<table>
<thead>
<tr>
<th>Oral session 1 – International topics</th>
</tr>
</thead>
<tbody>
<tr>
<td>- Status report of the utilisation of research reactors in the new and candidate members of the European Union</td>
</tr>
<tr>
<td>I. Vidovszky, KFKI Atomic Energy Research Institute, Hungary</td>
</tr>
<tr>
<td>- The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO): General description and implications for the research reactor infrastructure needed for R&amp;D</td>
</tr>
<tr>
<td>Y. Sokolov, IAEA, Austria</td>
</tr>
<tr>
<td>- The European Fusion Programme and the role of research reactors</td>
</tr>
<tr>
<td>R. Lässer, R. Andreani, E. Diegele, EFDA Close Support Unit, Germany</td>
</tr>
<tr>
<td>- Production of Mo99 in Europe: Status and perspectives</td>
</tr>
<tr>
<td>H. Bonet, B. David, IRE, Belgium - B. Ponsard, SCK-CEN, Belgium</td>
</tr>
<tr>
<td>- Irradiation of fuels and materials in the Jules Horowitz reactor: The 6th European Union JHR Co-ordination action (JHR-CA)</td>
</tr>
<tr>
<td>D. Iracane, D. Parrat, CEA Cadarache, France</td>
</tr>
<tr>
<td>- Status report on the cost and availability of enriched uranium for research reactors</td>
</tr>
<tr>
<td>H. Müller, J. Laucht, RWE Nukem, Germany</td>
</tr>
<tr>
<td>- Why a fast-flux reactor? The Phénix experimental program</td>
</tr>
<tr>
<td>J. Guidez, L. Martin, J.Dumesmil, CEA Phénix, France</td>
</tr>
<tr>
<td>- A business operations and decommissioning strategy for the Imperial College London research reactor – A financial risk management approach</td>
</tr>
<tr>
<td>S. Franklin, Imperial College London, U - D. Gardner, J. Mumford, R. Lea, J. Knight, PURE Risk Management, UK</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Oral session 2 – Fuel development, qualification, fabrication and licensing</th>
</tr>
</thead>
<tbody>
<tr>
<td>- High density U-Mo fuels – Latest results and reoriented qualification programs</td>
</tr>
<tr>
<td>J. Snelgrove, ANL, USA - P. Lemoine, CEA Saclay, France – L. Alvarez, CNEA, Argentina</td>
</tr>
<tr>
<td>- The French U-Mo group contribution to new LEU fuel development</td>
</tr>
<tr>
<td>J.-M. Hamy, AREVA / Framatome-anp, France - P. Lemoine, CEA Saclay, France – F. Huet, CEA Cadarache, France – C. Jarousse, AREVA / CERCA, France – J.L. Emin, AREVA / COGEMA Logistics, France</td>
</tr>
<tr>
<td>- Main results and status of the development of LEU fuel for Russian research reactors</td>
</tr>
<tr>
<td>A. Vatulin, A. Morozov, V. Suprun, I. Dobrikova, VNIINM, Russia</td>
</tr>
</tbody>
</table>
Main results and status of the CNEA qualification programme for the fabrication of high density MTR fuel
L. Alvarez, N. Boero, J. Fabro, CNEA, Argentina

Post-irradiation examinations on U-Mo full-sized plate: IRIS-2 experiment
F. Huet, J. Noirot, V. Marelle, S. Dubois, CEA Cadarache, France – P. Boulcourt, P. Sacristan, S. Naury, P. Lemoine, CEA Saclay, France

Mechanical calculations on U-Mo dispersion fuel plates with MAIA
V. Marelle, F. Huet, CEA Cadarache, France – P. Lemoine, CEA Saclay, France

Out of pile French research programme on the U-Mo / Al system: First results
S. Dubois, F. Mazaudier, J.P. Piron, P. Martin, J.C. Dumas, F. Huet, CEA Cadarache, France - H. Noël, O. Tougait, University of Rennes, France - C. Jarousse, AREVA / CERCA, France - P. Lemoine, CEA Saclay, France

Qualification of high-density aluminide fuels for BR-2 reactor
A. Beeckmans de West Meerbeeck, P. Gubel, B. Ponsard, SCK-CEN, Belgium - T. Pin, J.-L Falgoux, AREVA / CERCA, France

Oral Session 3 – Reactor operation, fuel safety, core conversion

Status report on the nuclear start-up of FRM-II
K. Schreckenbach, H. Gerstenberg, Munich Technical University, Germany

Reduced enrichment programme for FRM-II. Status 2004/05
A. Röhrmoser, W. Petry, N. Wieschalla, Munich Technical University, Germany

Evaluation of research reactor fuel reliability in support of R&D regulatory requirements
E. Sokolov, AECL, Canada

Review of the accident source terms for aluminide fuel: Application to the BR2 reactor
F. Joppen, SCK-CEN, Belgium

Results of out-of-pile tests of the MIR reactor irradiated fuel at high temperatures
A. Izhutov, V. Alexandrov, A. Grachev, Y. Kosvintsev, A. Novoselov, Z. Chechetkina, RIAR, Russia

Analysis of power peaking in HEU and LEU fuel elements on the safety behavior of FRJ-2
R. Nabbi, G. Damm, I. Neuhaus, Research Center Jülich, Germany

HEU to LEU conversion experience at the UMass-Lowell research reactor
J. White, L. Bobek, University of Massachusetts Lowell, USA
Oral session 4 – Spent fuel management, back-end options, transportation

- The United States Foreign Research Reactor Spent Nuclear Fuel Program: Proposal to modify the program
  C. Messick, US DOE, USA

- R&D for the final disposal of irradiated research reactor fuel elements
  H. Curtius, H. Brücher, Research Center Jülich, Germany

- Reprocessing U-Mo spent fuels: Dissolution experiments on non-irradiated and irradiated materials
  N. Herlet, G. Ferlay, J.P. Dancousse, A. Juvenelle, CEA VALHRO, France

- HABOG: One building for high level waste and spent fuel in The Netherlands. The first year of experience
  J. Kastelein, H. Codée, COVRA, The Netherlands

- Spent fuel from the Finnish TRIGA research reactor in the surroundings of BWR spent fuel final disposal repository. Safety assessment and comparison to the risks of BWR fuel
  S. Salmenhaara, H. Nordman, M. Anttila, VTT, Finland

- Update on NAC's transportation projects
  C. Anne, J. Patterson, NAC International, USA

Poster session 2 – Fuel development, qualification, fabrication and licensing

- The analysis of effect of gaseous fission products on serviceability of various types of fuel elements used in research reactors
  A. Vatulin, G. Kulakov, A. Kosaurov, A. Morozov, I. Dobrikova, VNIINM, Russia

- An investigation on overcoming the interaction problem of U-Mo dispersion fuel by designing simpler monolithic U-Mo rod type fuels and adding third alloying elements to U-Mo

- Interdiffusion between U-Mo alloys and Al or Al alloys at 340°C. Irradiation plan
  A. Fortis, M. Mirandou, M. Ortiz, S. Balart, A. Denis, A. Moglioni, P. Cabot, CNEA, Argentina

- Irradiation and post-irradiation examination facilities in Argentina for testing of high density U compound plates
  G. Ruggirello, S. Halpert, G. Estryk, J. Quintana, CNEA, Argentina

- Computational and experimental analysis of causes for local deformation of research reactor U-Mo fuel pin claddings in case of high burnups
  V. Popov, M. Khmelevsky, IPPE, Russia, Russia - V. Lukichev, RDIPE, Russia - O. Golosov, Institute of Reactor Materials, Russia
### Poster session 3 – Reactor operation, fuel safety, core conversion, spent fuel

- **Evaluation of 3-D energy distribution in a PWC / CCD device combining thermal balance methods with MCNP & Origen-S calculation model**
  S. Kalcheva, P. Gouat, SCK-CEN, Belgium

  - 211

- **The Valmont program: Improved experimental techniques to support the neutronics qualification of U-MoA1**
  J.-P. Hudelot, Ch. Döderlein, M. Antony, J. M. Girard, V. Laval, Ph. Fougeras, G. Willermo, P. Leconte, CEA Cadarache, France

  - 218

- **Core conversion study from silicide to molybdenum fuel in the Indonesian 30 MW multipurpose reactor GA Siwabessy (RSG-GAS)**
  T.M. Sembiring, I. Kunturo, BATAN, Indonesia

  - 225

- **Progress of activities with regard to reconstruction of the research reactor IRT-Sofia**

  - 230

- **WWR-M fuel elements as objects of permanent study and upgrading**
  G. Kirsanov, K. Konoplev, A. Zacharov, A. Poltavski, Petersburg Nuclear Physics Institute, Russia

  - 235

- **The CERCA fuel elements instrumentation manufacturing**
  G. Harbonnier, C. Jarousse, T. Pin, M. Febvre, P. Colomb, AREVA / CERCA, France

  - 239

- **Burn-up measurements of LEU fuel for short cooling times**
  C. Pereda, C. Henriquez, J. Klein, J. Medel, J. Klein, CCEN, Chile

  - 240

- **ETRR-2 in-core fuel management strategy**
  M. Khalil, Alexandria University, Egypt - E. Amin, M. Belal, Atomic Energy Authority, Egypt

  - 245

### Poster session 4 – Spent fuel management, back-end options, transportation

- **Status of the VIND programme. September 2004**
  M. Pesic, VINCA Institute of Nuclear Sciences, Serbia & Montenegro

  - 251

- **New cask for transportation of irradiated rods**
  D. Ohayon, P. Naigeon, AREVA / COGEMA Logistics, France

  - 257

- **Strontium and barium precipitation as carbonates in molten eutectic LiC-KCl**
  C. Caravaca, G. Córdoba, M.J. Tomás, CIEMAT, Spain

  - 258

- **IR-100 reactor and proposals for spent nuclear fuel management**
  S. Bulkin, V. Tikhonov, V. Parabin, RDIE, Russia - S. Smirnov, V. Podtynnykh, SNINE, Ukraine

  - 264
EVALUATION OF 3-D ENERGY DISTRIBUTION IN A PWC/CCD DEVICE COMBINING THERMAL BALANCE METHODS WITH MCNP&ORIGEN-S CALCULATION MODEL

S. KALCHEVA and Ph. GOUAT
SCK•CEN, Belgium Nuclear Research Centre
Boeretang 200, 2400 Mol, Belgium
silva.kalcheva@sckcen.be; philippe.gouat@sckcen.be

ABSTRACT
The on-line determination of the fuel power during fuel irradiation ramp tests, performed in the BR2 reactor, is based on combination of thermal balance methods with 3-D Monte Carlo (MCNP) calculations. The thermal balance methods use thermally insulated rigs, equipped only with temperature and flow rate sensors, allowing measurement of the global energy, deposited in the irradiation device. The detailed 3-D energy distribution in the different parts of the device, including the fuel rod and surrounding structure, is evaluated with MCNP. The developed combined method is able to predict the relative and absolute power distribution in the fuel rod, and the agreement between the forecasts and the measured values is within 4%.

1. Introduction
During the last 30 years, SCK•CEN has gained experience in conducting fuel power transient experiments by means of thermal balance methods. Those methods use thermally insulated rigs equipped only with temperature and flow rate sensors. Consequently, the determination of the power in fuel elements contained in such devices requires a model allowing deduction of the power deposited in each subassembly. This kind of model is built by means of reactor physics codes.

Until recently, mainly deterministic 1-D or 2-D codes were used to obtain the necessary theoretical input. Due to the severe geometric approximations in the models, the calculated absolute fuel powers deviated significantly from the measured values. Since 2001 the Monte Carlo code MCNP is used as the most sophisticated tool for modelling of the specific BR2 reactor core configuration and environment of the irradiation devices.

In the frame of the campaign for fuel transient tests, performed at BR2 in 2003, a combined 3-D MCNP-4C&ORIGEN-S calculation model has been implemented for evaluation of the energy, deposited by neutrons and photons in all parts of the irradiation device PWC/CCD, including the fuel rod and surrounding structures. The developed model is able to predict the relative and absolute power distribution as well, and the agreement between the forecasts and the measured values is within 4%.

2. Neutronic modelling of BR2
The full-scale 3-D heterogeneous geometry model of BR2 was developed using the Monte Carlo code MCNP-4C [1,2] and presented at Fig. 1. The model describes the actual twisted hyperboloid reactor core, formed from skew beryllium prisms with individual orientation of the loaded fuel elements, control rods and engineering devices inside the test holes.
A combination of the 3-D Monte Carlo code MCNP with 1-D depletion code ORIGEN-S [3] was used for the modeling of the 3-D space dependent isotopic fuel depletion in the core. MCNP is used for evaluation of the 3-D specific power distribution in the fuel elements. For this purpose, each of the 6 fuel plates of all 32 fuel elements is divided into axial zones with 6 cm height. ORIGEN-S is used for evaluation of the isotopic fuel depletion versus fuel burn up. The distribution of the fuel burn-up in the fuel elements is calculated, using the dependence of the isotopic fuel depletion on power peaking factor. The total number of the spatial cells with varied fuel depletion in the model is about 4600.

MCNP can only solve neutron-induced photon transport, the contribution from the delayed photons from the fission products can not be computed directly by the code. A separate geometry model of BR2 has been developed for evaluation of the heating from the delayed photons. For this purpose the code ORIGEN-S of the SCALE system is used for evaluations of the photon spectra and the photon intensity from fission products accumulated in the fuel elements during irradiation in BR2 reactor. Using the power peaking factors calculated with MCNP in each fuel element, the axial and radial distributions of the intensity of the delayed photon sources in the core are performed and used as an external source in the independent photon transport calculation in the MCNP model of BR2.

3. Fuel irradiation devices, used in BR2: PWC/CCD

The CCD (Calibration and Cycling Device) is a classic flow calorimeter allowing monitoring the thermal performance of the coolant flowing through it, using a diaphragm flow meter and thermocouples placed at inlet and outlet. A 1 m high helium-4 screen is placed on the outer surface of the CCD. The fuel rod heating is adjusted by varying the BR2 reactor power during dedicated short reactor cycles. The PWC (Pressurized Water Capsule) is an instrumented irradiation capsule for the test of single fuel rod segments with a diameter of 8-15 mm and an active length up to 1000 mm, under steady-state or transient conditions. The target fuel segment is placed into the stainless steel capsule filled with demineralised stagnant pressurized water. The heat generated in the rod is dissipated in radial direction through the stagnant water towards the outer surface of the pressure
capsule by natural convection. The PWC capsule is cooled by the reactor water flow. The calculation model of PWC/CCD is given at Fig. 2.

Figure 2. Calculation model of PWC/CCD – exactly the same model is used in MCNP calculations and in on-line power determination (based on thermal balance method).

4. On-line power determination based on thermal balance method

From the equation of energy conservation (in differential form), we have:

$$dQ = G\left(\frac{dh}{\rho} - \frac{dP}{\rho} \right)$$  \hspace{1cm} (1)

For the coolant flow in the irradiation device. Integration over the height (between the inlet and outlet thermocouples) leads to the equation of the net heat flow rate through the coolant:

$$Q_{MEASURED} = G\left[h(T_{OUTLET}, P_{OUTLET}) - h(T_{INLET}, P_{INLET}) \right] - \frac{G}{\rho} \sum_i P_{OUTLET_i} - P_{INLET_i} \left( T_{AVG_i}, P_{AVG_i} \right) - 2G^2 \sum_i \frac{A_i}{\rho} \left( T_{AVG_i}, P_{AVG_i} \right)^2$$

$$= Q_{TOTAL} - Q_{LOSS\_TOTAL} = (Q_{FUEL} + Q_{STRUCTURE}) - Q_{LOSS\_TOTAL}$$  \hspace{1cm} (2)

Where: $Q$ is the power (W); $G$ is the mass flow rate (kg/s); $h$ is the specific enthalpy (J/kg); $P$ is the pressure (Pa); $\rho$ is the specific mass (kg/m$^3$); $F$ is the specific friction work (J/kg); $T$ is the temperature ($^\circ$C); $A$ is the cross section (m$^2$); $\varsigma$ is the dynamic friction loss coefficient.
The power \( Q_{TOTAL} \) includes the heat generated not only in the fuel \( Q_{FUEL} \), but also in the structural parts of the device \( Q_{STRUCTURE} \). Both contributions must be disentangled in order to get information on the total fuel power. The method consists in computing the relative power distribution in all parts of the device (with the fuel rod loaded) by the MCNP reactor physics code with best estimate modelling to take into account all particles transport and the delayed phenomena. Formally we have:

\[
Q_{STRUCTURE} = W_{STRUCTURE} \times Q_{TOTAL}, \quad Q_{ROD} = W_{ROD} \times Q_{TOTAL},
\]

(3)

in which \( W_{STRUCTURE} \) and \( W_{ROD} \) are the calculated relative power fractions in all structure parts and in the fuel rod, respectively. The MCNP calculations are performed for a number of BR2 control rod height values and \( W \)-values corresponding to the actual control rod height are obtained via interpolation. To obtain the average linear power of the fuel rod, \( Q_{ROD} \) is divided by the length \( l \) of the fuel rod.

For most experimental fuel irradiation programs, not the total power in the fuel rod \( Q_{ROD} \) itself, but rather the maximum linear power \( q_{ROD,MAX} \) is the crucial parameter. To access this parameter, information on the (instantaneous) axial power profile along the fuel rods is needed:

\[
q_{ROD,MAX} = \frac{Q_{ROD}}{l} B, \quad B = \frac{q_{ROD,MAX}}{q_{ROD,AVG}}
\]

(4)

The determination of the maximum linear power in the fuel rod is based on the relative power data obtained from the MCNP calculations. The MCNP calculations yield data for the deposited power in the fuel, subdivided in axial segments of typically 1 to 2 cm length. The resulting axial profiles for the fuel heating can be fitted to determine the \( B \)-factors (axial peaking factors) for every control rod position. The resulting values for \( B \) as a function of the control rod position are subsequently fitted with polynomial functions, which are implemented in the on-line power determination program to determine instantaneous \( B \)-values and thus deduce \( q_{ROD,MAX} \) data from the \( Q_{ROD} \) values.

5. MCNP evaluations of 3-D energy distribution from various contributors

MCNP whole core model (Fig. 1) with incorporated detailed model of PWC/CCD (Fig. 2) was used for evaluation of the 3-D space dependent distribution of the heating, deposited by neutrons and photons into the different structure parts and in the fuel rod. The PWC/CCD device was located into channel E30 of BR2, which was surrounded by 3 fuel channels, 2 reflector channels and 1 channel, containing Control Rod. Detailed axial and radial distributions of the prompt/delayed neutron, prompt/delayed photon heating in the different structure parts (Fig. 3a) and of the linear fission power in the fuel rod (Fig. 3b) have been calculated with MCNP for different positions of the Control Rods.

MCNP calculates the heating caused by neutrons or by \( \gamma \)-rays via the formula:

\[
Q_{i} = \frac{\rho_{a}}{\rho_{g}} \int_{V} \int_{E} H_{i}(E) \Phi_{i}(r,E) dE \frac{dV}{V},
\]

(5)

where \( (i = n, \gamma) \) is the particle type (neutron or photon), \( \rho_{a} \) the atom density (atoms per cm\(^3\)), \( \rho_{g} \) the mass density (g/cm\(^3\)), \( \Phi_{i} \) (cm\(^{-2}\).s\(^{-1}\).J\(^{-1}\)) the flux per unit energy and \( H_{i}(E) \) the heating response function taking into account all energy deposition processes during neutron and photon transport.

The total fuel rod power is defined as the thermal (fission) power \( P_{fis}^{dep} \) deposited in the rod by fission products, betas, gammas and neutrons from the rod itself and by gammas and neutrons, originating from the rest of the reactor:

\[
P_{fis}^{dep} = N_{n} \sum_{i} E_{fis,i}^{dep} R_{fis,i} = \frac{P_{BR2}^{*}}{E_{fis,BR2}} \sum_{i} E_{fis,i}^{dep} \rho_{f} \int_{E} dE \int_{r} d\sigma_{f,i}(E) \varphi_{n}(E,r)
\]

(6)
where: \( R_{\text{fiss},i} [fiss/n] \) is the number of fission reactions, generated by the fissionable isotope \( i \) in the fuel meat; \( N_n [n/s] \) is the intensity of neutrons generated per second in the whole BR2 at the total reactor power \( P_{\text{BR2}} \); \( \rho_i \) and \( \sigma_{f,i} \) are atomic density and microscopic fission cross section of the fissionable isotope \( i \); \( \varphi_n \) is the neutron flux \( [\text{cm}^{-2}] \) per source neutron; \( v_f = 2.43 \) is the average number of fast neutrons emitted per one fission event, \( E_{\text{fiss},BR2} \approx (193 \pm 2) \text{MeV} \) is the effective energy released per fission reaction of \(^{235}\text{U} \) nucleus in BR2 (the standard fuel used in BR2 contains 93% \(^{235}\text{U} \)); \( E_{\text{dep},i} \) is the effective energy deposited in the fuel meat by the fissile nuclei \( 'i' \) and normalized per fission event in this fuel.

**Figure 3:** Examples of: a) calculated neutron and photon heating in a part of the structure around the dummy rod, made of steel AISI304; b) evaluated linear power profile for a 40 cm long fuel rod in PWC/CCD with the reactor control rods at a height of 480 mm, scaled to a reactor power of 56 MW.

An important parameter in the calculation for the fuel irradiations is the amount of energy deposited into the test fuel per fission event in this fuel, \( E_{\text{dep},i} \). This parameter is determined using the known decomposition of the fission energy (for the relevant fissile nuclei, \( 'i' \)) into its constituents: the kinetic energy of the fission products and of the emitted neutrons, betas and antineutrinos, and the energy of the prompt and delayed gammas emitted. It is assumed that the antineutrino energy is completely lost. The track length of fission fragments and \( \beta \)-particles is small and they deposit all their energy locally in the fuel meat of the fuel rod. The fission neutrons have higher free path length and they lose their energy during slowing down in elastic and inelastic collisions outside the fuel meat. The prompt and delayed photons may escape from the fuel rod and interact with the surrounding structural elements. On the other hand, gammas originating from outside the test fuel also deposit some power in the fuel rod, which is calculated separately (using the same calculation as for the gamma heating in the structure parts) and is included in order to obtain an effective value for the deposited energy per fission.

Dedicated MCNP calculations were performed to determine the fraction of the gamma energy deposited inside the fuel rod. The source of prompt photons is calculated in the MCNP automatically for fission and neutron capture reactions. The energy of the deposited delayed photons is calculated using MCNP and ORIGEN-S codes as commented in § 2. The heating in the experimental device includes contributions from particles originating from all geometrical parts of the calculation model (from the irradiation channel itself, from all other channels in the reactor core, from the beryllium
reflector, from all other experimental devices present during the irradiation, from the cooling water, etc.). The results of MCNP calculations of the various contributors into the total fuel rod power (100%) are demonstrated at Fig. 4.

Figure 4. MCNP calculations of the relative contributions (%) into the total fuel rod power by different contributors.

6. Validation: comparison with gamma spectrometry data

The preliminary calculations by MCNP [4] were used together with the thermal balance and \( \gamma \)-spectroscopy methods [5] for the on-line determination of the linear power in the fuel rod during transient ramp tests, which have been performed at BR2 in 2003. The measured activity of certain isotopes of suitable half-life (e.g. \(^{140}\)Ba, \(^{140}\)La) provides data on the time-averaged fission rate during a certain period or on the instantaneous fission rate for a certain reference reactor power. Using the appropriate effective energy per fission these data can be converted to linear power data and compared with the on-line linear power data. Table 1 presents such a comparison for four fuel rod irradiations in the PWC/CCD device [4,5]. The third column summarizes the measured (average) fission rates, normalized to the BR2 powers quoted in the second column. These fission rates were converted into total fuel power values (4th column) and subsequently into linear power data (5th column). The average linear fuel power data from the on-line power determination are collected in the 6th column and the percentage difference values are quoted in the last column. The data are in very good agreement, especially in view of the assessed uncertainties of both methods: 4.2 % for the on-line power determination and 2.7 % for the gamma spectroscopy method. Moreover, the fact that the gamma spectroscopy data are slightly higher is qualitatively consistent with the presumed biasing of the gamma spectroscopy results by + 4 % (on the basis of previous intercomparison exercises [5] with various analysis methods). The resulting axial shapes are in good agreement with the axial fission power profiles obtained by gamma spectroscopy: an average axial shape factor for the four tabulated fuel rods of 1.072 was calculated, compared to an experimental value of 1.080 from gamma spectroscopy. Taking all these factors into account, one obtains a typical uncertainty of 4 to 6 % for the maximum linear power data and 3 to 5 % for the average linear power values.
Table 1. Comparison of on-line power determination, based on combined MCNP&thermal balance method with the γ-spectroscopy measurements.

<table>
<thead>
<tr>
<th>Fuel rod number</th>
<th>BR2 power (MW) Reference</th>
<th>BR2 power (MW) Measured</th>
<th>Fission rate (s⁻¹) Measured</th>
<th>Total fuel power (W) Measured</th>
<th>Average linear fuel power after the transient (W/cm)</th>
<th>Gamma spectr. (Measured)</th>
<th>On-line power det. (MCNP&amp;th. bal.)</th>
<th>Difference (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>14.0</td>
<td>3.924·10¹⁴</td>
<td>12510</td>
<td>301.3</td>
<td>297</td>
<td>+ 1.4 %</td>
<td></td>
<td></td>
</tr>
<tr>
<td>2</td>
<td>20.5</td>
<td>5.797·10¹⁴</td>
<td>18482</td>
<td>446.3</td>
<td>425</td>
<td>+ 4.8 %</td>
<td></td>
<td></td>
</tr>
<tr>
<td>3</td>
<td>20.7</td>
<td>5.537·10¹⁴</td>
<td>17653</td>
<td>423.5</td>
<td>426</td>
<td>- 0.6 %</td>
<td></td>
<td></td>
</tr>
<tr>
<td>4</td>
<td>16.6</td>
<td>5.068·10¹⁴</td>
<td>16158</td>
<td>388.7</td>
<td>369</td>
<td>+ 5.1 %</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

6. Conclusion

BR2 is equipped with irradiation devices PWC/CC D for irradiation ramp tests on fresh or pre-irradiated fuel elements. A combination of thermal balance methods with validated 3-D Monte Carlo (MCNP) calculations has been implemented for the on-line monitoring of the local linear power in the fuel elements. On-line power data from recent experiments have been confirmed by post-irradiation gamma spectroscopy measurements, proving the validity of the combined on-line procedure. The data show that fuel irradiations in BR2 can be performed in a well-controlled way, with an accurate and reliable on-line follow-up of the fuel power.

7. References

1. MCNP-4C. Monte Carlo N-Particle Transport Code System. Oak Ridge National Laboratory, RSICC Computer Collection, CCC-700/MCNP-4C
2. MCNPDATA. Standard Neutron, Photon, and electron Data Libraries for MCNP-4C. Oak Ridge National Laboratory, RSICC Computer Collection, DLC-200/MCNPDATA.
5. L. Borms, Y. Parthoens and A. Gys, “GERONIMO Third campaign: gamma spectroscopy PIE after ramp test on fuel segments GZR02, GZL33, GZL32 and GZR03”, SCK•CEN-R-3783 (September 2004).
ABSTRACT
The VALMONT program aimed at qualifying the HORUS3D (HOrowitz Reactor simulation Unified System) neutronics calculation route that is used for the development of the JHR core, and to verify the correct treatment of UMo/Al (20% enrichment in $^{235}$U) fuel.

The program is composed of two parts. The first part was devoted to the measurement by the oscillation technique of the reactivity effect of UAl/Mo fuel with an accuracy around 1% (1).

The second part consisted of gamma-spectroscopy experiments on a dedicated UMo/Al fuel sample in order to characterize, through axial power profiles and modified conversion ratio of $^{238}$U measurements, the production and absorption effects inside the UAl/Mo fuel.

The overall excellent agreement between high accuracy experiments and calculations allowed to qualify the HORUS3D neutronics calculation route for UMo/Al fuel.

1. Introduction

In the framework of preliminary design on JHR (Jules Horowitz Reactor) which will use the UMo/Al LEU high-density fuel, the experimental VALMONT (Validation of ALuminium MOlybdenum uranium fuel for NeuTronics) program in the MINERVE reactor of CEA Cadarache was launched in 2004.

The program is composed of two parts. The first part was devoted to the measurement of reactivity effect of UAl/Mo fuel. The oscillation technique was used and allowed an accuracy on measurements around 1% (1). Differential effects of the fuel density, the $^{235}$U enrichment, the Molybdenum content and the matrix were studied. The second part consisted of gamma-spectroscopy experiments on a dedicated UMo/Al fuel pin in order to characterize the production and absorption effects inside the fuel. Axial power profiles and modified conversion ratio of $^{238}$U were particularly investigated.

This paper resumes the utilized experimental techniques and the results that were obtained. It also gives a comparison of calculations to experiments.

2. Experiments

MINERVE is a pool type reactor operating at a maximum power of 100 watts. The core is submerged under 3 meters of water and is used as a driver zone for the different experiments located in a central square cavity with a size of about 70 cm by 70 cm.
The core is contained in a rectangular tank containing about 100 m³ of water. The driver zone consists of enriched metallic uranium/aluminium plates clad with aluminium. These are standard Materials Testing Reactor (MTR) fuel elements. Figure 1 shows the core loading for a typical configuration (R1-UO2).

Several lattices corresponding to different neutron spectra can be built in the central region of the MINERVE Reactor. For the VALMONT program, measurements were performed in the PWR/UOx spectrum (R1-UO2 lattice) that is detailed in Figure 2 and corresponds to a moderation ratio Vm/Vf=1.4.

Fig 1. Radial view of the MINERVE Reactor

Fig 2. PWR/UOx experimental lattice (R1-UO2)
2.1. Manufacturing of oscillation samples and of the dedicated UMo/Al fuel pin

The VALMONT samples (see Table 1) were manufactured at CERCA Romans. They are made of a matrix of aluminium containing different fuel components at different concentrations in order to differentiate the physical effects. A sixth previously existing sample containing natural Molybdenum in an Al₂O₃ matrix ("AMo"), was also studied to verify the correct treatment of the different resonant Molybdenum isotopes.

Each sample is made of fuel pellets with standard PWR dimensions (save for UMo/Al 8 and UMo/Al 2.2 samples, consisting of annular pellets in order to adapt the reactivity worth to experimental requirements) and of a double clad in Zircaloy with an external diameter of 1.06 cm and a length of 10.35 cm. High accuracy chemical, isotopic and geometrical characterization was performed for each sample.

### Sample Characteristics

<table>
<thead>
<tr>
<th>Sample</th>
<th>Characteristics</th>
</tr>
</thead>
<tbody>
<tr>
<td>APur</td>
<td>neutral Al₂O₃, serving for the normalization of the sequence</td>
</tr>
<tr>
<td>UappAl</td>
<td>Uₐl₁ with depleted Uranium at low density (2.2 gU/cm³), to assess the effect of the ²³⁸U</td>
</tr>
<tr>
<td>UA120</td>
<td>Uₐl₁ with 20%-enriched Uranium at low density, to measure the effect of the enrichment</td>
</tr>
<tr>
<td>UMo/Al 2.2</td>
<td>UMo/Al with 20%-enriched Uranium at low density, to analyse the effect of Molybdenum</td>
</tr>
<tr>
<td>UMo/Al 8</td>
<td>reference JHR-fuel, with 20%-enriched Uranium at high density (8 g/cm³), to evaluate the effect of the Uranium density</td>
</tr>
</tbody>
</table>

Tab 4. Samples of the VALMONT oscillation program

Besides, a dedicated fuel pin containing UMo/Al with 20%-enriched Uranium at high density (8 g/cm³) was fabricated. It contains annular fuel pellets (in order to limit the reactivity worth of the pin). The other parts of the pin were exactly the same as for R1-UO₂ pins of the lattice. This specific fuel pin was used for -spectroscopic measurements.

2.2. Reactivity worth measurements by the oscillation technique

The technique consists in oscillating samples in the centre of the experimental R1-UO₂ lattice in order to measure the associated reactivity variation with an accuracy (reproducibility of the measurements) better than 1% (at 1σ). Each sample is placed in an oscillation rod and moved periodically and vertically between two positions located in and out of the experimental zone Error! Reference source not found.].

The studied sample is compared to a reference sample that is placed in the bottom of the oscillation rod. Each sample is measured at least 5 times in order to significantly decrease systematic errors. A measurement corresponds to 20 oscillations of 60 seconds each.

The variations of flux induced by the oscillation are detected by a fission chamber placed in the driver zone, that commands to a rotary automatic pilot rod. The pilot rod uses an adjustable surface of cadmium sectors to compensate the reactivity variations. The pilot rod is calibrated using ²³⁵U samples whose reactivity worth is known with uncertainties better than 1% through deterministic calculations.

2.3. -spectroscopic measurements

A critical aspect of the experimental program VALMONT was also to characterize the absorption and production effects inside UMo/Al fuel. In consequence, the program also included -spectroscopic measurements of modified conversion ratio of U-²³⁸ and of the axial power distribution.
2.3.1. Axial power distribution

The axial profile of the total fission rate was obtained by integral gamma spectroscopy on irradiated fuel pins. The gamma-spectroscopy device was made of a standard HPGe detector and adapted electronics, with a lead collimation window allowing the measurement of the fuel pins with axial resolution of 2 cm.

2.3.2. Modified conversion ratio of U-238

The modified conversion ratio is defined as the ratio of the $^{238}\text{U}$ capture rate to the total fission rate inside the pin. It was obtained via gamma spectroscopy measurements of irradiated fuel pins. The determination is based on the measurement of the integral photopeak of a high-yield fission product (FP) relative to the total fission rate inside the pin. The specific fission product gamma ray line that is used is the 293.27 keV line from $^{143}\text{Ce}$. The 277.60 keV gamma ray line from $^{239}\text{Np}$ is used to measure the $^{238}\text{U}$ capture rate because it is related to the number of $^{238}\text{U}$ captures through subsequent beta decays:

$$^{238}\text{U} + n \rightarrow ^{239}\text{U} \rightarrow ^{239}\text{Pu}$$

To perform these measurements, a special device has been developed to detect the low energy-rays with high accuracy using a Low-Energy Germanium detector and adapted electronics.

3. Experimental results – Comparison to calculation results

Figure 3 plots the measured reactivity effect (ordinate in arbitrary unit called “pilot unit”) versus the calculated reactivity effect (abscissa) of $^{235}\text{U}$ calibration samples. The slope of the regression line allows to convert the pilot units to calculation units (c.u.).

Figure 4 shows the results of the measurements of the VALMONT samples by plotting the measured reactivity effects (converted to c.u.) versus the calculated values. The y-error bars correspond to the measurement uncertainties ($2\sigma$). The calculation uncertainties (x-error bars) are due to uncertainties in the material balance of the samples. The plot is normalized to the signal of the inert $\text{Al}_2\text{O}_3$ (“APur”) sample.

The overall excellent agreement between measurement and calculation is illustrated by the proximity of all points to the diagonal. It is confirmed by the comparison of calculations to experiments given in Table 2. A detailed analysis leads to the following comments:

- The absorbing effect of the "AMo" natural Molybdenum sample is calculated with an acceptable deviation of 3.7% (experimental uncertainty: 1.9% at $2\sigma$). Note that a "natural" Molybdenum isotope cross section, that enabled to take into account with high precision the self-shielding interactions of the different natural isotopes, was used for calculations.
- One remarks a change of sign passing from the depleted $\text{UO}_2$ sample ($\text{U}_{\text{appAl}}$) to the 20%-enriched $\text{UAl}_x$ sample ($\text{U}_{20}\%\text{Al}$) and from the low- to the high-density $\text{UMo}/\text{Al}_{2.2}$ and $\text{UMo}/\text{Al}_{8}$ samples; these deviations, however small, are thought to be linked to approximations in the self-shielding calculation scheme.
- The deviations of the $\text{UAl}_x$ and $\text{UMo}/\text{Al}$ fuel samples with the same Uranium loading of 2.2 g/cm$^3$ are identical within the experimental uncertainties. This proves that the Molybdenum effect is correctly treated in the neutronics calculations.
- The deviation experiment.calculation of the RJH reference fuel sample ("$\text{UMo}/\text{Al} 8"$) amounts to 2.4 c.u. and is thereby within the experimental uncertainty of 2.8 c.u. ($2\sigma$). From a qualitative point of view, this result marks the qualification of the neutronics route for this fuel.
Concerning $\gamma$-spectroscopy measurements, 3 fuel pins were analysed (see Figure 2) during the VALMONT program: the UMo/Al pin (V) in the centre of the lattice, the fuel pin close to the UMo/Al pin (U0) and another one farther away (U1), supposed to be representative for the asymptotic homogenous lattice.

The results of modified conversion ratio (MCR) are given in Table 3. The usual over-estimation of calculation of about 2% for the UO$_2$ 3% fuel pin U0 and U1 is found, even for the perturbed neutron spectrum near U0 pin. The deviation for UMo/Al pin (V) is in agreement with the ~2% (1) measurement uncertainty, what evidences the absence of compensation phenomena with regard to the self-shielding.

The results of axial power profiles of fuel pins V, U0 and U1 are shown in Figure 5. Bucklings are presented in Table 4.

An excellent agreement within the statistical uncertainties at 2 is obtained between calculations and experiments. This confirms the ability of HORUS3D to correctly take into account the production and absorption effects inside the UMo/Al fuel.
Fig 5. Axial power profile inside V, U0 and U1 fuel pins

<table>
<thead>
<tr>
<th>Measurement</th>
<th>Calculation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel pin</td>
<td>B_z² (mm⁻²)</td>
</tr>
<tr>
<td>V</td>
<td>1.865E-05</td>
</tr>
<tr>
<td>U0</td>
<td>1.941E-05</td>
</tr>
<tr>
<td>U1</td>
<td>1.918E-05</td>
</tr>
</tbody>
</table>

Tab 4. comparison of measured axial bucklings to calculated axial bucklings

4. Conclusion

The excellent agreement between calculations and experiments about reactivity effect measurements performed by the oscillation technique with an accuracy around 1% (1), shows the ability of HORUS3D to correctly take into account the 20% enrichment in ²³⁵U, the high density and the presence of Molybdenum absorber inside UMo/Al fuel.

A good agreement was also found between MCR and axial power profile experiments and calculations, what qualifies HORUS3D for correctly simulating production and absorption effects inside UAl/Mo fuel.

In the end, the overall excellent agreement between high accuracy experiments and calculations allowed to qualify the HORUS3D neutronics calculation route for UMo/Al fuel.

5. References


CORE CONVERSION STUDY FROM SILICIDE TO MOLYBDENUM FUEL IN THE INDONESIAN 30 MW MULTIPURPOSE REACTOR G.A. SIWABESSY (RSG-GAS)

T.M. SEMBIRING and I. KUNTORO
Center for Development of Research Reactor Technology
National Nuclear Energy Agency (BATAN)
Kawasan PUSPIPTEK, Serpong, Tangerang, 15310 - Indonesia

ABSTRACT
This paper describes the core conversion from silicide to molybdenum core through a series of silicide (2.96 gU cm$^{-3}$) - molybdenum (3.55 gUcm$^{-3}$) mixed transition cores for the Indonesian 30 MW-Multipurpose G.A. Siwabessy (RSG-GAS) reactor. The core calculations are carried out using the two-dimensional multigroup neutron diffusion method code of Batan-EQUIL-2D. The calculated results showed that the proposed silicide-molybdenum mixed transition cores, using the same refueling/reshuffling scheme, meet the safety criteria and it can be used in safely converting from an all-silicide core to an all-molybdenum core.

1. Introduction
The equilibrium silicide core of the Indonesian 30 MW multipurpose G.A. Siwabessy (RSG-GAS) was achieved through a series of the mixed oxide-silicide transition cores with the same uranium meat density of 2.96 gcm$^{-3}$. The fuel management strategy using the mixed cores is chosen due to the nominal power of 30 MWth can be kept as long as the core conversion program. In the core conversion program, the same refueling and reshuffle scheme was also applied for all transition cores, in order to get the efficient and effective core fuel management.

To anticipate the program to use molybdenum (U-Mo) fuels instead of silicide fuels for research reactors, an equilibrium molybdenum core has been designed for RSG-GAS reactor, completely. The previous study showed that the U-Mo fuel with uranium meat density of 3.55 gcm$^{-3}$ and contains 6 w/o Mo and 9 w/o Mo, can be used and proposed as the candidate of the equilibrium molybdenum cores for RSG-GAS reactor [1].

The equilibrium molybdenum core can be achieved in two ways. The first one is through several smaller transition cores which utilise only higher-loading molybdenum fuel elements. In this case, the reactor power must be reduced during the transition period since the active volume of the transition core is smaller than that of the equilibrium core. The other option, which permits operation at the present nominal power, is through a series of silicide-molybdenum mixed transition cores which use both the present (probably partly burned) silicide fuel elements and higher loading molybdenum fuel elements. Therefore, the main objective of this paper is to obtain the transition cores of the RSG-GAS reactor to convert the core from an all-silicide core to an all-molybdenum core using the silicide-molybdenum mixed transition cores.

The disparate study by Suparlina, L. and Sembiring, T.M. [2] showed the equilibrium silicide core with uranium meat density of 3.55 gcm$^{-3}$ can be achieved by the transition cores using mixed uranium meat densities of 2.96 – 3.55 gcm$^{-3}$, without changing the refueling and reshuffle scheme. Based on those studies [1-2], we supposed the equilibrium molybdenum core of the RSG-GAS reactor can be safely achieved using silicide (2.96 gUcm$^{-3}$) - molybdenum (3.55 gUcm$^{-3}$) mixed transition cores. In this paper the core calculation focus on the equilibrium molybdenum core with 6 w/o Mo (U-6Mo-Al).
All core calculations were carried out using the two-dimensional multigroup neutron diffusion code of Batan-EQUIL-2D [3]. The same safety criteria and refueling/reshuffling scheme [4] were imposed as constraints of the calculations.

2. Design and Calculation Method

**RSG-GAS core fuel management**
The RSG-GAS reactor is a multipurpose open-pool type reactor. The reactor has nominal power of 30 MWth using 40 standard fuel elements (FE, each consisting of 21 fuel plates), 8 control fuel elements (CE, each consisting of 15 fuel plates) and 8 absorbers of AgInCd on the \(10 \times 10\) core grid positions as shown in Fig.1. The beryllium and light water are used as the reflector and the moderator and coolant, respectively.

As shown in the Figure 1, the 40 FEs and eight CEs are grouped into eight burn-up classes (batches or zoning). Consequently, at the beginning of cycle (BOC) five FEs and one CE are loaded after discharging the same number of old FEs and CE from the core. As already stated above, the refueling/reshuffling scheme proposed can be categorized as scatter loading, as shown in Fig.1, and detailed information on the FE and CE movement during reshuffling is indicated in Table 1 [4].

**Design constraints**
The mixed silicide-molybdenum cores must fulfill the following design criteria:

a. No modification on the reactor plant, core and fuel element. The number as well as the performance of irradiation positions and facilities must be maintained.

b. The fuel refueling/reshuffling scheme is same as that of the present core (Table 1).

c. The limit of shutdown margin is \(0.5 \% \Delta k/k\).

d. The limit of excess reactivity at the end of cycle (EOC), at the hot and equilibrium-Xe and Sm condition, is \(1.07 \% \Delta k/k\) [4]. The reactivity is provided for experiments, xenon override and flooding-unflooding beam tubes.

e. The limit of radial power peaking factor is 1.4.
Computational procedure and codes

The generation of macroscopic cross-section for molybdenum (U-Mo-Al) and silicide (U₃Si₂-Al) fuels was prepared with the WIMSD-5B lattice calculation code [5] using the '1986' WIMS Nuclear Data Library [6] in the MTR_PC V2.6 system [7]. The cross-section was prepared in the 4-groups using the structure of the neutron energy boundaries of 10 MeV, 0.821 MeV, 5.531 keV, 0.625 eV and 0. eV. The cross-section was generated as a function of %loss of 235U, uranium meat density, temperature and fission-product poisons (Xe and Sm) condition. The cross-section sets were then arranged in a library for the core calculation.

All transition cores were modeled in the 2-dimensional X-Y reactor geometry. The core calculation was carried out using the multigroup neutron diffusion method of the Batan-EQUIL-2D code while the axial buckling was corrected by the 3-dimensional core calculation, the blackness coefficient $\alpha (DB)$, for each neutron energy group, was used in the calculation for the insertion of AgInCd absorbers.

\[
\begin{array}{cccccc}
\text{From} & \text{To} & \text{From} & \text{To} & \text{From} & \text{To} \\
H-9 & F-10 & F-5 & F-8 & C-7 & B-8 \\
H-8 & C-4 & F-4 & F-6 & C-6 & G-5 \\
H-7 & F-7 & F-3 & C-10 & C-5 & D-4 \\
H-6 & D-10 & E-10 & B-4 & C-4 & D-5 \\
H-5 & E-5 & E-9 & G-6 & C-3 & H-8 \\
H-4 & F-9 & E-8 & D-3 & B-9 & C-9 \\
G-9 & E-8 & E-5 & A-8 & B-8 & \text{out} \\
G-8 & \text{out} & E-3 & A-7 & B-7 & \text{out} \\
G-6 & B-7 & D-10 & G-4 & B-5 & \text{out} \\
G-5 & G-8 & D-8 & \text{out} & B-4 & A-6 \\
G-4 & C-7 & D-5 & H-5 & A-9 & A-4 \\
F-10 & G-9 & D-4 & E-9 & A-8 & B-5 \\
F-9 & A-5 & D-3 & C-6 & A-7 & H-7 \\
F-8 & C-5 & C-10 & E-3 & A-6 & B-9 \\
F-7 & F-4 & C-9 & D-8 & A-5 & H-6 \\
F-6 & \text{out} & C-8 & F-5 & A-4 & E-10 \\
\end{array}
\]

Tab 1. Reshuffling and refueling scheme for the present equilibrium core of RSG GAS.

3. Results and Discussions

Table 2 shows an all-molybdenum core is achieved on the eight-transition core (TR-8). It is caused five FEs and one CE are loaded into the core for every operation cycle.

<table>
<thead>
<tr>
<th>Label of the transition cores</th>
<th>Number of silicide fuel (FE/CE)</th>
<th>Number of molybdenum fuel (FE/CE)</th>
<th>Operation cycle length, MWD</th>
<th>$\rho_{SM}$ (%$\Delta k/k$)</th>
<th>Max. discharged burn-up(%)</th>
<th>Max. Radial PPF</th>
</tr>
</thead>
<tbody>
<tr>
<td>TR-1</td>
<td>5/1</td>
<td>35/7</td>
<td>1093.6</td>
<td>0.97</td>
<td>60.4</td>
<td>1.31</td>
</tr>
<tr>
<td>TR-2</td>
<td>10/2</td>
<td>30/6</td>
<td>753.2</td>
<td>1.93</td>
<td>61.4</td>
<td>1.33</td>
</tr>
<tr>
<td>TR-3</td>
<td>15/3</td>
<td>25/3</td>
<td>767.5</td>
<td>1.32</td>
<td>62.2</td>
<td>1.32</td>
</tr>
<tr>
<td>TR-4</td>
<td>20/4</td>
<td>20/4</td>
<td>922.5</td>
<td>0.96</td>
<td>64.5</td>
<td>1.30</td>
</tr>
<tr>
<td>TR-5</td>
<td>25/5</td>
<td>15/3</td>
<td>892.5</td>
<td>1.17</td>
<td>66.2</td>
<td>1.28</td>
</tr>
<tr>
<td>TR-6</td>
<td>30/6</td>
<td>10/2</td>
<td>965.2</td>
<td>0.96</td>
<td>68.1</td>
<td>1.27</td>
</tr>
<tr>
<td>TR-7</td>
<td>35/7</td>
<td>5/1</td>
<td>1050.0</td>
<td>0.96</td>
<td>71.2</td>
<td>1.26</td>
</tr>
<tr>
<td>TR-8</td>
<td>40/8</td>
<td>0/0</td>
<td>1079.2</td>
<td>0.70</td>
<td>69.1</td>
<td>1.27</td>
</tr>
</tbody>
</table>

Note: $\rho_{SM} = shutdown\ margin$

Tab 2. The neutronic calculated parameters for the mixed oxide-silicide cores
As seen in Table 2 and Figure 2, we focused on the imposed constraints, i.e. the shutdown margin, the maximum discharged burn-up and the radial power peaking factor (PPF). By imposing these constraints the shutdown margin on the TR-8 core has slightly higher than the limit value of 0.5 %Δk/k. It is caused the spectrum on the core get harder, particularly, the FEs located around the CEs.

The maximum discharged burn-up of 71.2 % on the TR-7 core is slightly higher than the limit value of 70%. This burn-up can be reduced if the previous transition cores are operated more than the designed operation cycle length listing in Table 2. However, this condition gives an implication that the U-Mo-Al fuel can be licensed-irradiated around 72%. All transition cores have the value of maximum radial power peaking factor (PPF) lower than the limit value of 1.40. It can be seen the maximum radial PPF value tends to decrease with increasing number of the molybdenum fuel elements. This is a consequence of the less heterogeneity core give more flattest radial PPF.

![Figure 2: The selected survey parameter as a function of the transition cores](image)

Table 3 shows only the operation cycle length parameter on the first all-molybdenum core (TR-8) give a significance difference of 11% compared to the equilibrium core one [1]. It is caused the TR-8 core is achieved using lower (silicide) - higher (molybdenum) mixed uranium meat density. For other parameters, there are no significance difference compared to the equilibrium core because in the range of 2% - 3%. Although it is not shown in this paper, the condition of the equilibrium molybdenum core will be achieved on the fifth all-molybdenum core.

<table>
<thead>
<tr>
<th>Parameters</th>
<th>Unit</th>
<th>Limit</th>
<th>TR-8</th>
<th>Equilibrium Molybdenum Core</th>
<th>Difference (TR-8 / Equilibrium Core)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Operation cycle length</td>
<td>days</td>
<td>&gt; 30.</td>
<td>36.</td>
<td>32.5</td>
<td>1,11</td>
</tr>
<tr>
<td>Cold, without Xe and Sm, BOC reactivity</td>
<td>%Δk/k</td>
<td>-</td>
<td>9.37</td>
<td>9.58</td>
<td>0.98</td>
</tr>
<tr>
<td>Shutdown margin</td>
<td>%Δk/k</td>
<td>&gt;0.5</td>
<td>0.70</td>
<td>0.72</td>
<td>0.97</td>
</tr>
<tr>
<td>Maximum Radial Power Peaking Factor</td>
<td>-</td>
<td>≤ 1.4</td>
<td>1.27</td>
<td>1.25</td>
<td>1.02</td>
</tr>
</tbody>
</table>

Tab 3. The Comparison between the full-molybdenum core (TR-8) and equilibrium molybdenum core
4. Conclusion

The calculated result showed that the proposed silicide-molybdenum mixed transition cores, for conversion the lower uranium meat density of silicide core to the higher uranium meat density of molybdenum core, using the same refueling/reshuffling scheme, meet the safety criteria. The calculation results proved that the all-molybdenum core can be safely achieved using the proposed mixed transition cores and the neutronic parameters of the core is almost close to the equilibrium molybdenum core’s parameter.

5. References


PROGRESS OF ACTIVITIES WITH REGARD TO RECONSTRUCTION OF THE RESEARCH REACTOR IRT – SOFIA

T. APOSTOLOV, E. ANASTASOVA, D. DRENSKI, V. ANASTASOV, A. STOYANOVA, E. MOSKOV, S. BELOUSOV, S. KADALEV

Nuclear Scientific Experimental Centre,
Institute for Nuclear Research and Nuclear Energy,
72, Tzarigradsko chausse, Sofia 1784, Bulgaria

and

L. MANEV, M. BALTIYSKI
Atomenergoproekt Ltd,
51 James Bauchier, Sofia 1407, Bulgaria

ABSTRACT

The strategy for the Research Reactor IRT-Sofia, after decision of the Government for its reconstruction, is a partial dismantling of the old systems and equipment, conversion of the new fresh nuclear fuel and transportation of the spent fuel to Russia.

Removal of the reactor core and replacement of old equipment will not pose any significant problems for dismantling. For most efficient use of resources there is a need for implementation of the engineering project, “General plan for partial dismantling of equipment of the IRT-Sofia as a part of the reconstruction into low power RR”, which has been already prepared.

The new research reactor IRT-Sofia is jointly studied with the RERTR Program at Argonne National Laboratory /ANL/ to examine the feasibility of conversion from the use of fuel containing highly enriched uranium /HEU, 36 % 235U/ to use of fuel containing low enriched uranium /LEU, 19,7 % 235U/.

Other important task connected with the reconstruction is shipment of the spent nuclear fuel (EK-10 and C-36 type) to Russia.

1. Introduction

GENERAL PLAN FOR PARTIAL DISMANTLING OF RESEARCH REACTOR IRT-SOFIA PRIOR TO ITS RECONSTRUCTION INTO LOW POWER REACTOR

On the basis of [1] and the General plan for partial dismantling of the reactor and its systems it was specified that the reconstruction shall comprise the following reactor systems:

- Core – liable to full scale replacement. The new core loading shall be pursuant with the type of the new converted (low enriched) fuel (IRT-4M type 20 % enrichment).
- Primary cooling system – replacement of the aluminum lining of reactor pool of 60 m³ capacity, piping and fitments, as well as the heat exchangers.
- Secondary cooling system – partial replacement of piping and fittings and the cooling open-air pools.
- Horizontal experimental channels – their number shall be reduced from the existing 11 to 7 and a new channel is to be set for boron neutron capture therapy.
- Spent fuel storage – replacement of the aluminum reservoir of 12 m³ capacity
- Electric power supply (EPS)- full scale replacement of cables and equipment
- Control and Protection System (CPS)- full scale replacement of cables and equipment
Radiation Monitoring and Dosimetry System (RMDS)—full scale replacement of measuring lines and equipment

Civil Engineering Part - new rooms in the main reactor hall are planned to provide for installation of the main control panels and desks of the reactor

Heating and Climatic Systems – new systems shall be built and installed

Ventilation Systems – partial reconstruction

The disassembly of these systems will be within the capability of the operator organization – (INRNE) with support from contractors, to provide tools and skills, required for size reduction, handling and transportation of the wastes. The General plan for partial dismantling identifies the roles, chain of commands and responsibilities within the dismantling team, and also defines the interfaces with the supporting organizations involved on the INRNE site and supporting sub-contractors.

PROVIDING OF SAFETY AND SECURE SPENT NUCLEAR FUEL TRANSPORTATION FROM IRT - SOFIA SITE. PROCEDURES FOR OBTAINING OF LICENCE

Current report is also focused on one of the priority goals stipulated in Bulgarian Governmental Decision # 332 from May 17, 1999 – removal of spent nuclear fuel (SNF) from IRT- Sofia site and its exporting for reprocessing and/or for temporary storage on NPP Kozloduy site. Readiness of the Bulgarian Government to implement this decision was additionally supported by the “Strategy for management of spent nuclear fuel and radioactive waste” issued in the year 2004 concretely stipulating the actions required for its implementation. By signing of international conventions [2, 3 and 4] related to the safe SNF transportation, the Republic of Bulgaria committed itself to harmonize its secondary legislation both with the international documents and these ones of EU. Current document stipulates the main provisions required for issuing of permit for transportation of such fuel in compliance with the Bulgarian secondary legislation updated recently.

Main requirements in view of the nuclear safety related to the SNF transportation are regulated in [5, 6 and 7]. In these documents the main attention is paid to the technical characteristic of transportation means and equipment. Requirements stipulated in them are in conformity with these ones set out in the recommendable [9 and 10] as well as in the mandatory [11] documents of IAEA. Requirements related to the physical protection during the transportation of nuclear material are determined in [8].

Current common practice is to prepare technical documents needed for granting of permit for SNF transportation specially elaborated and approved by competent State bodies. These documents are not norms and rules but they are directed to solve some concrete cases. Present updated secondary legislation [12 and 13] establishes concrete requirements to the granting of permit for transportation of nuclear material. List of documents required when applying for execution of this activity are enclosed hereto, including administrative acts issued by the respective competent bodies for approval of transportation packing. Lists include:

- Different certificates and evidences related to the applicant;
- Documents in view of nuclear material;
- Documents related to the transportation and technical means used for its execution;
- Program for nuclear safety and radiation protection during the implementation of the activities;
- Documents, related to the staff involved in the transportation activities;
- Documents regulating the relationships between the Shipper and Recipient of the goods as well as between the Applicant and Subcontractors.

Also, the secondary legislation stipulates the conditions for issuing, amendment, renewal and termination, revocation and control of these licences and permits. When elaborating the documents required for obtaining of permit the most important are the criteria for selection of optimum transportation scheme for SNF transportation. The main ones of these criteria are as follows: technical, political and economic.
Concerning the technical criteria the effective secondary legislation mostly establishes the restrictions and provides sufficiently clear instructions for elaboration of needed investigations and analysis proving the capability for safety transportation of this fuel.

Economic criteria as well as their application are banal ones and the only problem faced is to consider precisely all factors forecasting the change of their impact during the process.

In view of the political criteria it is well known that the former social and political system in the Eastern European Countries practically did not require consideration of both internal and external political situation. Situation changed in the recent 15 years requires consideration of the impact of internal factors (influence of ecological movements, public opinion both entirely and in separate regions etc.) as well as of foreign political relationships and (in addition) of the internal situation in foreign states. It is not valid only for the neighbour countries but is also effective worldwide.

The conclusion could be made that currently in the Republic of Bulgaria there is secondary regulation clearly regulating the transportation of SNF from IRT-Sofia, both in the territory of the country and in case of international transportation.

PROGRESS OF CONVERSION FROM HEU TO LEU FUEL AT IRT-200, SOFIA

The new 200 kW IRT-Sofia research reactor of the Institute for Nuclear Research and Nuclear Energy (INRNE) of Bulgarian Academy of Science Sofia, Bulgaria is jointly studied with the RERTR Program at Argonne National Laboratory (ANL) to examine the feasibility of conversion from the use of fuel containing highly enriched uranium (HEU, 36% $^{235}$U) to use of fuel containing low enriched uranium (LEU, 19.75 $^{235}$U).

The reference design had a core configuration using 14 IRT-2M fuel assemblies (four 4-tubes and ten 3-tubes) with 36% HEU. This HEU fuel is no longer available since it was transported from Bulgaria to Russia in December 2003 as part of an agreement with the US DOE and IAEA. An LEU core configuration using 14 IRT-4M fuel assemblies (four 8-tubes and ten 6-tubes) which yields a similar flux performance when compared with the HEU design was created.

Results of detailed calculations comparing the new LEU core with the reference HEU core design are presented. From these results it is concluded that the LEU core performance (both in term of fluxes for the experiments, fuel consumption and temperature solution) is very similar to the HEU reference core.

2. Brief description of contents of material characterization process and measurements in the developed General plan for partial dismantling of IRT – Sofia:

RADIOLOGICAL CHARACTERIZATION ACTIVITIES DESCRIPTION

- Characterization program includes the following steps:
  - Review of historical information
  - Calculation methods implementation
  - Sampling and analyses plan preparation
    - Characterization of activated materials
    - Samples and smears taking and implementation of measurements in the reactor vessel
    - Samples and smears taking and implementation of measurements in premises 103 \first cooling circle\
    - Results from smears and samples
    - Solid RAW, Toxic materials, Non radioactive materials and Liquid radioactive wastes

- Measurements sampling and analyses performance
  - Measurements, sampling and analyses performance
  - Review, evaluation and comparison of data obtained

MEASUREMENTS, SAMPLING AND ANALYSES PERFORMANCE

The performance of the Sampling and analyses plan includes measurements before draining of the water of the first cooling circle and measurements, taking smears and samples after emptying the water.
EXPECTED AMOUNTS OF RAW AT THE PARTIAL DISMANTLING OF IRT-SOFIA EQUIPMENT

I. Expected amounts of RAW – II category according [14].

1. Aluminium and aluminium alloys from the reactor pool - 1100 kg.
2. Steel Sr 1X18H9T + Al alloy AB from the reactor pool - 170 kg.
3. Shield of the thermal column corpus – casing of St 3, filled up with paraffin and boron carbide:
   - St 3 - 289 kg;
   - paraffin with density 1 g/cm³ - 210 kg;
   - Boron carbide with density 1,5 g/cm³ - 60 kg.
4. Lead slab - Pb + Al - 211 kg.
5. Graphite assemblies from the deflector of RC - 172 kg.
6. Ion-exchange resin - 320 kg.
7. According previous measurings, quoted in section 1, it is expected approximately 1000 kg of the thermal column graphite to be activated and to be classified in this category.

   Expected average activity - 10⁵ Bq/kg

In addition:

   Contaminated materials from I CL/Al and St/ from I CL - 3780 kg

It is expected after performing of suitable deactivation the materials of I CL to be converted in RAW of I category or in non radioactive wastes.

II. Expected amounts of RAW – I category, or non radioactive wastes according [14].

1. The concrete shielding of thermal column – concrete with density 4,5 t/m³ - 5,5 m³; 24831 kg (part of the concrete, situated moste near to RC is expected to be volume activated and to pass in upper category waste, according to the classification for RAW)
2. Block with shutters of the thermal column – St3 - 5033 kg

   Note: For RAW is defined the β and γ activity to be in Bq/kg and Bq/cm², because it is measured that RAW do not contain long-lived α – radio nuclides with specific activity over 4.10⁶ Bq/kg.
3. References

[1] Reconstruction of Reactor IRT-2000 at INRNE-BAS to a Low Power Reactor


WWR-M REACTOR FUEL ELEMENTS
AS OBJECTS OF PERMANENT STUDY AND
MODERNIZATION

G. A. KIRSANOV, K. A. KONOPLEV, A.S. POLTAVSKI,
A.S. ZAKHAROV
Department of Reactors Physics and Techniques
Petersburg Nuclear Physics Institute
188300, Gatchina, Russia

ABSTRACT
Brief description of WWR-M5 thin-walled fuel elements and review of possible improvement of parameters for reactor type WWR-M and WWR-SM during transition from fuel elements HEU and LEU WWR-M2 to LEU WWR-M5 is presented.

1. MAIN FEATURES OF WWR-M5 FUEL

An increasing of research scope at the WWR-M reactor set the reactor engineers a task to increase the reactor’s productivity, i.e. to raise the neutron flux in experimental channels. Reactor was designed for 10 MW power. Created at the end of 1950s, domestic fuel assemblies consisting of tubular seamless fuel elements had significant thermophysical margin, which allowed to increase the reactor power systematically even at the first years of its operation. By 1966, the steady-state reactor power was raised to 16 MW. A test increasing of power even to 18 MW took place. The thermal and hydraulic research [1, 2] performed in 1970s has allowed to draw the conclusions that the possibilities of further increase the reactor’s specific power by using of fuel assemblies of the WWR-M2 type have already been exhausted. The maximal fuel element clad temperature has already reached the saturation temperature; and although the margin to the critical thermal load was still sufficiently large (> 2.5), any further increase of power and, consequently, the neutron flux density, in this case, caused the appearance of nucleate boiling on the fuel element surface. Therefore, opportunities to increase the specific thermal power by means of changing fuel element design have been investigated.

For the WWR-M reactor conditions, the optimal combinations of the fuel elements thickness and the gap between them have been determined for various number of the fuel elements in the fuel assembly by the conservation of elementary cell dimensions, i.e. without changing the design of support plate and the beryllium reflector [3,4]. The final choice was the 6-element assembly with the fuel element thickness of 1.25 mm and the gap of 1.5 mm. The overall dimensions of cell boundary have remained as same as WWR-M2.

The new type of fuel element provided an increase of the specific heat-transfer surface by 1.8 times, that under some decreasing of coolant velocity, gave the gain in the specific thermal power by 1.5 times while conservation of previous limitation on the fuel element clad temperature. The stem design of the fuel assembly provides equal cooling velocity of all fuel elements and record specific heat-transfer surface in the core equals to 6.6 l/cm is achieved.

We began with about the same uranium loading as WWR-M2 fuel. Later on the fuel loading was almost increased twice as much. It reduced the fuel component of operational expenses and raised the multiplication factor of the active core.
Evolution of cross section:

- WWR-M2 assembly
- Fuel: U-Al alloy
- WWR-M5 assembly based on U-Al alloy (LEU variant of WWR-M2 - WWR-SM)
- WWR-M5 assembly based on UO$_2$

Fig. 1 WWR-M5 assembly and elements.
After going over to the WWR-M5 the reactor power was increased up to 18 MW in spite of the reduction of the number of fuel assemblies in the core from ≈ 200 to ≈ 130. The increase of the total and specific reactor power was accompanied by the neutron flux growth in the experimental channels. Besides that the additional experimental devices were placed in the released core cells. The experimental device loaded practically every second core cells.

In the process of transition to fuel assemblies of the WWR-M5 type, its comprehensive tests were performed up to operation under spiking specific loads up to 900 kW/l, that is a record for fuel elements of pool-type research reactors [5].

Seamless tubular fuel elements enable to shape a reactor core without loss of volume for structural components. Covers of fuel assemblies for fuel rods and side plates of fuel plate assemblies substantially reduce useful volume of reactor core, thus reducing neutron fluxes applicable for experiment.

Only two types of reactor cores are known, which are free from structural components, namely: reactor cores loaded with tubular fuel elements and reactor cores consisting of a mono type block, fabricated from fuel plates. The first type includes multi-purpose reactors, for example: WWR-M, IRT, IVV-2M and the second type: research beam reactors ILL and FRMII. The latter two are solely intended for extraction of high-density neutron beams and have no irradiation devices within reactor core. In both types, specific heat-emitting surface of fuel elements and reactor core as a whole match with each other.

2. WWR-M5 FUEL ASSEMBLIES FOR WWR-M and WWR-SM REACTORS CONVERSION

We compared reactor cores for reactor type WWR-M during application of LEU WWR-M2 and WWR-M5 fuel elements. The calculations were performed under the MCNP-4C code. Simplified burnup model with one stable fission product was used. Possible burnup in discharge fuel is equal about 55-60 % in both cases.

For LEU WWR-M5 based on uranium dioxide we used uranium density in the matrix approximately 3g/cm³. For LEU on UMo basis - approximately 5.5 g/cm³. Geometry of fuel elements in both cases preserved: thickness of nuclear fuel layer 0.39 mm and cladding 0.43 mm. Therefore, fuel assembly loading by U-235 for active layer height 500mm was approximated to 42 and 72 g. Certain increase of loading is possible after an increase of fuel layer thickness due to reduction of cladding thickness. Application of uranium dioxide in WWR-M5 fuel elements gives only heat-emitting advantages, e.g., factor of non-uniformity of energy release \( K_v \) may be increased from 3 to 5. Heat-emitting advantages can hardly repay increased fuel expenses during transition from 3 to 6 fuel elements assembly.

Application of UMo in WWR-M5 fuel elements enables to increase the volume of experimental devices within reactor core. For example, we present charts of probable loading for WWR-M reactor core with WWR-M2 and WWR-M5 fuel elements (fig.2 and3).

Evidently, transition for UMo of WWR-M5 fuel element doubles the number of channels for radiation within reactor core. This calculation is made as an example of opportunities only, which may be achieved during application of WWR-M5 fuel elements. As far as other reactors and other tasks are concerned, loading will be different, but in all cases UMo-based WWR-M5 will give noticeable advantage vs UO₂-based HEU and LEU WWR-M2 fuel elements in all cases with equal reactivity margin.

Development of LEU WWR-M5 fuel elements to replace WWR-M2 enables to improve the quality of reactors in Budapest and Kiev vs. existing HEU WWR-M2 fuel elements.
3. SUMMARY

During transition of general-purpose reactors (WWR-M type) to LEU with simultaneous expansion of their experimental opportunities, the most advantageous is a fuel assembly, which combines high specific loading of uranium-235 with high specific surface of heat release. To the great extent, LEU WWR-M5 modification on the basis of high-density fuel is the most preferable. To manufacture such fuel assembly there is no requirement to conduct basically new technology. Required fuel density for uranium will not exceed 5.5 g/cm³.

Authors express gratitude to Y.V. Petrov, A.N. Erykalov and M.S. Onegin for offered publications dealing with issues for possible reduction of enrichment in fuel of WWR-M reactor.

4. REFERENCES

When research and test reactors wish to further understand the Fuel Elements behavior when operating as well as mastering their irradiation conditions, operators carry out neutron and thermo hydraulic analysis. For thermal calculation, the codes used have to be preliminary validated, at least in the range of the reactor safety operational limits. When some further investigations are requested either by safety authorities or for its own reactor needs, instrumented tools are the ultimate solution for providing representative measurements. Such measurements can be conducted for validating thermal calculation codes, at nominal operating condition as well as during transients ones, or for providing numerous and useful data in the frame of a new products qualification program. CERCA, with many years of experience for implanting thermocouples in various products design, states in this poster his manufacturing background on instrumented elements, plates or targets.
BURNUP MEASUREMENTS OF LEU FUEL FOR SHORT COOLING TIMES

C. PEREDA B., C. HENRÍQUEZ A., J. KLEIN D., J. MEDEL R.

Comisión Chilena de Energía Nuclear
Amunátegui 95, Santiago, Chile.

ABSTRACT
The measurements presented in this work were made essentially at in-pool gamma-spectrometric facility, installed inside of the secondary pool of the RECH-1 research reactor, where the measured fuel elements are under 2 meters of water. The main reason for using the in-pool facility was because of its capability to measure the burning of fuel elements without having to wait so long, that is with only 5 cooling days, which are the usual times between reactor operations. Regarding these short cooling times, this work confirms again the possibility of using the $^{95}$Zr as a promising burnup monitor, in spite of the rough approximations used to do it. These results are statistically reasonable within the range calculated using codes. The work corroborates previous results, presented in Santiago de Chile [1], and it suggests future improvements in that way.

1. Introduction
In the near future, the RECH-1 research reactor will be completely converted to the use of LEU (19.75% of $^{235}$U) fuel. The current reactor core loads 22 HEU (45% of $^{235}$U) fuel assemblies fabricated by the UKAEA in Deanery, Scotland, and 12 LEU fuel assemblies fabricated by the Chilean Fuel Fabrication Plant (PEC). The meat composition of the experimental LEU fuel assembly is U$_3$Si$_2$-Al, whereas the HEU fuel assemblies have a meat composed by UAl$_x$-Al. The first two LEU fuel assemblies were loaded in the reactor core in December 1998, and the second two in July 1999. LEU fuel assemblies have been gradually loaded in the core to replace HEU fuel assemblies which have reached the discharged burn-up. The total conversion of the RECH-1 reactor will be achieved during the first semester 2006. The first four LEU fuel assemblies loaded in the reactor core are supporting a local qualification program to know the behavior under irradiation of fuel assemblies fabricated by the PEC.

In order to measure the fuel burn up of irradiated fuel assemblies, the CCHEN has two completely independent facilities using gamma spectroscopy technique: a hot cell facility and an in-pool facility described in earlier works [1],[2]. The first facility is mainly used to measure burn up of spent fuel assemblies with decay periods larger than three months. With the purpose to measure burnup of fuel assemblies with shorter decay periods, it was decided to build an in-pool facility.

The measurement of burnup using gamma spectroscopy technique after long decay period is very well known and $^{137}$Cs as monitor gives reliable results[2],[3],[4],[5]. However, the same measurement with short decay periods (few days) produces serious difficulties in the treatment of the collected experimental data. The origin of these difficulties is the high activity generated by a large number of fission products of short life time, which increases the dead time and background reducing the quality of the statistics of the monitor [6], and in our experience submerging completely the $^{137}$Cs under the background radiation, even with 4 months operation and 5 cooling days. Monitors like $^{95}$Zr, $^{140}$La, $^{103}$Ru, $^{95}$Nb, etc. have good statistics; however, they have too short life to keep the accumulated burn up for long irradiation time.

The burn up of one fuel assembly of the RECH-1 research reactor with short decay period was measured at the in-pool facility using $^{95}$Zr (724 keV, peak) as monitor in 2003 [1]. The methodology presented in this paper should be taken as a verification of our first attempt to use $^{95}$Zr as a monitor.
2. Measurements and results

First of all, a full re-evaluation of utilized numbers of our cited previous work [1], was made including a refinement of basic burnup algorithm [3], [4], taking into account the differences of fuel density between external and internal plates of our fuel assemblies, more precise geometric measurements of in-pool measuring system, recalculation of gamma attenuation coefficients involved, and the use of new values of average fission yields adapted to the real operational conditions of RECH-1, an essentially thermal reactor [7]. The outcomes are showed in Table 1 where sub-index 1 and 2 represent the old [1] and new values, as much for $^{95}$Zr as for $^{137}$Cs.

We used essentially the same simple physical assumptions and basic values to formulate the factor $F$ in [1] for $^{95}$Zr except for a generalization that includes the possible non routine decay times. As it is known, this factor is needed to compensate for cumulative decay of $^{95}$Zr occurring during the different irradiation periods between the measurement and the relevant initial core operation. For $^{137}$Cs, a similar $F$ factor is known for a long time [3],[4],[5],[8]. Then, the refined expression for $F$ is:

$$F = \left[ k + e^{-\lambda T_1} (p + 1 - k) - pe^{-\lambda T} - e^{-\lambda T_1} \right]$$

(1)

where, $k$: total numbers of reactor operations,
$\tau$: time between operations,
$T$: maintenance period, and
$p$: number of maintenance periods while the fuel assembly was in the reactor core.
$\lambda$: $^{95}$Zr Disintegration Constant.
$T_1$: non routine decay time.

<table>
<thead>
<tr>
<th>Fuel Assembly Identification</th>
<th>Date of Measurement</th>
<th>Decay Period Days and Monitor</th>
<th>Burnup$_1$ %</th>
<th>Burnup$_2$ %</th>
<th>Calculated Burnup %</th>
</tr>
</thead>
<tbody>
<tr>
<td>LR-04L</td>
<td>30-Jan-03</td>
<td>5 ($^{95}$Zr)</td>
<td>27.88</td>
<td>29.08</td>
<td>22.78</td>
</tr>
<tr>
<td>LR-04L</td>
<td>13-Mar-03</td>
<td>5 ($^{95}$Zr)</td>
<td>23.68</td>
<td>24.29</td>
<td>23.00</td>
</tr>
<tr>
<td>LR-04L</td>
<td>12-Jun-03</td>
<td>5 ($^{95}$Zr)</td>
<td>25.08</td>
<td>25.13</td>
<td>24.52</td>
</tr>
<tr>
<td>LR-04L</td>
<td>24-Sep-03</td>
<td>53 ($^{137}$Cs)</td>
<td>22.75</td>
<td>25.35</td>
<td>25.40</td>
</tr>
</tbody>
</table>

Table 1. Measured burnup results obtained for the LR-04L fuel assembly using $^{95}$Zr and $^{137}$Cs (at fourth row) as monitors, refined in column 2 and explained in the text, compared with Citation code results (fifth column).

Two LEU assemblies made by the PEC were measured during January of 2005; the LR-47 which has nearly 5 months in core, and again (to follow its behaviour) the LR-04L, that in the moment of measuring has nearly 66.5 months in operation. Additionally, this last assembly had an intermediate decay interval of 209 days specifically made to use $^{137}$Cs in measuring the burnup of it, and therefore to have a more reliable result to compare. The LR-04L was taken from the core on 2 August 2003, was measured on 24 September 2003 (see Table 1), and returned to the core on 27 February 2004, and continues in operation until now.

The following two tables, show the relevant results using the aforementioned factor $F$, and two different ways to evaluate the burnup of LR-04L:

(i) The first way, presented in Table 2, consists in repeating the already known procedure [1] taking into account in $F$, that $T_1 = 209$ days for LR-04L. Table 2 includes, besides the result for LR-47 whose operational conduct is much more simple, therefore not requiring the second proposed way.
Table 2. Burnup results obtained for two LEU assemblies measured in January 2005, with the same refinements than Table 1 and using first evaluation way. The calculated burnup using Citation code, corresponds to 17/01/05 (fifth column).

(ii) The second evaluation way, used only in LR-04L for comparison reasons, measures the burnup of this fuel element from the first day it was put on the core again (27/02/04) until the measurement dates. This is equivalent to suppose that it is fresh, or that $P$, $T$ and $T_1$ in (1) are zero. Then the real burnup is obtained adding this number to the previous burnup measured of LR-04L with $^{137}$Cs, whose refined value (25.35%) is in Table 1. The outcomes of this addition is showed in the fourth column on Table 3.

Table 3. Burnup results for the same LR-04L assembly in table 2, using now the burnup results of 24/09/03, with $^{137}$Cs monitor showed in Table 1 as zero reference and explained in text, compared with CITATION code outcomes corresponds to 17/01/05. (fifth column).

The full results of the two described ways of measuring are showed in Graph 1. The averaged differences used in graph are given by:

$$\text{Averaged Diff.} = \frac{1}{2} \left( |B_1 - B_2| + |B_2 - B_3| \right)$$

(2),

where, $B_1$: Measured percentage Burnup with $^{95}$Zr, by first evaluation way.

$B_2$: Measured percentage Burnup with $^{95}$Zr, by second evaluation way.

$B_3$: Calculated percentage Burnup.

The averaged difference is used because of the differences between the burnup measurements dates and the calculated burnup date, not allow to make a precise comparison date to date.
3. Conclusions

-Considering the simplicity of the physical assumptions used to formulate F, and the supposed periodical operation of RECH-1 used on the algorithm to calculate F, the results are promising. The tables show that the averaged differences between the measured burnup and the codes or calculated burnups, are lower if the assembly operational biography is simpler. Similar conclusion is obtained from measurement of the LR-47 assembly. Then, the results using 95Zr are reasonably within the range calculated using codes.
-The measurements confirm that 137Cs is a very reliable burnup monitor even with nearly 50 days cooling period.
-Due to the unavoidable non periodic operation of the RECH-1 and to the relatively short life of 95Zr, the obtained results with this monitor have unavoidable variability, too. It is seen specially in the interoperation measured burnup results. The actual behavior of operation conspire against the simplicity of the algorithm.
-Three possible improvements (in increasing order of importance) are suggested in algorithmic and experimental fields:
(i) Algorithmic: Take into account the actual power history of RECH-1 reactor in a very much detailed way, as with the 137Cs monitor. It is possible that the hypothetical benefit in precision could be smaller than the loss in the simplicity of the present F.
(ii) Experimental: The HPGe detector must be fixed to the mechanical measurement system, due to the fact that it produces important variations in activity measurements, which only contributes to increase the uncertainty in the efficiency of system.
(iii) Experimental and Algorithmic: It is necessary to make continuous measurements during a sufficient time (3 or 4 months) for a specific LEU assembly, in order to improve the empirical knowledge of 95Zr behavior, between and in routine reactor operations.

4. Acknowledgements

The authors wish to express their gratitude to Dr. A. Kestelman, from Bariloche Atomic Center, Argentina, for his scientific assistance in the great challenge to measure fuel burnup using spectrometry technique. We thank also to the operation team of RECH-1 research reactor for its support during the measurements.

5. References


ETRR-2 IN-CORE FUEL MANAGEMENT STRATEGY

M.Y. KHALIL
PhD, Professor of Nuclear Engineering
Department of Nuclear Engineering, Alexandria University
Alexandria, Egypt

ESMAT AMIN
National Center for Nuclear Safety and Radiation Control
Atomic Energy Authority
El Zohoor District, Nasr City, Cairo, Egypt

and

M.G. BELAL
Egypt Second Research Reactor
Atomic Energy Authority, Inshas, 13579, Egypt

ABSTRACT

The Egypt second research reactor has many irradiation channels, beam tubes and irradiation boxes, inside and outside the reactor core. The core reload configuration has great effect on the core performance and fluxes in the irradiation channels.

This paper deals with the design and safety analysis that were performed for the determination of ETRR2 in-core fuel management strategy which fulfills neutronic design criteria, safety reactor operation, utility optimization and achieve the overall fuel management criteria.

The core is divided into 8 zones, in order to obtain the minimum and adjacent fuel movement scheme that is recommended from the operational point of view. Then a search for the initial core using backward iteration, one get different initial cores, one initial core would assume the equilibrium core after 250 full power days of operation, while the other assumes equilibrium after 199 full power days, and shows a better performance of power peaking factor.

1. Introduction

The ETRR-2 core consists of an array of 6x5 with 29 fuel elements and a central irradiation position. In order to determine the fuel management strategy, a lot of variables and boundary conditions should be studied simultaneously.

WIMSD4 is used to generate cross section libraries and CITATION is used for core calculation with a simplified core model of one homogenized fuel region and calculation model of three energy groups, to decrease the computer running time, then a more detailed core and calculation models are used for farther studies and load follow.

Out-in, fixed end of cycle excess reactivity basis along with basic core variables and conventional strategies are used for the first estimation of an appropriate fuel management strategy.

The main reactor operation requirements that are to be satisfied are:- control rod pattern movement strategy that fulfills safe operation and reactor optimum utilization;- A good shutdown margin along with the fuel cycle length that exceeds the maximum begin of cycle excess reactivity;- Appropriate end of cycle reactivity to prevent power transients (which usually appears at the end of core fuel cycle);- Enough EOC reactivity for fixed and non-fixed experiments demands;- the number of discharged fuel elements that have maximum allowable burn-up at the end of each fuel cycle with optimum fuel cycle length for economical fuel usage (the maximum allowable fuel discharge burn up is of the order 63.5% for MTR fuel plate type which is in agreement with NRC recommendation for the fuel plate types to prevent the buildup of the fuel cladding oxidation, that is an exothermic reaction.
which will raise fuel temperature; Core reactivity produced per U-235 gram consumed that achieves
safe reactor operation, limitation on power peaking factor.
A procedure was followed in the design, first a reference core and fuel management schemes were
suggested according to the previous accumulated experience and the reactor design purposes. This
reference core and its reference management strategy was studied and improved for the verification of
operational conditions. From this study, a reference equilibrium core was obtained and a large amount
of data and knowledge for the reactor core is obtained.
Starting with the reference equilibrium core, some variables were studies simultaneously, the
equilibrium cores, the proposed fuel movement schemes and the obtained initial core and its depletion
variable parameters.
The design of reload configuration depends on the stage of the reactor life, batches of fuel elements
available, and on the goal of the reactor operation.
In the paper, and to study the know how to design a reload configuration, the reactor stage of life is
considered to be the reactor before start to operate and produce power.
Concerning the design before start to produce power, there were two reduced fuel elements proposed
in order to obtain an initial core close to equilibrium, standard fuel elements with 404.7gm of U-235
and two reduced, 209gm and 148 grams of U-235, which are equivalent to 48% and 63.5% burn up
respectively and relative to the standard type.
The study represents the result of different stage of designing the fuel management, dividing the core
into two zones, four and six zones, and its corresponding fuel management strategies and core
performance.
Then core is divided into eight zones, to achieve the requirement or minimizing fuel movements in
each chain, which is recommended from the operational point of view.
Concerning the eight zones core, a detailed description of the know how to design a fuel management
strategy that fulfills the operational conditions and safety limits is represented.
A backward iteration was used to estimate the initial core from the estimated equilibrium core, the
iteration starts with the estimated equilibrium core burn up distribution while in the back group there
will be a core with all fresh fuel elements of standard type. The equilibrium core BU distribution and
fresh core represent a starting package of the iteration, which assumes that it has the targeted
equilibrium core.
In each stage of the iteration, when transforming from the BOC of a core to the EOC of the its
supposed previous core, it inserts the three fuel types, and three different EOC core is obtained. After
each stage of the iteration, the obtained previous core is reviewed and some cores are rejected based
on experience.
The result of this iteration is some estimated initial core.
Then starting with this estimated initial cores, using a two dimension code, the core is depleted in
order to test the depletion behavior, during this step some obtained initial cores from the iteration are
rejected as it violate some neutronic and fuel management criteria.
From the last stage equilibrium cores are obtained, compared with the reference equilibrium core,
some are rejected and others remain, a further analysis is performed in the intermediate cores and the
equilibrium core for the verification of the operational conditions.
Some iteration results along with fuel movement schemes give an accepted result.
In the accepted package from the iteration, the fuel management scheme, the initial cores are analyzed
using three-dimension diffusion code.
The work presents the performance of a proposed strategy, and some of recommended initial core, it
was obvious that some initial cores for the same strategy have different results. As one of the initial
cores tends to assume equilibrium after 250 full power days and during depletion have a maximum
power peaking factor of 2.81, while, another initial core tends to the equilibrium after 199 full power
days and during depletion have a maximum power peaking factor of 2.4.

2. Procedure and results

2.1. The design was obtained, using a well known previously tested reference fuel management
strategy, Figure 1 shows the previously tested fuel management strategy.
Table 2 shows the main neutronic data for this strategy the reference core was depleted, and the reference core passing through the intermediate cores up to equilibrium was analyzed.

Then a verification of the operational conditions on the first and second shutdown system is performed.

2.2. A reference core at the average burn up of 45000 MWD/MTU is used to obtain the power destiny distribution in order to predict the burn up and fuel residence time while designing the fuel movement schemes, table 2 list the average burn up core data which will be used in the next design stage.

Tab. 2 list the average burn up core data which will be used in the next design stage

<table>
<thead>
<tr>
<th>Cycle length</th>
<th>BOC Reactivity</th>
<th>EOC Reactivity</th>
<th>Discharge burn up.</th>
<th>FE 16</th>
<th>FE 22</th>
<th>Average</th>
<th>Max PPF</th>
<th>BOC Cold without Xenon Reactivity</th>
<th>EOC cold with xenon Reactivity</th>
<th>EOC hot with xenon Reactivity</th>
</tr>
</thead>
<tbody>
<tr>
<td>19</td>
<td>4380</td>
<td>2510</td>
<td>63.7</td>
<td>59.7</td>
<td>61.7</td>
<td>2.43</td>
<td>8000</td>
<td>6200</td>
<td>7000</td>
<td>5800</td>
</tr>
</tbody>
</table>

Fig. 1 Reference fuel management strategy

Fig. 2 the two zones core (left) and four zones (right), numbers indicated the power released from each zone to sum the 22 nominal reactor power
Using table 2, each sub zone is further divided into sub zones, to collect the 8 zones core, using trail and error, fuel management schemes in each sub zone is suggested and the expected DBU is listed in tables 3, in order to expect how successful the schemes are.

Figs. 3 sub zone sub division (different contrast), numbers show the energy release per sub zone

2.4. Then in order to design the chain sequence, trail and errors used, for the interference in the chains as shown in table 4.
This strategy is studied, starting with the reference equilibrium core, depleting the core until it reaches the equilibrium cycle.

Tab. 5 shows the resultant fuel management main neutronic parameters, which shows that this strategy fulfills the operation constrains and the number of fuel movements is less than the case of two zones core.

The resultant values show a good agreement with the expected values.

2.5. Using the sequence suggested above, initial cores proposed using a backward iteration, different initial cores suggested, figures 4 and 5 shows the different performances for two different initial cores for the same fuel management strategy

<table>
<thead>
<tr>
<th>Cycle</th>
<th>Cycle length</th>
<th>BOC (Rpcm)</th>
<th>EOC (Rpcm)</th>
<th>PPF</th>
<th>DB %</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sub cycle 1</td>
<td>16.94</td>
<td>4260</td>
<td>2350</td>
<td>1.908</td>
<td>62.44</td>
</tr>
<tr>
<td>Sub cycle 2</td>
<td>17.61</td>
<td>4350</td>
<td>2456</td>
<td>1.96</td>
<td>61.25</td>
</tr>
<tr>
<td>Sub cycle 3</td>
<td>17.05</td>
<td>4405</td>
<td>2500</td>
<td>1.853</td>
<td>63.03</td>
</tr>
<tr>
<td>Sub cycle 4</td>
<td>17.34</td>
<td>4405</td>
<td>2484</td>
<td>2.029</td>
<td>60.94</td>
</tr>
</tbody>
</table>

Fig. 4 the performance of the power peaking factor for different initial cores, while depleting up to the equilibrium core

Fig. 5 the power peaking factor for the two proposed starting cores, the second core will assume equilibrium at 199 full power days, and have the same neutronic behavior as the first, which assumes equilibrium at 250 full power days
3. Load follow

As the reactor starts to produce power, the nuclear engineer responsible for the core fuel management should maintain a close load follow, where, during each operation time, the related full power days is calculated, then the core is burned to simulate the actual burn up.

During core burn up, and in order to compare the calculation and the actual case, the control rod positions at different states are compared, and the standard deviation could be obtained to measure the confident on the calculation line used to perform load follow, figure 7 Shows the error obtained during load follow and the standard deviation was 39%, using five group energy boundaries in the calculation model.

![Figure 6](image1.png)  
![Figure 7](image2.png)

Figure 6 shows the error between actual and calculated critical reactivities during the characterization of the fourth ETRR-2 core, using three energy groups in the calculation model.

It shows the error obtained during load follow and the standard deviation was 53%, using three energy groups in the calculation model.

Conclusion

The research reactor fuel management strategy depends on the user demand and the fuel batches available; the core performance is highly dependent on the strategy applied.

The above technique of using the reference core at the average burn up data for predicting the fuel future history has shown a good agreement with the resultant calculated values.

The backward technique is similar to the backward depletion analysis that is used in the power plant fuel management design [3,4] give a good results for the initial cores for a specific strategy.

One stage of the design procedure was not presented in this paper which studies the improvements to the performance of the reference core and utilization optimization, in this stage a data concerning the behavior of the reference core is accumulated, this technique is similar to the heuristic and knowledge search used in the power plant optimization techniques. [3,4]

References


