easy monitoring after canning (observation of tiny bubbles and/or periodic weight measurements of the tube—which should normally be nearly weightless in water);
— It leaves open all foreseen ways for a final disposal solution including transportation.

In the two photos depicted in Fig. 2, the upper part of the capsule is presented in the two subsequent phases of the canning procedure. Photo 1

FIG. 2. The upper part of the capsule presented in the two subsequent phases of the canning procedure.

shows the capsule head pressed into the canning tube before welding,[2] while the capsule head secured by welding can be seen in Photo 2.

As can be seen in the photos the capsule head is designed to have the same geometry as the head of the NSF assemblies so the same tools can be used for handling the capsules. In the photos in Fig. 2, the identification numbers of the canning tubes can also be seen.

2.2. Canning equipment

The design concept applied for elaborating the canning equipment took into account that the equipment should:

- Fully implement the technology described in Section 2.1;
- Not demand a new working area (canning is required to be carried out in the AFR pond);
- Be compact;[3]

[2] Prior to pressing in the capsule head, the canning tube with the NSF is dried and filled with inert gas. It is also important to mention that the capsule head and the canning tube diameters are made such that there is an overlap of 0.2 mm; and in accordance with the technology, the canning tube is overheated while the capsule head is at room temperature during the pressing. This overlapping means that after an equal-warming (2–3 s) the canning tube will shrink on the capsule head thereby ensuring hermetic sealing for the welding.

[3] It was desirable to avoid any crane manipulation during the canning activities, therefore, a compact construction was preferred instead of a modular one [2].

— Ensure a closed technology (the operator should supervise the canning process and take part in the manipulations of the NSF assemblies only before and after the canning);
— Be able to open any leaking capsules (enabling the canning process to be repeated);
— Be transportable.

The canning equipment comprises the canning unit itself and a cropping machine for cutting the fuel legs if necessary and to ensure that the package can be opened (e.g. in case of faulty encapsulation).

2.2.1. Canning unit

The conceptual layout of the canning unit that implements the technology and fulfils the design requirements described above can be seen in Fig. 3. The layout shows the compact canning container body in its operation position seated in the service hole of the AFR pond.[4] In the drawing, for the sake of better presentation of the complete technology, the technology accessories connected to the compact canning container are also shown.

The central part of the canning unit is the container body made of stainless steel. In the inner part of the container body (in its centre-line) a cylindrical cavity (operation chamber) is constructed to accommodate the capsule during the operation steps of the canning technology. The operation chamber comprises an eddy-current coil, which serves to dry the capsule and the NSF placed into the capsule and to keep them at a given heat. The top and bottom ends of the operation chamber are closed by vacuum-tight sealing. There is a shutter at the bottom end that ensures vacuuming and provides over-pressure inside the operation chamber (see Fig. 3).

The lower part of the container body is submerged to a depth of about 50 cm below the surface of the pond (to provide biological shielding). A transfer pipe is connected to the lower part of the container body, which consists of two sections: the upper one is fixed, the lower one is a vertically movable pipeline. The length of the transfer pipe, as well as the up and down end positions of the movable pipeline are adjusted in such a way that the movable pipe in the down position should settle onto the reception seat of the pond, thereby ensuring a closed sucking-up circuit between the reception seat and the container body.

---

[4] This service hole (∅ 680 mm) accommodates the transfer container used for NSF transportation from the AR pond to the AFR one.

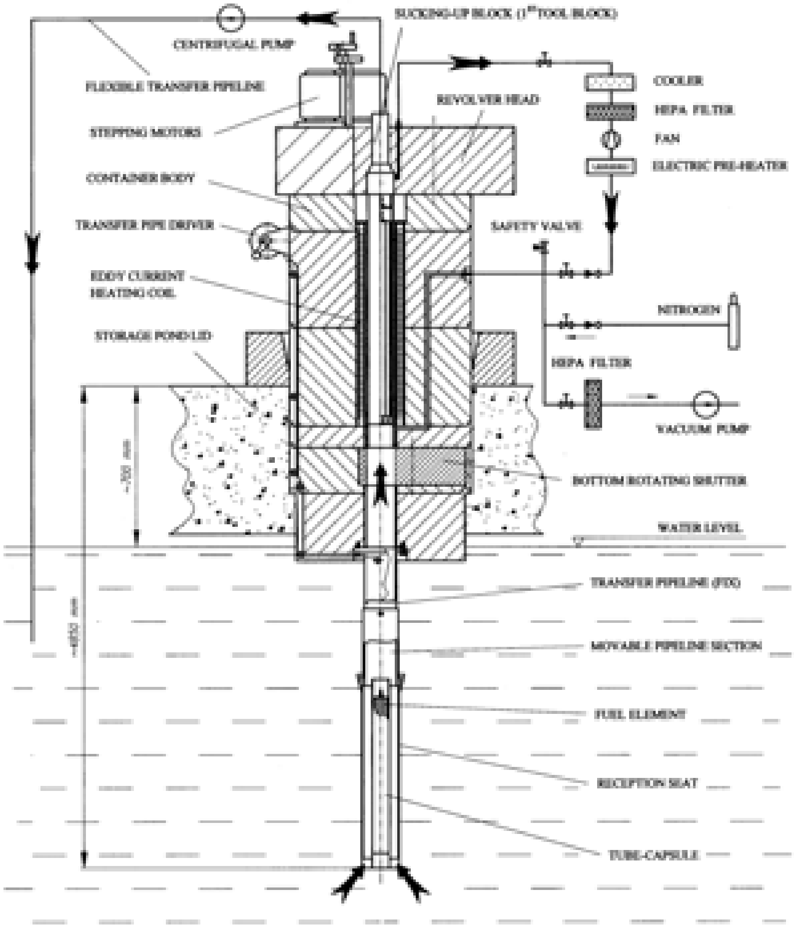


FIG. 3. Conceptual layout of the canning unit.

A rotating head (built on top of the container body), closes the upper part of the operation chamber which, in accordance with the canning sequences, contains five service blocks. The first block is a sucking-up block to transfer the capsule into the operation chamber; the second is a discharge block ensuring removal of the water from the capsule (water discharge is monitored by CCTV); the third one is a drying block whose role is to circulate the hot air in the drying phase of the NSF. The fourth block is for vacuuming and pressing: it creates a vacuum and fills the capsule with nitrogen, as well as pressing the capsule head into the capsule. Finally, the fifth one is a filling-up and welding block, which performs the over-pressurization of the operation chamber and the welding of the capsule head.

A stepping motor actuates (closes and opens) the bottom shutter; two other stepping motors respectively rotate the head and the canning tube inside the operating chamber. The three stepping motors are assembled on top of the container body.

Various accessories of the canning technology are also connected to the container body, such as the flexible transfer pipeline, mentioned previously, with a powerful centrifugal pump that serves to float-up the canning tube; a closed drying air circuit unit with a cooler system for vapour condensation and removal; a HEPA filter; a fan for ensuring air circulation; and a pre-heater of the air before it returns to the operation chamber. Another HEPA filter is installed in front of the vacuum pump in the vacuuming system. The task of this unit is to remove the oxygen content (by vacuuming) before the capsule is filled up with inert gas (the listed units are accommodated on an accessory trolley). In addition, a welding unit and supply systems (electrical power, compressed air, nitrogen and argon bottles) are also connected to the canning container (they are not illustrated in Fig. 3). The units of the canning equipment and the stages of canning are controlled by a PLC system located in a control board. This control board accommodates the manual controls, the LCD display, signalling lamps and a number of indicators.

Figure 4 shows the canning container with its accessories, as installed in the service hole of the AFR pond (the cropping machine is described in Section 2.2.2).

2.2.2. Cropping machine

The canning technology includes a cropping machine: on the one hand, to cut off a small piece from the leg of the VVR fuel assembly, and on the other hand to complete the technology (viz. any faulty encapsulated capsule can be opened by this cropping machine whereupon the whole canning cycle can be repeated).

The cutting unit of the cropping machine is accommodated in an underwater container for collecting the cutting waste. This container is lowered into the AFR pond to the highest position of the reception area (at a depth of about 4 m below the water surface), thereby ensuring safe underwater manipulation. Depending on the type of fuel, there are two different cutting positions in the fuel assembly container (one position is for single VVR fuel, the second position is for triple fuel assemblies) for the leg-cutting process; a third position accommodates the encapsulated capsule to enable the head of faulty capsules

FIG. 4. Canning container.

(1) Container body	*(2) Rotating head*	*(3) Driving motor (for moving the movable pipeline section)*
(4) Stepping motors	*(5) Flexible sucking up pipeline*	*(6) Welding unit*
(7) Cropping machine		

to be cut off. The cutting unit itself is a circular saw driven by an electric motor. The driving motor and the controls of the cropping machine are located above the lid of the pond, as can be seen in Fig. 4.

2.3. Canning procedure

The canning equipment works semi-automatically under computer control (PLC) with operator supervision (acknowledgement between operation phases), or under manual control if necessary, or a mixture of both at the operator's discretion. The whole canning procedure includes a preparation phase, five operation phases, a control phase and a closing phase.[5]

2.3.1. Preparation phase

The capsule is placed in the reception seat and the designated fuel assembly (assemblies) is (are) inserted into it (if they are VVR assemblies the leg-cutting process takes place before this phase).

2.3.2. First operation phase—float up

In automatic mode, the operator starts the canning procedure by pressing a pushbutton. The movable pipeline is lowered and the centrifugal pump starts to generate an intensive water flow. This causes the capsule to float up into the operation chamber, the bottom shutter closes and the pump stops.

2.3.3. Second operation phase—removing the water from the capsule

The rotating head turns to the second position and a small membrane pump removes the water from the capsule.

2.3.4. Third operation phase—drying and maintaining a given heat

The rotating head turns to the third position and the heating starts (switching on the eddy-current heater, the fan and the cooler). In this phase, the fuel within the capsule undergoes a powerful drying and heat maintaining process (130°° C for about 40 min) heated by an eddy current.

2.3.5. Fourth operation phase—vacuuming, filling up with nitrogen and pressing in the capsule head

The rotating head turns to the fourth position, the operation chamber is closed hermetically and by means of vacuuming and nitrogen charge systems, the operation chamber (inside the dried and still hot capsule with the fuel

[5] It is assumed that system adjustment and the obligatory test programme before starting any canning activity are successfully carried out. Before starting (in the preparation phase), the canning unit is in the starting position; in other words, the rotating head is in the first position, the bottom shutter is open, the movable pipeline section is in the upper position.

assembly) is vacuumed, filled, emptied, then again filled with dry nitrogen (vacuum <50 mbar, N_2 overpressure >2.5 bar). When the nitrogen pressure exceeds 2.5 bar for the second time, the cold capsule head is pressed into the hot capsule; because of the overlapping, the capsule head is shrink-fixed in place under 2.5 bar over-pressure).

2.3.6. Fifth operation phase—welding

In the fifth position of the rotating head, the operation chamber is vacuumed, filled with nitrogen and then the capsule head is secured by welding. The welding process is controlled by CCTV. After welding, the vacuum-tight sealing of the operation chamber is released.

2.3.7. Control phase

After the welding process, the rotating head turns to the second position where the welded seam can be inspected by CCTV. After this visual inspection, the rotating head turns back to the first position, the bottom shutter opens and the package floats back to the reception seat; the movable pipeline section then raises to its upper position to enable visual underwater observation (if a capsule is leaking, bubbles can be observed).

2.3.8. Closing phase

If the welded seam is flawless and there is no bubbling, it means that the encapsulation is successful and the package can be placed to its designated position. If the capsule is leaking or has otherwise failed during or after canning, then the capsule head can be removed in the cropping machine and the encapsulation process can be repeated.

In automatic mode, the phases numbered from 1 to 5 are carried out semi-automatically (operator assistance is needed only for control and acknowledgement between phases), while the preparation, control and closing phases are made manually (by underwater manipulation). In case of any malfunction or failure, the operator can switch to manual mode and control the process manually.

With regard to storage after encapsulation, details are given in Ref. [1]. Briefly, the package can be stored underwater (in the same pool). Checking of the packages has essentially been reduced to weight measurements, which should be carried out periodically every 5 to 10 years (obviously, the water quality requires regular checking). The packages can be stored under dry conditions, too (in which case the bottom weight can be removed). An important factor is that canning does not jeopardize the shipment to other storage sites or even back to the country of origin (the overall dimensions of the capsule allow most of the existing transfer containers to be used). But, in any case, the capsules can easily be opened by the cropping machine in order to ship the NSF.

2.4. Technical data

The main technical data of the canning tube and the cycle time of the whole canning procedure are summarized in Table 1.

—

TABLE 1. MAIN TECHNICAL DATA OF THE CANNING TUBE AND THE CYCLE TIME OF CANNING

Overall dimensions of the canning tube:	
— Outer diameter	∅ 100 mm
— Height	939 mm
— Wall thickness	3 mm
Material of the canning tube	AlMgSi alloy
Bottom weight	Steel disc clad with AlMgSi alloy
Fixing of the bottom weight	M30 screwed joint (removable)
Fuel type and quantity that can be canned	(a) 1 EK-10 fuel assembly (b) 3 single VVR fuel assemblies (c) 1 triple VVR fuel assembly
Drying	130° C, with min. 40 min maintaining heat (heated by eddy current)
Vacuuming	<50 mbar (twice)
Filling gas:	
— Quality	Nitrogen T50 (N_2>99.999 %; H_2O<5 ppm)
— Over-pressure	2.5 bar (applied twice)
Closing	Min. 0.2 mm overlapped shrink-fitting, plus welding (min. 2.5 times rotating cycles)
Cycle time of the whole canning procedure	≈120 min (preparation: ≈20 min; 1 to 5 phases: ≈90 min; control and closing phases: ≈10 min)

3. CANNING HISTORY AT THE BRR

In 2001, manufacturing of the canning equipment was completed with a successful factory test (performed by Ganz Co. Ltd., Hungary) and at the beginning of 2002, the installation work was also completed at the AFR pond of the BRR. System commissioning was completed (on-site tests passed with success) and the regulatory licence for changing the storage mode of NSF from wet to semi-dry (i.e. for NSF encapsulation) was granted in March 2002. Regular canning has continued since May 2002.

3.1. System verification

At the beginning, based on the first encapsulation schedule, a strict canning programme was planned (10 capsules/week), but it soon became obvious that this new state-of-the-art compact canning unit was facing some 'teething troubles'. In the first few months, the Q-factor[6] was around 50% only. The majority of faults occurred in the phases of pressing (insufficient capsule head pressing) and welding (defective welding line).

3.1.1. Insufficient pressing

Causes: heating problems (container over-warmed due to the resistance of the eddy-current coil); partial cooling of the upper end of the capsule during the nitrogen filling (due to the local cooling effect of upper gas inlet), too long pressing time (cross-section problems of the pneumatic pipelines).

3.1.2. Defective welding

Causes: sticking of the rotating head (faulty bearings); pulsating over-pressure in the operation chamber (due to periodic blow-off of the safety valve); sticking of the rotating units of the capsule (due to dragging caused by over-heating).

As the failures appeared, in the first phase of regular canning, we modified the canning schedule and commenced a verification programme from the aspect of serial canning in order to make the system capable of safe and effective operation. Together with the manufacturing company`s experts, we eliminated the weaknesses and found the root cause of the defective canning.

---

[6] $Q = T_S/T_T$ in %, where T_S is the number of successful encapsulations; T_T is the total number of encapsulations.

Focusing on safe and faultless canning, we modified some technical solutions, changed the undersized elements and made some modifications on the PLC software, as well. Thanks to these modifications, adjustments and fine tuning, the Q-factor of the canning technology is now better than 90%. We completed the validation programme at the end of November 2002 when the system adequacy was verified with the successful canning of 10 defect-free series.

3.2. Canning record (canning in practice)

The following four figures summarize the statistics illustrating the progress of system validation. In the figures, only the effective months are presented; i.e. when the work was interrupted (in the winter and summer holiday periods), the months are not shown. In Fig. 5, the monthly quantity of the flawless and defective canned fuel assemblies can be seen, while Fig. 6 illustrates how the Q-factor changed during the 12 effective month canning period. As can be seen in Fig. 6, after the validation (from the seventh month), the monthly Q-factor was over 80% and in the last few canning months it exceeded 90%.

Although we completed the validation in the sixth month and in the meantime we encapsulated all of our EK-10 fuel assemblies, the rate of encapsulation increased in a significant way only in the last three months of canning activities. This was because after successful validation, we had to make further repairs and adjustments on the canning container. The change of the rate of canning is well demonstrated in Fig. 7, where the accumulated quantities (both flawless and defective) are illustrated. Figure 8 displays the same progress as a percentage (it can be seen that more than 50% of the total number of cans was encapsulated in the last three months).

Our canning results until the end of September 2003 are that all of our EK-10 fuel assemblies have been encapsulated (this means 82 capsules), and 135 single VVR fuel assemblies and 115 triple VVR assemblies have already been encapsulated in 45 and 115 capsules, respectively. Until that time, there were 82 damaged capsules.

The improvement of the statistics of the canning activities was not solely because of the adjustment and fine tuning of the canning equipment, but the human factor also contributed significantly to a better Q-factor and an increased canning rate. It is obvious that the working procedure also underwent further development. The experience gained during the encapsulation was made use of in adjusting and/or modifying the canning equipment (e.g. timing and setting of the limits of the PLC software), but manual operations, even in the preparation phase, were also modified.

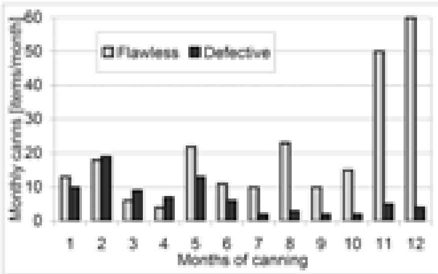


FIG. 5. *Monthly canning performance.*

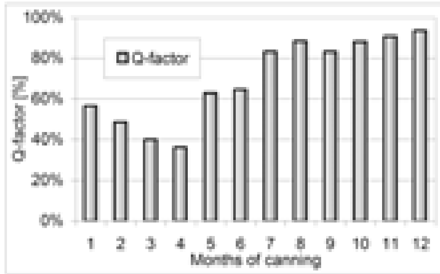


FIG. 6. *Q-factor.*

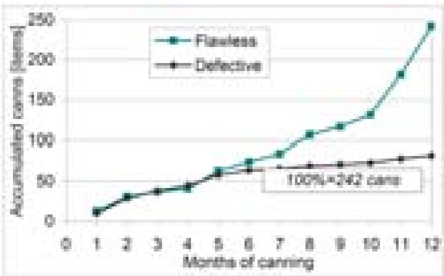


FIG. 6. *Accumulated canning performance.*

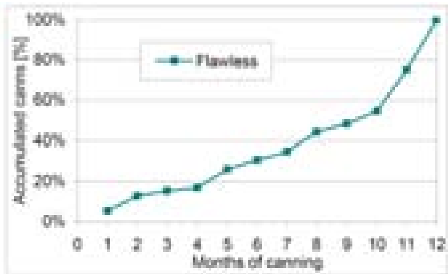


FIG. 8. *Performance pointer.*

Analysis of the faulty packages indicated the importance of the pressing action and the preparation process. The pressing action is very sensitive to the slightest pollution of the pond water; we therefore installed a mechanical water purification system that operates continuously during encapsulation, and we separated the effective canning preparation from the general fuel manipulation in time to avoid stirring up the stagnant water (of course, there is no need to emphasize that if, in spite of these preventive measures, any pollution appears in the pool the canning procedure would be interrupted). The second item emphasized the importance of the preparation process of the capsules and heads, therefore a final inspection of the surface was prescribed (there should be no deformation, cracks or scratches on the contacting surfaces to be shrunk).

Although the capsule head ensured hermetic sealing in most cases by virtue of the shrink bond, several encapsulations were classified as faulty due to defective welding: because the welded seam was not smooth, even and bright, and/or there were spots where the welded seam had an oxide coating.

Defective welding suggested the need to adjust the welding current, the electrode gap and protective gas feeding (viz. we had no experience on protective gas welding under over-pressure); it was also found necessary to eliminate the pulsating over-pressure caused by the safety valve mentioned in Section 3.2 (we installed a nozzle to provide dynamic equilibrium between the protective gas coming in and blowing off while the pressure in the operation chamber develops at constant pressure; the equilibrium pressure ranges around 3.2 bar at the present setting). Apart from the details of the preparation process given above, a link was found between the alkaline etching applied for oxide layer removal and the welding quality.

Operator performance was carefully evaluated. It could be observed that the operators' skills developed: they have become adapted to the practice of canning, and their canning experience is fed back to the job of technical adjustment as well as to their own work. Thanks to this interaction, in the last month the canning performance numbered 60 items, while the Q-factor was around 94%. In this last period, the regular daily quantity was four items/day. During an IAEA safeguards inspection in July 2003, the maximum was six items/day.

At present, encapsulation is continuing. During the summer 2003, we transported a further 180 VVR fuel assemblies from the AR pond to the AFR pond, to give a total of 403 cans after the termination of our present canning programme. Our plan is that this quantity be encapsulated by the end of the year.

3.3. Reflections

The development, manufacturing, system commissioning and even the starting period of regular canning were followed with interest by the Division of Nuclear Fuel Cycle and Waste Technology of the IAEA; then, after system commissioning, our canning activities aroused the interest of the experts of other research reactors in the region.

The IAEA's experts considered the technology and the canning unit itself as a possible storage solution for spent fuel until it can be transferred to the country of origin or some other final solution is found. Similar conclusions were drawn by the experts both of the Radioactive Waste Management in Swierk, Poland, and of the Vinca Institute of Nuclear Science, Belgrade, Serbia and Montenegro, who emphasized that this encapsulation technology may offer a safe and effective solution to extend the safe storage time of NSF located at their storage facilities. They also highlighted the fact that the canning process does not jeopardize any shipment in the future.

4. SUMMARY

The main aim of the paper was to validate both the canning technology and the safe and reliable operation of the canning equipment. It is concluded that, in accordance with the design requirements, the canning cask and its accessories, including the cropping machine, form a compact and mobile technology that ensures an almost completely automatic, safe and reliable encapsulation procedure. The cropping machine, as an integral element of the unit, makes the technology complete, ensuring the handling of any defective closed cans. Not only can the capsule accommodate EK-10 and VVR type NSF assemblies for semi-dry storage and ensure further safe and long-term temporary storage in the same wet conditions where the fuel assemblies were previously stored (i.e. in the same pool), it does not jeopardize the shipment of spent fuel back to the country of origin for reprocessing or to any other storage site—even to dry storage.

With the detailed presentation of the system validation, we aimed to introduce the final phase of the technical improvement, as well as to present the experience gained during regular encapsulation. We have shown how enhanced usability together with the reduced error rate led to an improved in-service performance and how the operator assistance progressed. Having used the opportunity of final validation, one part of the experience was utilized in realizing the canning unit, the other one was integrated into the improvement of canning conditions including the human factor.

So far as the human factor is concerned, it seems—as might be expected—to represent a significant risk factor to the final results. Although the canning process is more or less automatic, it is essential that the human factor be taken into consideration in the encapsulation procedure. It is of paramount importance that the operators be fully familiar with the system control and all steps of manipulation. The operators have to form a well trained and shaken down group. Three operators form an optimum team to carry out the canning.

The technology is very sensitive to the careful preparation of the capsule and capsule head (clean and undamaged surfaces) on the one hand, and on the other hand to the cleanliness of the pool water where the canning takes place. These factors should be considered as an important prerequisite for safe and effective canning. During the modifications, it was observed that the canning cask does not become contaminated to any great extent, so it can be decontaminated for transport and the whole system can be loaded easily on a standard truck.

On the basis of these results, it is concluded that the experience gained during the system commissioning, validation and the encapsulation of

242 capsules, demonstrates that the canning equipment can be operated safely and effectively. It is also thought that both the canning technology and the equipment itself provide a reliable solution to the spent fuel storage problems of other research and training reactors.

Although we already have applied a great deal of experience, there are further valuable suggestions that should be taken into consideration in relation to improving the canning technology. Decisions on further improvement will be taken after completing our encapsulation programme, based on the demand for other canning jobs.

ACKNOWLEDGEMENT

In the paper's research and development, staff members of the BRR, experts of Ganz Co. Ltd., and Festo Budapest Co. Ltd. were involved. The authors are grateful to all those who actively participated or who offered constructive advice. In particular, the authors would like to thank K. Késmárky, A. Lvei, L. Szabolcs (members of the operators' team); K. Mesterházy, S. Horányi (staff members of the BRR); I. Porubcsánszky, J. Nagy (Ganz Co. Ltd.); and A. Hanusovszky (Festo Budapest Co. Ltd.) for their contribution.

REFERENCES

[1] HARGITAI, T., VIDOVSZKY, I., "New storage mode for spent fuel at the Budapest research reactor", Proc. 6th Int. Topical Mtg on Research Reactor Fuel Management—Transactions, Ghent, Belgium, 2002, ENS RRFM, (2002), 109.

[2] GADO, J., MARIY, A., TOZSER, S., KHATTAB, M., New Spent Fuel Storage and Encapsulation Facility in Working report submitted to the Egyptian Atomic Energy Authority, Budapest–Inshass (November 2001).

BLANK

CORROSION OF ALUMINIUM-CLAD SPENT FUEL AT RA RESEARCH REACTOR

M. PEŠIC, T. MAKSIN, R. DOBRIJEVIC, Z. IDJAKOVIC
Vinca Institute of Nuclear Sciences,
Belgrade, Serbia and Montenegro
Email: mpesic@vin.bg.ac.yu

Abstract

Almost 95% of all spent fuel elements of the RA research reactor in the Vinca Institute of Nuclear Sciences, Belgrade, Serbia and Montenegro, are stored in 30 aluminium barrels and about 300 stainless steel channel-holders in the temporary spent fuel storage water pool. The first activities of sludge and water samples, taken from the pool, were measured in 1996–1997 and were followed by analysis of chemical composition of samples. Visual inspections of fuel elements in some stainless steel tubes and of the fuel channels stored in the reactor core have shown that some deposits cover aluminium cladding. Stains and surface discolouration are noted on many of the spent fuel elements that were examined visually during the core unloading and inspections carried out in 1979–1984. Some of the water samples, taken from pool, about a 150 stainless steel tubes and 16 barrels have shown very high ^{137}Cs activity compared with low activity measured in pool water. It was concluded that aluminium cladding of the fuel elements was penetrated due to the corrosion process. A study on the influence of water corrosion processes in the RA reactor storage pool was started within the framework of the IAEA coordinated research project entitled Corrosion of Research Reactor Aluminium-Clad Spent Fuel in Water in 2002. The first test rack with various aluminium and stainless steel coupons, supplied by the IAEA, was immersed in the pool already in 1996. New racks were immersed in 2002 and 2003. The rack immersed in 1996 was taken out from the pool in 2002 and the rack immersed in 2002 was taken out in 2003. Results of the examination of these racks carried out according to the strategy and the protocol, proposed by the IAEA, are described in the paper.

1. INTRODUCTION

The fuel elements used in 6.5 MW heavy water RA research reactor in the Vinca Institute from 1959 to 1984 are known as the Russian TVR-S fuel type. The TVR-S fuel element is a cylinder about 11.3 cm long with 3.72 cm outer diameter (see Fig. 1) produced either in the Elekhtrostal plant near Moscow (2% LEU metal fuel) or in the Novosibirsk Chemical Concentrates Plant (80% HEU-oxide fuel dispersed in aluminium matrix). The RA reactor operated from 1959 to 1976 using LEU fuel elements and from 1976 to 1984 using HEU fuel elements.

The tube-type fuel layer of the TVR-S element has an average length of 100 mm and inner/outer diameter of 31/35 mm. The mass of ^{235}U nuclide in the fuel element is 7.4 g in the case of the LEU fuel and 7.7 g in the case of the HEU fuel. The fuel layer is cladded on the inner and outer side by 1 mm thick aluminium. An inner tube (the 'expeller') designed from aluminium within the fuel element serves to adjust coolant flow rate. The top and bottom of the slug are covered by the aluminium 'stars' (each 3 mm thick) connected to the expeller tube. The aluminium, used in the TVR-S fuel elements, is known as the Russian SAV-1 alloy (0.985 weight fraction of aluminium). The main impurities in the alloy are magnesium and silicon. The content of impurities with high absorption cross sections for neutrons (boron and cadmium) are very low. The volume of the TVR-S element is measured as (58 ± 2) cm^3. The volume of the SAV-1 material, excluding the fuel layer, is estimated at 40 cm^3. The area of the SAV-1 surfaces in contact with the water is calculated at 420 cm^2.

The temporary storage water pool (see Fig. 2), consisting of four interconnected basins and an annex to the fourth basin, store almost 95% of all spent fuel elements of the RA reactor. Each basin is covered by a thin carbon iron lid plates and can be closed, i.e. isolated from the other ones, using a door, manufactured from carbon iron. Thick concrete walls and the bottom of the pool are lined by 1 cm thick stainless steel plate. The pool is filled with about 200 m^3 of stagnant ordinary tap water. It is connected by a special underground water transfer channel to the reactor body. Tap water is added to the pool once per year to replace an amount of the water, lost due to natural evaporation.

FIG. 1. TVR-S fuel element.

In the beginning of the reactor operation, the LEU spent fuel elements were stored in the original stainless steel channel-type containers (SSC) filled with de-mineralized water and immersed in the water of the basins. There was a plan to transfer the spent fuel elements back to the former USSR after four to five years of their cooling, but it did not happen. All 304 SSC became filled by spent fuel elements after a few years of the reactor's operation. In order to increase the storage capacity, new aluminium containers ('barrels') were designed. Each barrel could be filled with a maximum of 180 spent fuel elements. From the beginning of the 1960s until 1984, about 5000 of the oldest LEU fuel elements were repackaged from the stainless steel tubes in 30 sealed barrels. The barrels are stored in two rows in the water of the annex to basin no. 4. About 1600 LEU spent fuel elements and about 900 HEU spent fuel elements remained in the stainless steel tubes until the present. Drained RA reactor core still contains 480 HEU spent fuel elements, since 1984. Design of the TVR-S fuel elements was very reliable. Only one LEU fuel element (from a total of 6656) failed in the core during 18 years of the RA reactor operation. The fuel channel containing the damaged fuel element was identified quickly, and replaced in the core by a new one. The failed fuel element was stored in the specially labelled SSC in the spent fuel storage pool.

The elements stored in the barrel are covered with de-mineralized water. Before the barrel was sealed, thin plates of cadmium were inserted in the water in the barrel with the aim to ensure the nuclear sub-criticality. The exact

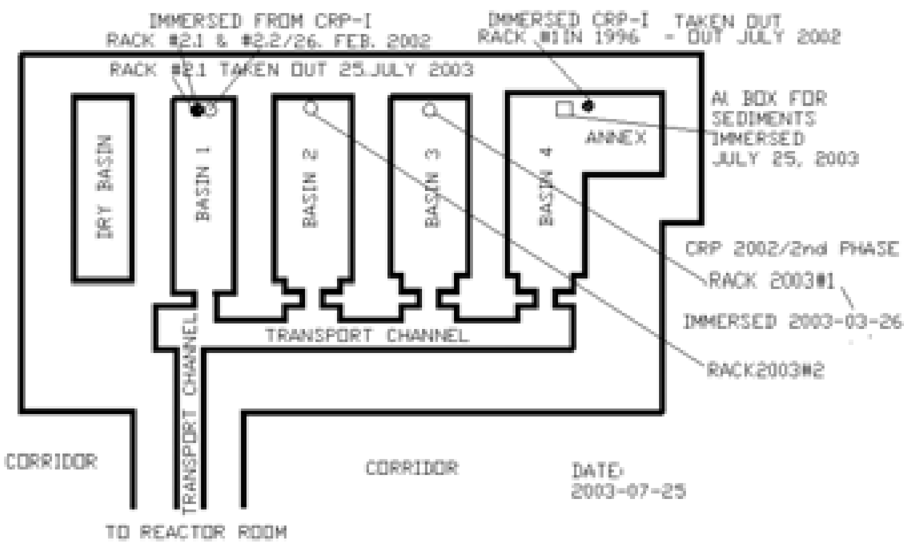


FIG. 2. Sketch of the RA reactor spent fuel storage pool.

composition of aluminium, used for construction of the barrels, is not known. It is believed that this aluminium was produced in former Yugoslav factories in the 1960s. Any difference in composition of this aluminium, compared with the SAV-1, could increase a corrosion rate, because the galvanic couples are created. Also, the galvanic couple could be created between aluminium and cadmium, and increase a corrosion rate of both materials within the barrels. Even more, there are the same assumptions that corrosion rate was so intensive that the cadmium plates were practically completely dissolved by now.

The first report on bad water quality in the spent fuel storage pool can be found as early as 1962, with a proposal for mechanical purification of the water, but no actions were taken. In 1984, for the first time, activity of ^{137}Cs nuclide was measured in the pool water, but no actions were taken, unfortunately, until the mid-1990s. The first systematic monitoring of the pool water parameters was started in 1996–1997. The activities of the samples of the sludge and the water, taken from the spent fuel pool, were measured and were followed by the analysis of the chemical composition of samples. Analyses have shown that the pool water is highly corrosive [1, 2] to aluminium alloys. Visual inspections of the fuel elements in several stainless steel tubes and in the fuel channels stored in the reactor core have shown that deposits cover aluminium cladding of the fuel elements. Stains and surface discolouration were present on many of the spent fuel elements that were examined visually during the core unloading and inspections, carried out in 1979–1984.

Serious actions for remedies of the water in the spent fuel storage pool of the RA reactor are under way since 1997. Debris and about 3 m^3 of the sludge were removed from the basins. Many corroded elements were removed, but some large ones remained in the pool. Further activities in the spent fuel pool will include removing the carbon iron structure from basin no. 4, removing the rest of the sludge and corrosion deposits using hydro-monitor, pool water purification and regular monitoring of the water chemistry, and ^{137}Cs activity in the pool.

2. RADIOACTIVITY AND CHEMISTRY OF POOL WATER

The typical average activity of pool water in 2002 and 2003 is (90 ± 9) Bq/mL, originated dominantly from ^{137}Cs nuclide [3]. Activity of ^{60}Co nuclide in the pool water is under the detectable limit (1 mBq/mL). Presence of the ^{137}Cs nuclide in the pool water is a consequence of the cladding failure of few fuel elements and damaged walls of the storage aluminium barrels, due to the corrosion process. A few water samples, taken from several stainless steel tubes and barrels have shown very high ^{137}Cs specific activity (2 kBq/mL–400 kBq/

mL) compared with the low ^{137}Cs specific activity (50 Bq/mL–400 Bq/mL), measured in the samples taken from the other barrels and from the pool water. Results of measuring specific activity of ^{137}Cs nuclide in the basins of the spent fuel pool of the reactor RA, from 1995 to 2003, are given in Fig. 3.

Activities of the sludge samples were measured in 1996–1997 by coaxial germanium gamma-ray spectrometers in the Vinca Institute and in the IAEA laboratories. The chemical composition of two sludge samples was determined in the radiochemical laboratories of the IAEA [2]. It was shown that the main component of the sludge is Fe_2O_3 (about 83% by weight) that, beside aluminium oxides, gives the dark red–brown colour to the sludge. Visual inspections of TVR-S fuel elements in several SSC and in the fuel channels stored in the reactor core have shown that deposits (determined lately to be mainly aluminium-hydroxide) cover aluminium cladding of the fuel elements. Stains and surface discolouration are present on many of the spent fuel elements that were examined visually during the core unloading and inspections carried out in 1979–1984. Creation of these deposits, stains and surface discolouration was attributed to the poor chemical parameters and to the reduced flow rate of the heavy water—the primary coolant in the RA reactor core.

Chemical parameters of the water samples taken from the basins of the storage pool were made a few times per year since 1997 [4, 5], and approximately once per month, since mid-2001. The water samples were analysed for the main parameters related to the corrosion process. Typical values of the basins water parameters, obtained during 2002 and 2003, are given in Table 1.

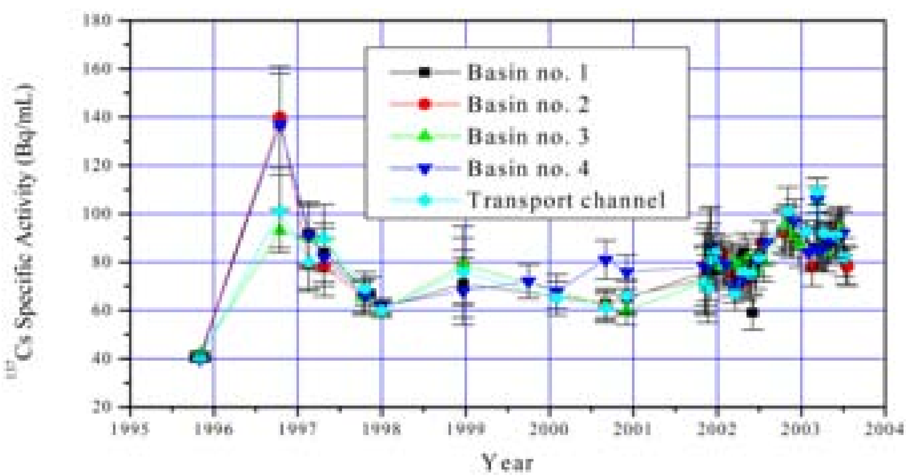


FIG. 3. Cs-137 activity in the pool water.

TABLE 1. Chemical parameters of the water in the basins

Water parameter	Range in 2002–2003	Average(mean) value	Standard deviation of the mean
pH factor	7.31–8.39	7.53	0.12
El. conductivity (μS/cm)	360–570	451.3	27.6
Fe ions (mg/L)	0.07–0.21	0.13	0.01
Cu ions (mg/L)	<0.01	<0.01	
Al ions (mg/L)	<0.01	<0.01	
Chlorides, Cl (mg/L)	65–85	73.0	0.9
Sulphate ions, SO_4 (mg/L)	31–70	48.6	1.5
Nitrate ions, NO_3 (mg/L)	<0.05	<0.05	
Hardness, dH	6.5–7.5	6.9	0.1
Temperature (°C)	14–28		

From this data, it is obvious that water parameters significantly contribute to the development of the corrosion process in aluminium. Up to now, only mechanical purification of the water was induced by operation, during working hours, the pump with 25 μm filter papers.

3. A STUDY ON CORROSION OF ALUMINIUM CLADDING

An initial study on the influence of the water corrosion process at the SAV-1 aluminium cladding of the TVR-S type of enriched uranium spent fuel elements in the storage water pool of the RA research reactor was started within the framework of the IAEA coordinated research project (CRP) Corrosion of Research Reactor Aluminium-Clad Spent Fuel in Water, II phase in 2002 [6]. But, the first test rack with various aluminium and stainless steel coupons, supplied by the IAEA during the first phase of the CRP, was immersed in the pool water already in 1996. This rack, labelled Rack no. 1, was taken out from the pool in July 2002, after approximately six years of exposition time to the influence of the water from the pool. The other two racks from the first phase of the CRP, labelled Rack no. 2.1 and Rack no. 2.2 were immersed in the pool water in February 2002. Rack no. 2.1 was taken out from the pool in July 2003, after approximately 16 months of exposition time. At the same time, a cylindrical aluminium pot is immersed in the pool water of the basin no. 4 (see Fig. 2) at a depth of about 1 m above the basin's bottom, with the aim to collect the deposits in the next three months, i.e. the pot will be taken out at the end of October 2003.

Two racks, received in the second phase of the CRP, were immersed in the pool water in March 2003 and labelled Rack2003 no. 1 and Rack2003 no. 2, respectively. We are planning to take out Rack no. 2.2 from the pool in February 2004, after 24 months of the exposition time. Rack2003 no. 1 will be taken out in September 2003, after six months of exposition time. Rack2003 no. 2 will be taken out from the pool after one year of the exposition time, in March 2004. In the second phase of the CRP, it is also planned that the assembling parts of Rack2003 no. 1 and new coupons made from aluminium and stainless steel will be used to assemble the new rack (labelled Rack2003 no. 1R) that will be immersed in the pool water in spring 2004 and exposed for another year.

Results of the examination of Rack no. 1 and Rack no. 2.1, carried out according to the strategy and the protocol proposed by the IAEA, are described in this paper.

Visual examination of Rack no. 1 (Fig. 4) has shown that the front (top) sides of the coupons are covered by a considerable amount of dark red sludge from the pool water, collected over the six years of exposition time. It was clearly that the corrosion process effects could not be seen without cleaning

FIG. 4. Rack no. 1 withdrawn from the pool water, after six years of exposition time.

surfaces of the coupons. Also, the white deposits were noted at the front sides of a few coupons. It was also noted that there were no deposits at the bottom (back) sides of the coupons, and the effects of the corrosion process could be seen easily. Pitting, as the main localized form of the corrosion of aluminium in the water basins, was also noted at the surface of all coupons. Some black spots were seen at the front sides of the coupons. It is believed that these spots are made of aluminium-oxide. Oxides of different shades of grey were observed, too.

Immediately after the withdrawal of Rack no. 1 from the pool, the measured pH factor of wet coupons' surfaces was about 7. Gamma ray dose equivalent rate, measured near the surface of the coupons, was 3.5 μSv/h. Rack no. 1 was placed in a glass beaker and covered by the pool water from July 2002 to January 2003, when the first examinations were started. Two coupons on the top of the rack were dry, so the pH was not possible to measure. Values of pH on the external surfaces of the other coupons were in the range of 4.5–6.0. The pH factor of the bulk water in the glass beaker was about 7.0. Rack no. 1 was disassembled and coupons were removed from the rack according to the IAEA test protocol. The dark red sludge and the white deposits were removed carefully from the coupons using soft wood plate and collected for further chemical and activity analysis. The coupons were cleaned in 5% phosphoric acid solution according to the IAEA protocol.

A metalographical study on the influence of corrosion on the surface of the aluminium coupons was made using a microscope with 10 times and 20 times magnification. Many pits and effects of crevice corrosion are noted (Fig. 5).

Visual examination of Rack no. 2.1 (Fig. 6) has shown that the top sides of the coupons are covered by a dark red sludge from the pool water, collected for only one year of exposition time. The white deposits were noted at the front sides of a few coupons, too. It was also noted that there were no deposits at the back sides of the coupons. Some black spots were seen at the front sides of the coupons. It is believed that these spots are made of aluminium-oxide. Oxides of different shades of grey were observed, too. Immediately after the withdrawal of Rack no. 2.1 from the pool, the measured pH factor of wet coupons' surfaces

FIG. 5. Various corrosion effects on aluminium coupons of Rack no. 1.

were in the range of 4.5–6.0. Rack no. 2.1 was disassembled and coupons were removed from the rack according to the IAEA test protocol. The dark red sludge and the white deposits were removed carefully from the coupons using soft wood plate and collected for further chemical and activity analysis. The coupons were cleaned in 5% phosphoric acid solution according to the IAEA protocol. A metalographical study on the influence of corrosion at the surface of the aluminium coupons will be made soon using a microscope with 10 times and 20 times magnification.

4. CONCLUSION

Corrosion of aluminium-clad spent nuclear fuel in water of the spent storage pool of the RA research reactor is studied in the Vinca Institute of Nuclear Sciences as part of the IAEA CRP. The information related to the spent nuclear fuel and the storage pool, including results of monitoring of chemistry and radioactivity of the pool water, are given. Data on the corrosion process of aluminium, obtained from an analysis of Rack no. 1 (after six years of exposition time) and Rack no. 2.1 (after one year of exposure time) are shown. Effects of the corrosion process are confirmed at all aluminium coupons. Further analyses of the racks with the aluminium coupons will be made in the near future.

FIG. 6. Rack no. 2.1 withdrawn from the pool water, after one year of exposure time.

ACKNOWLEDGEMENT

This study is supported by the Ministry of Science, Technologies and Development of the Republic of Serbia through project no. 1958/2002 entitled Transport Processes of Particles in Fission and Fusion Systems and the IAEA through the CRP YUG-11968/RBF Corrosion of Aluminium-Clad Spent Fuel in the Research Reactor RA Storage Pool.

REFERENCES

[1] KOPENI, M., MATAUŠEK, M.V., VUKADIN, Z., MAKSIN, T., "Corrosion Problems in the Research Reactor RA Spent Fuel Storage Pool", NATO ARW Microbial Degradation Processes in Radioactive Waste Repository and in Nuclear Fuel Storage Area, Budapest (1996).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Mission Report to Vina Institute in February 1997, Appendix II, IAEA, Vienna (1997).

[3] MATAUŠEK, M.V., VUKADIN, Z., PAVLOVI, S., MAKSIN, T., IDJAKOVI, Z., MARINKOVI, N., "Detection of fission products release in the research reactor RA spent fuel storage pool", Proc. Annual Mtg Nuclear Technology, Aachen, Germany (1997) pp. 415–418.

[4] MAKSIN, T., DOBRIJEVI, R.P., IDJAKOVI, Z.E., PEŠI, M.P., "Corrosion of research reactor aluminium-clad spent fuel in water: Chemical and microbiological influence", Int. Conf. 4th Yugoslav Nuclear Society YUNSC 2002, Belgrade (30 September–4 October 2002), Book of Abstracts, Vina Bulletin **7** supplement 2 (September 2002) 27 (Proc. on CD ROM, to be issued in 2003).

[5] IDJAKOVI, Z., MILONJI, S., UPI, S., Quality of water from the pool, original containers and aluminium drums used for storage of spent nuclear fuel, Nukl. Tehnol. **XVI** 2 (2001) 46.

[6] PEŠIC, M., MAKSIN, T., DOBRIJEVI, R., IDJAKOVI, Z., "Study on influence on corrosion at aluminium-clad spent fuel of research reactor", ENS 7th Int. Topical Mtg on Research Reactor Fuel Management (Aix-en-Provence, France 2003), ENS RRFM 2003, Transaction, pp. 240–244.

CHAIRPERSONS OF SESSIONS

Opening Session		
Session 1	J.P. BOOGARD	Netherlands
	J.R. GRANADA	Argentina
	S. MOROZOV	Russian Federation
	H. ABOU YEHIA	France
Session 2	H. BÖCK	Austria
	R.N. SAXENA	Brazil
	M. RAMANADHAM	India
Session 3	M. PESIC	Serbia and Montenegro
	C.S.B. PIANI	South Africa
Session 4	J. KLEIN	Chile

CO-CHAIRPERSONS OF SESSIONS

Session 1	T. HARGITAI	IAEA
Session 2	N. RAMAMOORTHY	IAEA
Session 3	R. BURCL	IAEA
Session 4	I. RITCHIE	IAEA

SECRETARIAT OF THE CONFERENCE

R. BURCL	Scientific Secretary
T. HARGITAI	Scientific Secretary
S. PARANJPE	Scientific Secretary
N. RAMAMOORTHY	Scientific Secretary
I. RITCHIE	Scientific Secretary
R. MUÑOZ	Conference Coordinator (Chile)
H. SCHMID	Conference Coordinator
V. SAGEISCHEK-KAEGI	Secretarial Support
F. FLAKUS	Scientific Editor (IAEA)

P. ADELFANG	Scientific Editor (IAEA)
G.V. RAMESH	Coordinating Editor (IAEA)
S. CLEMENTS	Proceedings Editor

HOST GOVERNMENT LIAISON OFFICER

G. TORRES OVIEDO, Chile

PROGRAMME COMMITTEE

H. BÖCK	Austria
I. RAMANATHAN	Brazil
G. TORRES OVIEDO	Chile
J. KLEIN	Chile
H. ABOU YEHIA	France
S.K. SIKKA	India
CH. PIANI	South Africa
R. BURCL	IAEA
T. HARGITAI	IAEA
S. PARANJPE	IAEA
I. RITCHIE	IAEA

BLANK

LIST OF PARTICIPANTS

Abou Yehia, H.	Institut de radioprotection et de sûreté nucléaire (IRSN), B.P. 17, F-92265 Fontenay-aux-Roses, France Fax: +33146575412 Email: hassan.abouyehia@irsn.fr
Alten, S.	Department of Nuclear Safety, Turkish Atomic Energy Authority, Eskisehir Yolu – Lodumlu, 06530 Ankara, Turkey Fax: +903122859055 Email: salten@taek.gov.tr
Anastasova, E.	Institute for Nuclear Research and Nuclear Energy, Bulgarian Academy of Sciences, 72, Tzarigradsko Chaussee, 1784 Sofia, Bulgaria Fax: +35929743955 Email: apos@inrne.bas.bg
Apostolov, T.	Institute for Nuclear Research and Nuclear Energy, Bulgarian Academy of Sciences, 72, Tzarigradsko Chaussee, 1784 Sofia, Bulgaria Fax: +35929743955 Email: apos@inrne.bas.bg
Barnea, Y.	Israel Atomic Energy Commission, 24, Chaim Levanon Street, P.O. Box 7061, 61070 Tel Aviv, Israel Fax: +97236462539 Email: ybarnea@tau.ac.il
Barón, J.	Cuyo National University, Ayacucho 666, 5500 Mendoza, Argentina Fax: +542614380120 Email: jbaron@uncu.edu.ar

Blaumann, H.	Centro Atómico Bariloche, Comisión Nacional de Energía Atómica (CNEA), Avenida Bustillo, Km 9,500, 8400 S.C. Bariloche, Rio Negro, Argentina Fax: +542944445225 Email: blaumann@cab.cnea.gov.ar
Boeck, H.	Atominstitut der Österreichischen Universitäten, Stadionallee 2, A-1020 Vienna, Austria Fax: +4315880114199 Email: boeck@ati.ac.at
Bolshinsky, I.	U.S. Department of Energy, NN-42, 1000 Independence Avenue, S.W., Washington, DC 20585, USA Fax: +12025866789 Email: igor.bolshinsky@aul.gov
Bond, D.	Imperial College Reactor Centre, Silwood Park Campus, Buckhurst Road, Ascot, SL5 7TE, United Kingdom Fax: +4402075942272 Email: dave.bond@imperial.ac.uk
Boogaard, J.P.	Quality, Safety and Environment, NRG (Nuclear Research & Consultancy Group), P.O. Box 25, NL-1755 ZG Petten, Netherlands Fax: +31224568490 Email: boogaard@nrg-nl.com
Briso, H.A.	Comisión Chilena de Energía Nuclear, Amunategui 95, Casilla188-D, Santiago, Chile Fax: +5623646300 Email: hbriso@cchen.cl
Burcl, R.	Division of Nuclear Fuel Cycle and Waste Technology, International Atomic Energy Agency, P.O. Box 100, A-1400 Vienna, Austria Fax: +43126007 Email: r.burcl@iaea.org

Bustamante, M.	Comisión Chilena de Energía Nuclear, Amunategui 95, Casilla188-D, Santiago, Chile Fax: +5626554000 Email: mbustama@cchen.cl
Calderon, D.	Comisión Chilena de Energía Nuclear, Amunategui 95, Casilla188-D, Santiago, Chile Fax: +5623646277 Email: dcaldero@cchen.cl
Calzetta Larrieu, O.	Centro Atómico Bariloche, Comisión Nacional de Energía Atómica (CNEA), Avenida Bustillo, Km 9,500, 8400 S.C. Bariloche, Río Negro, Argentina Fax: +542944445225 Email: calzetta@cab.cnea.gov.ar
Chávez, J.	Comisión Chilena de Energía Nuclear, Amunategui 95, Casilla188-D, Santiago, Chile Fax: +5623554331 Email: jcchavez@cchen.cl
Ciocanescu, M.	Institute for Nuclear Research Pitesti, P.O. Box 78, RO-0300 Arges, Romania Fax: +4048262449 Email: ciocanescum@lycos.com
Conde, M.	EDLOW Internnational Company, 1666 Connecticut Avenue, N.W. Suite 201, Washington, DC 20009, USA Fax: +12024834840 Email: mconde@edlow.com
Cornejo, P.	Servico Nacional de Geologíca y Minería, SERNAGEOMÍN, Chile Fax: +56223853332 Email: jefe_lab@sernageomin.cl

Daie, J. | Comisión Chilena de Energía Nuclear,
Amunategui 95,
Casilla188-D,
Santiago, Chile
Fax: +5623646277
Email: jdaie@cchen.cl

Dalle, H.M. | Centro de Desenvolvimento da Technologia Nuclear (CDTN/CNEN),
Caixa Postal 941,
Cidade Universitária – Pampulha,
BR-30123-970 Belo Horizonte/MG, Brazil
Fax: +553134993411
Email: dallehm@urano.cdtn.br

De Wid, T.A. | US Embassy in Chile,
Av. Andres Bello 2800,
Las Condes,
Santiago, Chile
Fax: +562 3303118
Email: dewidta@state.gov

Dedik, P. | U.S. Department of Energy, NN-42,
1000 Independence Avenue, S.W.,
Washington, DC 20585, USA
Fax: +12025866789
Email: trisha.dedik@hq.doe.gov

Dillich, J.C. | Australian Nuclear Science and Technology Organisation (ANSTO),
New Illawarra Road, Lucas Heights,
PMB Menai NSW 2234, Australia
Fax: 61 2 9717 9263
Email: jcd@ansto.gov.au

Dody, A. | Nuclear Research Centre (NRCN),
P.O. Box 9001,
Beer Sheva 84190, Israel
Fax: +97286567690
Email: dodik@bgumail.bgu.ac.il

Dragolici, A. C. National Institute of R&D for Physics and Nuclear Engineering,
"Horia Hulubei",
P.O. Box MG – 6,
RO-79617 Bucharest- Magurele, Romania
Fax: +40214574440
Email: adrag@ifin.nipne.ro

Du Bruyn, J.F. South African Nuclear Energy Corporation,
Safari-1 Research Reactor, Building P-1800,
P.O. Box 582,
0001 Pretoria, South Africa
Fax: +27123055956
Email: safarjfb@necsa.co.za

Ellis, R.J. NSTD Oak Ridge National Laboratory,
P.O. Box 2008,
1 Bethel Balley Road,
Oak Ridge, Tennessee 37831 6172, USA
Fax: +18655749619
Email: ellisrj@ornll.gov

El-Morshdy, S.E. ETRR-2,
Nuclear Research Centre,
Atomic Energy Authority,
Cairo 13759, Egypt
Fax: +2024691740
Email: slmorshedy@etrrz-aea.org.eg

Estryk, D.S. Comisión Nacional de Energía Atómica (CNEA),
Instrumentacón y Control,
González y Aragón 15, Centro Atomica Ezeiza,
Buenos Aires 13180 ZA YA, Argentina
Fax: +541167798433
Email: daestryk@cnea.gov.ar

Etchepareborda, A. Centro Atómico Bariloche,
Comisión Nacional de Energía Atómica (CNEA),
Avenida Bustillo, Km 9,500,
8400 S.C. Bariloche, Río Negro, Argentina
Fax: +542944445257
Email: etche@cab.cnea.gov.ar

Eybert-Prudhomme, C. | Direction générale de la sûreté nucléaire et de la radioprotection,
BP 83,
F-92266 Fontenay-aux-Roses, France
Fax: +33143197166
Email: cecile.eybert- prudhomme@asn.minefi.gouv.fr

Farley, D.R. | EPRI Worldwide,
Level 9 Edobori Center Building,
2-1-1 Edobori, Nishi-ku,
Osaka 550-0002, Japan
Fax: +81662251111
Email: dfarley@epriww.com

Ferraz, W.B. | CNEN/CDTN,
Rua Professor Mario Werneck s/n, Cidade Universitária,
P.O. Box 941, Minas Gerais,
BR-30123-970 Belo Horizonte, Brazil
Fax: +553134993390
Email: ferrazw@cdtn.br

Finschi, E. | Comisión Chilena de Energía Nuclear,
Av. Nueva Bilbao 12.501,
Las Condes, Chile
Fax: +5623646263
Email: efinschi@cchen.cl

Freud, R. | Nuclear Research Centre-Negev (NRCN),
Israel Atomic Energy Commission,
P.O. Box 9001,
Beer Sheva 84190, Israel
Fax: +97286567077

Gambarini, G. | Department of Physics of the University of Milan,
Via Celoria, 16,
I-20133 Milan, Italy
Fax: +390250317630
Email: grazia.gambarini@mi.infn.it

Garonis, O. | Comisión Nacional de Energía Atómica de Argentina (CNEN),
Avenida General Paz 1499,
San Martin, Prov. de Buenos Aires, Argentina
Fax: 54 11 6772 7296
Email: garonis@cnea.gov.ar

Gobrecht, K.H.	28, Place Louis Jouvet, F-38100 Grenoble, France Fax: +13602378323 Email: klaus@ottosix.com
Gonçalves, I.F.	Instituto Tecnologico e Nuclear Phys. Dep., Estrada Nacional 10, P-2686-953 Sacavém, Portugal Fax: +351219941525 Email: ifg@itn.mces.pt
Graffunder, H.W.	Friedrichstrasse 28, D-76297 Stutensee, Germany Fax: +4972494568 Email: graffunder@nuklide.de
Granada, J.R.	Centro Atómico Bariloche, Comisión Nacional de Energía Atómica (CNEA), Avenida Bustillo, Km 9,500, 8400 S.C. Bariloche, Rio Negro, Argentina Fax: +542944445298 Email: granada@cnea.gov.ar
Green, M.	Nuclear Research Centre (NRCN), P.O. Box 9001, 84190 Beer-Sheva, Israel Fax: +97286567210
Gyurkócza, C.	Budapest University of Technology and Economics, Institute of Nuclear Techniques, Müegyetem Rakpart3-9, H-1111 Budapest, Hungary Fax: +3614631954 Email: csabagy@reak.bme.hu
Haddad, A.	Comisión Nacional de Energía Atómica (CNEA), Avenida del Libertador 8250, 1429 Buenos Aires, Argentina Fax: +541143798433 Email: haddad@cnea.gov.ar
Hainoun, A.	Atomic Energy Commission of Syria (AECS), P.O. Box 6091, Damascus, Syrian Arab Republic Fax: +963116112289 Email: ahainoun@aec.org.sy; A.Hainoun@gmx.net

Hansen, W.	Technische Universitaet Dresden, Institut für Energietechnik Ausbildungskernreaktor, Mommsenstrasse 13, D-01062 Dresden, Germany Fax: +4935146333831 Email: hansen@metrs1.mw.tu-dresden.de
Harari, R.	Nuclear Research Centre-Negev (NRCN), Israel Atomic Energy Commission, P.O. Box 9001, Beer Sheva 84190, Israel Fax: +97286567077 Email: rharari@zahav.net.il
Hargitai, T.	Division of Nuclear Installation Safety, International Atomic Energy Agency, P.O. Box 100, A-1400 Vienna, Austria Fax: +43126007 Email: t.hargitai@iaea.org
Henriquez, C.H.	Comisión Chilena de Energía Nuclear, Amunategui 95, Casilla188-D, Santiago, Chile Fax: +5623646277 Email: chenriqu@cchen.cl
Hergenreder, D.F.	INVAP S.E., F. P. Moreno 1089, P.O. Box 961, 8400 S.C. Bariloche, Rio Negro, Argentina Fax: +542944423053 Email: hergend@invap.com.ar
Hillebrand, I.	Forschungszentrum Karlsruhe GmbH, Hermann-von-Helmholtz Platz 1, D-76344 Eggenstein-Leopoldshafen, Germany Fax: +497247828282 Email: iris.hillebrand@knk.fzk.de
Hoffmann, W.	Intacta S.A., Antupiren 7988, Parque Violeta Cousiña, Peñalolen, Santiago, Chile Fax: +5622794763 Email: enmaco17@hotmail.com

Hojman, R.	Comisión Chilena de Energía Nuclear, Amunategui 95, Casilla188-D, Santiago, Chile Fax: +5624702598 Email: rhojman@cchen.cl
Howard, L.	Centronic House, King Henry's Drive, New Addington, Croydon, CR9 0BG, United Kingdom Fax: +441689808005 Email: lhoward@centronic.co.uk
Ionita, I.	Institute for Nuclear Research Pitesti, P.O. Box 078, 0300 Pitesti, Romania Fax: +4024826449 Email: ionionita@lycos.com; onu@scn.ro
Israr, M.	Pakistan Institute of Nuclear Science and Technology, CSO, HPD Pinstech, P.O. Nilore, Islamabad 45650, Pakistan Fax: +92519290275 Email: israr@pinstech.org.pk
Iturrieta, L.	Comisión Chilena de Energía Nuclear, Amunategui 95, Casilla188-D, Santiago, Chile Fax: +5623646277 Email: liturrieta@cchen.cl
Joppen, F.L.	Centre d'étude de l'énergie nucléaire (SCK-CEN), Boeretang 200, B-2400 Mol, Belgium Fax: +3214320513 Email: fjoppen@sckcen.be
Jova Sed, L.	Department of Nuclear Safety and Security, International Atomic Energy Agency, P.O. Box 100, A-1400 Vienna, Austria Fax: +43126007 Email: l.a.jova-sed@iaea.org

Karameldin, A.	Nuclear Research Centre, Reactor Department, Atomic Energy Authority, P.O. Box 13759, Inshas, Cairo, Egypt Fax: +2024620778 Email: alykarameldin@hotmail.com
Kennedy, G.G.	Ecole Polytechnique de Montréal, P.O. Box 6079, Downtown Montreal, Quebec, H3C 3A7, Canada Fax: +15143404192 Email: greg.kennedy@polymtl.ca
Kharpate, A.V.	CIRUS Research Reactor, Maintenance Division, Bhabha Atomic Research Centre (BARC), Trombay, Mumbai 400 085, India Fax: +912225505311 Email: avk@apsara.barc.ernet.in
Klein, J.K.	Comisión Chilena de Energía Nuclear, Amunategui 95, Santiago, Chile Fax: +5623646277 Email: jklein@cchen.cl
Koonen, E.	Centre d'étude de l'énergie nucléaire (SCK-CEN), BR2 Department, Boeretang 200, B-2400 Mol, Belgium Fax: +3214320513 Email: ekoonen@sckcen.be
Krass, A.	US Department of State, 2201 C Street NW, Washington, DC 20520, USA Fax: +12026470775 Email: krassal@t.state.gov
Kuk, Il-Hiun	Korea Atomic Energy Research Institute, P.O. Box 105, Yuseong-gu, Daejeon 305-600, Republic of Korea Fax: +82428625055 Email: ihkuk@kaeri.re.kr

Lahodova, Z.	Nuclear Research Institute, CZ-250 68 Husinec Rez, Czech Republic Fax: +420266172045 Email: laz@ujv.cz
Lamas, C.	Comisión Chilena de Energía Nuclear, Amunategui 95, Casilla188-D, Santiago, Chile Fax: +5623646277 Email: clamas@cchen.cl
Lavérie, M.J.	Commissariat à l'énergie atomique (CEA) – Saclay, DSNQ/MSN - Bâtiment 482, F-91191 Gif-sur-Yvette, France Fax: +33140561594 Email: michel_laverie@cea.fr
Lecot, C.A.	INVAP S.E., F. P. Moreno 1089, P.O. Box 961, 8400 S.C. Bariloche, Rio Negro, Argentina Fax: +542944423053 Email: lecot@invap.com.ar
Leszczynski, F.	Centro Atómico Bariloche, Comisión Nacional de Energía Atómica (CNEA), Avenida Bustillo, Km 9,500, 8400 S.C. Bariloche, Rio Negro, Argentina Fax: +542944445225 Email: lesinki@cab.cnea.gov.ar
Lipka, J.	Faculty of Electrical Engineering and Information Technology, Slovak University of Technology, Ilkovicova 3, SK-81219 Bratislava, Slovakia Fax: +421265427207 Email: lipka@elf.stuba.sk
Livne, Z.	Israel Atomic Energy Commission, Nuclear Research Centre (NRCN), P.O. Box 9001, 84190 Beer Sheva, Israel Email: zvilivne@vision.net.il

Loncomilla, M.	Comisión Chilena de Energía Nuclear, Amunategui 95, Casilla188-D, Santiago, Chile Fax: +5623646277 Email: moli@vtr.net
Longhino, J.M.	Centro Atómico Bariloche, Comisión Nacional de Energía Atómica (CNEA), Avenida Bustillo, Km 9,500, 8400 S.C. Bariloche, Rio Negro, Argentina Fax: +542944445225 Email: longhino@cab.cnea.gov.ar
Loose, V.W.	Los Alamos National Laboratory, P.O. Box 1663, MS F 604, Los Alamos, USA Fax: +5056675125 Email: loose@lanl.gov; vlsilver@msn.com
Macnab, D.	ARPANSA, P.O. Box 655, Miranda, NSW 1490, Australia Fax: +61295458348 Email: don.macnab@health.gov.au
Mahlers, Y.P.	Institute for Nuclear Research, Prospect Nauky 47, Kiev 03680, Ukraine Fax: +380442655105 Email: mahlers@gluk.org
Marques, J.G.	Instituto Tecnologico e Nuclear Phys. Dep., Estrada nacional 10, P-2686-953 Sacavém, Portugal Fax: +351219941039 Email: jmarques@itn.mces.pt
Matejka, K.	Czech Technical University, Faculty of Nuclear Sciences & Physical Engineering, V Holesovickach 2, CZ-180 00 Praha 8, Czech Republic Fax: +420284680764 Email: matej@troja.fjfi.cvut.cz

Matthews, S.	Servicio Nacional de Geologíca y Minería, Av. Santa María 0104, Providencia, Chile Fax: +5622385332 Email: smatthews@sernageomin.cl
Mazrou, H.	Commissariat à l'énergie atomique (COMENA), Centre de recherche nucléaire d'Alger, 2, Boulevard Frantz-Fanon, B.P. 399 16000 Algiers, Algeria Fax: +213434280 Email: mazrou_h@comena-dz.org; mazrou_h@yahoo.it
Mazufri, C.M.	INVAP S.E., F. P. Moreno 1089, P.O. Box 961, 8400 S.C. Bariloche, Rio Negro, Argentina Fax: +542944423053 Email: mazufri@invap.com.ar
Mazzi, R.	INVAP S.E., F. P. Moreno 1089, (8400) S.C. Bariloche, Rio Negro, Argentina Fax: +542944423051 Email: mazzi@invap.com.ar
Mazzocchi, V. L.	Instituto de Pesquisas Energétics e Nucleares (IPEN-CNEN/SP), C.P. 11049 Pinheiros, 05422-970 Sao Paulo SP, Brazil Fax: +55118169188 Email: vlmazzo@curiango.ipen.br
Medel, J.	Comisión Chilena de Energía Nuclear, Amunategui 95, Casilla188-D, Santiago, Chile Fax: +5623646277 Email: jmedel@cchen.cl
Mikhalevich, A.	Joint Institute of Power and Nuclear Research "Sosny", National Academy of Sciences, 99, Krasin Street, 220109 Minsk, Belarus Fax: +375172260698 Email: alex.mikhal@sosny.bas-net.by

Mikulski, A.T.	National Atomic Energy Agency, ul. Krucza 36, PL-00-522 Warsaw, Poland Fax: +48226959846 Email: mikulski@paa.gov.pl
Morozov, S.I.	Federal Nuclear and Radiation Safety Authority of Russia (RF Gosatomnadzor), Taganskaya Street 34, RU-109147 Moscow, Russian Federation Fax: +70959124041 Email: msi@gan.ru
Muñoz, R.	Comisión Chilena de Energía Nuclear, Amunategui 95, Casilla188-D, Santiago, Región Metropolitana, Chile Fax: +56 24702568 Email: rmunoz@cchen.cl
Murua, C.A.	Reactor Nuclear RA-0, Pabellón Ingeniería, Ciudad Universitaria, 5000 Córdoba, Argentina Fax: +543514334429 Email: cmurua@com.uncor.edu
Nadal, H.M.	Compagnie pour l'Etude et la Réalisation de Combustibles (CERCA), Tour Areva, 92084 Paris la Défense, France Fax: 33 1 47 96 58 92 Email: helios.nadal@ framatome-anp.com
Neves, C.F.C.	CNEN/CDTN, Rua Professor Mario Werneck, s/n Cidade Universitária, P.O. Box 941, Minas Gerais, BR-30123-970 Belo Horizonte, Brazil Fax: +553134993319 Email: caf@cdtn.br
Notari, C.	Universidad Technológica Nacional, Buenos Aires Faculty, Comisión Nacional de Energía Atómica (CNEA), Medrano 951 - Capital Federal, Argentina Fax: +54147988255 Email: cnotari@mail.retina.ar

Novara, O.	Centro Atómica Constituyentes, Comisión Nacional de Energía Atómica, Av. General Paz 1499 (1650), San Martín, Buenos Aires, Argentina Fax: +541167727434 Email: novara@cnea.gov.ar
Nyarko, B.J.B.	National Nuclear Research Institute, Ghana Atomic Energy Commission, P.O. Box LG 80, Legon, Accra, Ghana Fax: +23321400807 Email: bjbnyarko@yahoo.co.uk
Olivieira, P.S.P.	Instituto de Pesquisas Energéticas e Nucleares (IPEN/CNEN-SP), Av. Prof. Lineu Prestes, 2242, Cidade Universitária, BR-05508-900 São Paulo, SP, Brazil Fax: +551138169428 Email: patricia@ipen.br
Ozols, A.	Radiation Safety Centre, Maskavas 165, LV-1019 Riga, Latvia Fax: +3717032659 Email: a.ozols@rdc.gov.lv
Padilla, U.	Comisíon Chilena de Energía Nuclear, Amunategui 95, Casilla188-D, Santiago de Chile, Chile Fax: +5626554000 Email: upadilla@cchen.cl
Palomba, M.	ENEA - Research Centre Casaccia, Impianto TRIGA Sp. 40, Via Anguillarese, 301, I-00060 S. Maria di Galeria, Roma, Italy Fax: +390630483379 Email: palomba@casaccia.enea.it

Parente, C.B.R.	Instituto de Pesquisas Energéticas e Nucleares, Centro do Reactor de Pesquisas, C.P. 11049 São Paulo, SP, BR-05422-970 Pinheiros, Brazil Fax: +551138169188 Email: cparente@curiango.ipen.br
Park, C.	Korea Atomic Energy Research Institute (KAERI), 105, Yuseong-gu, Daejeon 305-600, Republic of Korea Fax: +82428688341 Email: cpark@kaeri.re.kr
Pasquini, M.	Agencia Nazionale per la Protezione dell'Ambiente (ANPA), Via Vitaliano Brancati, 48, I-00144 Rome, Italy Fax: +390650072941 Email: pasquini@aspat.it
Pavez, J.	Agustinas 1291, piso 7, Santiago, Chile Fax: +5626963499 Email: juan.pavez@kybs.cl
Pereda, C.P.	Comisión Chilena de Energía Nuclear, Amunátegui 95, Casilla188-D, Santiago, Chile Fax: +5623646277 Email: cpereda@cchen.cl
Perez de Arce, K.	Servicio Nacional de Geología y Minería, Avenida Santa Maria, 01014 Santiago, Chile Fax: +5622385332 Email: smatthews@sernageomin.cl; cperezde@sernageomin.cl
Perl, H.	Universidad Technológica Nacional, Comisión de Investigaciones Científicas (CIC), Gobierno, Provincia de Buenos Aires, Argentina Fax: +541147988255 Email: perl@cnea.gov.ar

Perrin, C.D.	Comisión Nacional de Energía Atómica (CNEA), Avenida del Libertador 8250, 1429 Buenos Aires, Argentina Fax: +541163231701 Email: cperrin@sede.arn.gov.ar
Pesic, M.	VINCA Institute of Nuclear Sciences, Centre for Nuclear Technologies Research, P.O. Box 522, 11001 Belgrade, Serbia and Montenegro Fax: +381112447457 Email: mpesic@vin.bg.ac.yu
Pham, Van Lam	Nuclear Research Institute, 01 Nguyen Tu Luc Street, Dalat, Vietnam Fax: +8463821107 Email: phvlam@hcm.vnn.vn
Piani, C.S.B.	Nuclear Energy Corp. of South Africa (NECSA), P.O. Box 582, 0001 Pretoria, South Africa Fax: +27123055956 Email: csbpiani@necsa.co.za
Pillette-Cousin, L.P.	Commissariat à l'énergie atomique, Direction de la recherche technologique, Route du Panorama, BP 6, F-92265 Fontenay-aux-Roses, France Fax: +33142536796 Email: lucien.pillette@cea.fr
Quintana Domínguez, J.A.	Comisión Nacional de Energía Atómica (CNEA), Avenida del Libertador 8250, 1429 Buenos Aires, Argentina Fax: +541143798282 Email: quintana@cae.cnea.gov.ar
Rachem, Z.	Blanco Encalada 1441 (9) B, C.P. 1428 C.F. Buenos Aires, Argentina Fax: +54 1147850869 Email: zrachem@terra.com; zrachem@hotmail.com

Ramamoorthy, N.	Division of Physical and Chemical Sciences, International Atomic Energy Agency, P.O. Box 100, A-1400 Vienna, Austria Fax: +431260021700 Email: N.Ramamoorthy@iaea.org
Ramanadham, M.	Solid State Physics Division, Bhabha Atomic Research Centre (BARC), Mumbai 400 085, India Fax: +912225505151 Email: ramu@magnum.barc.ernet.in
Ramanathan, L.V.	Instituto de Pesquisas Energéticas e Nucleares, Av. Prof. Lineu Prestes 2242, Cidade Universitária, BR-05508-900 São Paulo, SP, Brazil Fax: +551138169370 Email: lalgudi@ipen.br
Ramos, P.	Inst. de Pesquisas Energéticas e Nucleares, Av. Prof. Lineu Prestes 2242, Cidade Universitária, 05508-900 Sao Paulo, SP, Brazil Fax: +551138169188 Email: cparente@curiango.ipen.br
Rico, N.A.	Centro Atómico Bariloche, Comisión Nacional de Energía Atómica (CNEA), Avenida Bustillo, Km 9,500, 8400 S.C. Bariloche, Rio Negro, Argentina Fax: +542944445299 Email: rico@cab.cnea.gov.ar
Riesle, J.E.	Comisión Chilena de Energía Nuclear, Amunategui 95, Casilla188-D, Santiago, Chile Fax: +562364626300 Email: jriesle@cchen.cl
Ritchie, I.	Division of Nuclear Fuel Cycle and Waste Technology, International Atomic Energy Agency, P.O. Box 100, A-1400 Vienna, Austria Fax: 0043126007 Email: i.ritchi@iaea.org

Rosi, G.	ENEA Casaccia Center, Tapiro Reactor, Via Anguillaresc, 301, I-00060 S. Maria di Galeria, Rome, Italy Fax: +390630486436 Email: giancarlo.rosi@casaccia.enea.it
Salmenhaara, S.E.J.	Technical Research Centre of Finland, VTT Processes, P.O. Box 1608, FIN-02044 VTT, Finland Fax: +35894566390 Email: seppo.salmenhaara@vtt.fi
Sanchez, F.	Centro Atómico Bariloche, Comisión Nacional de Energía Atómica (CNEA), Avenida Bustillo, Km 9,500, 8400 S.C. Bariloche, Rio Negro, Argentina Fax: +542944445189 Email: sanchezf@cab.cnea.gov.ar
Sanhueza, A.	Comisión Chilena de Energía Nuclear, Amunategui 95, Casilla 188-D, Santiago, Chile Fax: +5626554287 Email: asanhuez@cchen.cl
Saud, M.	Comisión Chilena de Energía Nuclear, Amunategui 95, Casilla188-D, Santiago, Chile Fax: +5626554288 Email: msaud@cchen.cl
Saxena, R.N.	IPEN/CNEN-SP, Centro do Reactor de Pesquisa, Av. Prof. Lineu Prestes 2242, Cidade Universitária, BR-05508-000 São Paulo, SP, Brazil Fax: +551138169188 Email: rnsaxena@net.ipen.br
Schlamp, M.	Nuclear Engineering Department, Centro Atómico Bariloche, 8400 Bariloche, Río Negro, Argentina Fax: +542944445299 Email: schlamp@cab.cnea.gov.ar

Sepûlveda, C.	Comisión Chilena de Energía Nuclear, Amunátegui 95, Casilla188-D, Santiago, Chile Fax: +56233646300 Email: csepulve@cchen.cl
Sklenka, L.	Czech Technical University, Faculty of Nuclear Sciences & Physical Engineering, V Holesovickach 2, CZ-180 00 Praha 8, Czech Republic Fax: +420284680764 Email: sklenka@troja.fjfi.cvut.cz
Stevens, J.	Vitacura 2939, Piso 10, Las Condes, CP 6760176, Santiago, Chile
Summerfield, M.	Australian Nuclear Science and Technology Organisation, Building 16 A, Private Mail Bag 1, Menai, NSW 2234, Australia Fax: +61297173640 Email: msx@ansto.gov.au
Syarip, S.	Research and Development Centre for Advanced Technology, JL. Babarsari, P.O. Box 1008, Yogyakarta 55010, Indonesia Fax: + 62-274-487824 Email: syarip@batan.go.id
Szabo, J.	Israel Atomic Energy Commission, 24, Chaim Levanon Street, P.O. Box 7061, 61070 Tel Aviv, Israel Fax: +97236462974
Talanov, S.	Atomic Energy Committee of the Republic of Kazakhstan, Lisa Chaikina Street 4, 480020 Almaty, Kazakhstan Fax: +73272633356 Email: s.talanov@atom.almaty.kz

Taniguchi, T.	Department of Nuclear Safety and Security, International Atomic Energy Agency, P.O. Box 100, A-1400 Vienna, Austria Fax: +43126007 Email: t.taniguchi@iaea.org
Tapia, A.	Asociación Chilena de Seguridad, Vicuña mackenna 152, Santiago, Chile Fax: +5626852199 Email: atapia@achs.cl
Thorp, S.I.	Comisión Nacional de Energía Atómica (CNEA), Avenida del Libertador 8250, 1429 Buenos Aires, Argentina Fax: +541167798433 Email: thorp@cae.cnea.gov.ar
Tomlinson, A.	Servicio Nacional de Geologíca y Minería, Av. Santa María 01404, Providencia, Chile Fax: +562777680 Email: atomlins@sernageomin.cl
Torres, G.	Comision Chilena de Energia Nuclear, Amunategui 95, Casilla188-D, Santiago, Región Metropolitana, Chile Fax: +5623646277 Email: gtorres@cchen.cl
Torres, H.	Comisíon Chilena de Energía Nuclear, Amunategui 95, Casilla188-D, Santiago, Chile Fax: +5623646277 Email: htorres@cchen.cl
Tóth, K.	Hungarian Atomic Energy Authority (HAEA), Nuclear Safety Directorate, Fenyes Adolf u. 4, H-1036 Budapest, Hungary Fax: +3614364883 Email: tothk@haea.gov.hu

Tözsér, S.	KFKI, Atomic Energy Research Institute, Konkoly T. Str. 29-33, H-1121 Budapest, Hungary Fax: +3613959162 Email: tozser@sunserv.kfki.hu
Tretijakov, I.	Research and Development, Institute of Power Engineering, P.O. Box 788, 101000 Moscow, Russian Federation Fax: +70959752019 Email: tretijakov@entek.ru; tretjakov@nikiet.ru
Turgut, M.H.	Turkish Atomic Energy Authority, Cekmece Nuclear Research and Training Center, CNAEM, PK-55, Sefakoy-Istanbul, Turkey Fax: +2125482230/34 Email: turgutm@nukleer.gov.tr
Villarino, E.	INVAP S.E., F. P. Moreno 1089, P.O. Box 961, 8400 S.C. Bariloche, Rio Negro, Argentina Email: men@invap.com.ar
Wassilew-Reul, Ch.	Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit, Heinrich-von-Stephan-Strasse 1, D-53175 Bonn, Germany Fax: +4918883053963 Email: christine.wassilew-reul@bmu.bund.de
Winfield, D. J.	Atomic Energy of Canada (AECL), Chalk River Laboratories, Chalk River, Ontario K0J 1JO, Canada Fax: +16135848023 Email: winfieldd@aecl.ca
Yánez, V.	Chile/TI; Technology Investment & Solutions of Chile Ltd, Edificio Millenium, Vitacura 2939, Piso 10, Las Condes, Santiago, Chile Fax: +5624315050 Email: vvyanez@tischile.cl

Zambrano, J.	Comisión Chilena de Energía Nuclear, Amunátegui 95, Casilla188-D, Santiago, Chile Fax: +5623646277 Email: jzambran@cchen.cl
Zúñiga, S.	Chilectra, Santa Rosa 76, Piso 4, Santiago, Chile Email: szr@chilectra.cl

BLANK

AUTHOR INDEX

Abou Yehia, H.: 131
Akaho, E.H.K.: 491
Alegrechi, D.: 441
Arkhangelsky, N.V.: 527
Auterinen, I.: 313
Aytekin, A.: 415
Barón, J.: 45
Bars, G.: 131
Bertrand, L.: 57
Blaumann, H.: 507
Böck, H.: 515
Bolshinsky, I.: 631
Bond, D.S.: 249
Boogaard, J.P.: 167
Calzette Larrieu, O.: 507
Carballido, J.C.: 193
Chaplot, S.L.: 291
Correa, O.V.: 637
Couturier, J.: 57
Declercq, S.: 217
Dedik, T.: 631
Diamond, V.: 119
Dobrijevic, R.: 677
Dovalt, A.S.: 79
Du Bruyn, J.F.: 185
El-Morshdy, S.E.: 91
Ernest, J.: 365
Estryk, D.S.: 193
Etchepareborda, A.: 441
Fernandes, S.M.C.: 637
Flibor, S.: 365
Flury, C.A.: 441
Fulfaro, R.: 467
Gobrecht, K.H.: 71
Gonçalves, I.F.: 375
Gubel, P.: 217
Haddad, R.: 645
Hainoun, A.: 205
Hargitai, T.: 659
Hergenreder, D.F.: 353
Howard, L.: 249
Huynh Ton Nghiem: 429
Ibarra, G.: 441
Idjakovic, Z.: 677
Ionita, I.: 383
Israr, M.: 569
Joppen, F.: 217
Jova Sed, L.: 545, 593
Jun, B.J.: 457
Karameldin, A.: 613
Kennedy, G.G.: 177
Kharpate, A.V.: 227
Kim, Y.J.: 457
Koonen, E.: 217
Kropík, M.: 279
Kuk, I.H.: 457
Lahodova, Z.: 365
Lanzani, L.: 645
Lawrence, B.: 119
Le Vinh Vinh: 429
Lecot, C.: 353
Ljubenov, V.: 603
Lobo, R.M.: 637
Lopes, E.: 557
Lovotti, O.: 353
Luong Ba Vien: 429
Maciel, F.: 441
Macnab, D: 119
Mahlers, Y.P.: 395
Maksin, T.: 677
Marcondes, G.: 637
Marek, M.: 365
Mariy, A.H.: 91
Martinho, E.: 375
Matejka, K.: 279, 499
Matthews, S.: 299
Mazrou, H.: 339
Mazufri, C.M.: 79

Mazzocchi, V.L.: 477
Mikulski, A.T.: 151
Morozov, S.I.: 145, 573
Muguiro, M.: 441
Mukhopadhyay, R.: 291
Nguyen Kien Cuong: 429
Núñez, J. Mc Leod: 45
Nyarko, B.J.B.: 491
Oliveira, P.S.P.: 263
Osae, E.K.: 491
Paranjpe, S.K.: 291
Parente, C.B.R.: 477
Park, K.B.: 457
Pešic, M.: 603
Pešic, M: 677
Pérez de Arce, C.: 299
Perrint, C.D.: 107
Pervez, S.: 569
Pham van Lam: 429
Piani, C.S.B.: 185, 579
Pillette-Cousin, L.: 557
Quiroga, A.: 645
Ramanadham, M.: 291
Ramanathan, L.V.: 637
Rataj, J.: 365
Reisenweaver, D.: 545, 593
Rive, D.: 57
Rivera, S.: 45
Rodríguez, S.: 645
Rosi, G.: 325
Salgado, J.: 375
Salmenhaara, S.E.J.: 313
Saxena, R.N.: 263, 467
Serfor-Armah, Y.: 491
Shami, Q.D.: 569
Shayi, L.J.: 579
Sklenka, L.: 279, 499
Šotic, O.: 603
Subotic, K.: 603
Summerfield, M.: 25
Syarip, S.: 243
Tözsér, S.: 659
Turgut, M.H.: 415
Verrastro, C.A.: 193
Vidovszky, I.: 659
Viererbl, L.: 365
Villa, M.: 515
Villarino, E.: 403
Wiersemat, H.T.: 167

INDEX OF PAPERS BY NUMBER

IAEA-CN-100/2 339
IAEA-CN-100/03 17
IAEA-CN-100/07 107
IAEA-CN-100/10 45
IAEA-CN-100/12 353
IAEA-CN-100/14 79
IAEA-CN-100/15 441
IAEA-CN-100/17 46
IAEA-CN-100/18 403
IAEA-CN-100/20 507
IAEA-CN-100/25 193
IAEA-CN-100/32 25
IAEA-CN-100/33 119
IAEA-CN-100/34 515
IAEA-CN-100/39 217
IAEA-CN-100/43 467
IAEA-CN-100/44 637
IAEA-CN-100/45 477
IAEA-CN-100/48 263
IAEA-CN-100/53 177
IAEA-CN-100/55 299
IAEA-CN-100/58 279
IAEA-CN-100/59 499
IAEA-CN-100/61 365
IAEA-CN-100/62 613
IAEA-CN-100/64 91
IAEA-CN-100/67 313
IAEA-CN-100/69 131
IAEA-CN-100/70 57
IAEA-CN-100/72 71
IAEA-CN-100/75 557
IAEA-CN-100/80 491
IAEA-CN-100/81 659
IAEA-CN-100/83 593
IAEA-CN-100/84 545
IAEA-CN-100/85 291
IAEA-CN-100/87 227
IAEA-CN-100/89 243
IAEA-CN-100/97 325
IAEA-CN-100/102 457
IAEA-CN-100/104 167
IAEA-CN-100/107 569
IAEA-CN-100/108 151
IAEA-CN-100/111 375
IAEA-CN-100/117 383
IAEA-CN-100/121 573
IAEA-CN-100/123 145
IAEA-CN-100/125 527
IAEA-CN-100/128 579
IAEA-CN-100/129 185
IAEA-CN-100/131 395
IAEA-CN-100/132 249
IAEA-CN-100/134 429
IAEA-CN-100/135 603
IAEA-CN-100/136 677
IAEA-CN-100/137 205
IAEA-CN-100/141 415
IAEA-CN-100/143 631

More than 250 research reactors are currently operating around the world. The level of their operation, utilization, safety and management varies substantially, often depending on the level of development of the particular country. The IAEA has been promoting the exchange of scientific and technical information related to the safety and utilization of research reactors, and issues surrounding the research reactor fuel cycle. A major avenue for such an exchange has been the periodic organization of seminars, symposia and conferences. This international conference in Santiago was a major milestone in that series of meetings. These proceedings contain a summary of the conference, the major findings, the opening addresses and technical papers. Posters are included on a CD-ROM attached to the inside back cover of this publication.

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA
ISBN 92–0–113904–7
ISSN 0074–1884