The 26th Conference of the Nuclear Societies in Israel

February 21-23, 2012, Meridien Hotel, Dead Sea

Program & Papers

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Welcome

It is our pleasure to welcome you to the 26th Conference of the Nuclear Societies in Israel.

The subjects of the Conference are aspects of nuclear energy and nuclear technology, applications of radioisotopes and radiation, radiation measurements, issues in radiation protection and radiation in medicine. The consequences of the Fukushima accident are due to be between the main issues.

During the conference awards will be given for outstanding student works.

On behalf of the organizing committee

Organizing Committee:

Prof. Y. Ronen, Chairman - Ben Gurion University of the Negev
Dr. U. German - Nuclear Research Centre Negev
Dr. I. Orion - Ben Gurion University of the Negev
Dr. E. Shwageraus - Ben Gurion University of the Negev
Dr. I. Yaar - Nuclear Research Centre Negev
Dr. R. Bar-Deroma - Rambam Medical Center
G. Haquin - Soreq Nuclear Research Centre
# Program at a Glance

## Tuesday, February 21, 2012

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<th>Time</th>
<th>Event</th>
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<tr>
<td>13:00 - 14:00</td>
<td>Lunch (Bazaar – Hotel’s Restaurant)</td>
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<tr>
<td>14:00 - 16:15</td>
<td><strong>Plenary Session 1</strong> (Grand B)</td>
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<td></td>
<td>Welcome &amp; Invited Lectures</td>
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<td>Chairman: Dr. J. Koch</td>
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<tr>
<td>16:15 - 16:45</td>
<td>Coffee Break &amp; Visit the Exhibition (Atrium)</td>
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<tr>
<td>16:45 - 18:15</td>
<td><strong>Plenary Session 2</strong> (Grand B)</td>
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<td>Invited Lectures (continues)</td>
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<td>Chairman: Dr. U. German</td>
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<td>Time</td>
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<td>09:00 – 10:30</td>
<td>Grand A</td>
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<td>10:30 – 11:00</td>
<td>Grand C</td>
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## Program

**Tuesday, February 21, 2012**

<table>
<thead>
<tr>
<th>Time</th>
<th>Session/Session 2</th>
<th>Location</th>
<th>Speaker/Chairman</th>
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<td>Plenary Session 1: Welcome &amp; Invited lectures (Grand B)</td>
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<td>Chairman: Dr. J. Koch</td>
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<tr>
<td>14:00</td>
<td>Greetings and Opening Remarks</td>
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<td>Prof. Y. Ronen, Conference Chairman</td>
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<td></td>
<td>Department of Nuclear Engineering, Ben Gurion University of the Negev, Israel</td>
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<tr>
<td>14:15</td>
<td>Nuclear Safety and Security in Israel</td>
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<td>Dr. S. Chorev, Chairman, Israel Atomic Energy Commission</td>
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<tr>
<td>14:45</td>
<td>Nuclear Energy Policy in Israel after Fukushima</td>
<td></td>
<td>Dr. S. Wald, Chief Scientist, Ministry of National Infrastructures, Israel</td>
</tr>
<tr>
<td>15:15</td>
<td>Some Aspects of Introducing a Nuclear Power Plant in Israel Electrical System</td>
<td></td>
<td>Dr. D. Elmakis, Vice President of Planning, Development and Technology Division, Electric Corporation Israel</td>
</tr>
<tr>
<td>16:15-16:45</td>
<td>Coffee Break &amp; Exhibition Visit (Atrium)</td>
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<td>16:45-18:45</td>
<td>Plenary Session 2: Invited Lectures (Grand B)</td>
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<td>Chairman: Dr. U. German</td>
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<tr>
<td>16:45</td>
<td>Silicon Carbide Triplex Nuclear Fuel Cladding, An Alternative to Zircaloy That Avoid Sever Core Damage During Accidents</td>
<td></td>
<td>Dr. H. Feinroth, Ceramic Tubular Products LLC, USA</td>
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<tr>
<td>17:15</td>
<td>The Canadian Experience with Fukushima: Response to an Off-Shore Nuclear Emergency</td>
<td></td>
<td>Mr. T. Jamieson(^2), R. Jammal(^1), G. Rzentkowski(^1) (^) (^)  \  (^1)Canadian Nuclear Safety Commission, Canada (^) (^) (^)  \  (^2)Vice President, Technical Support Branch, Canadian Nuclear Safety Commission, Canada</td>
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<tr>
<td>17:45</td>
<td>IAEC Implementation of Nuclear Safety</td>
<td></td>
<td>Dr. M. Markowitz, Deputy Director General for Licensing and Safety, Israel Atomic Energy Commission</td>
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## Wednesday, February 22, 2012

### 08:30 - 10:00

**Plenary Session 3** (Grand B)
Invited lectures (continues)
Chairman: Prof. A. Galperin

### 10:00 – 10:30

**Coffee Break & Visit the Exhibition** (Atrium)

<table>
<thead>
<tr>
<th>Location</th>
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<tbody>
<tr>
<td>Grand B</td>
<td>Session 2B-1</td>
<td>Reactor Physics and Technology – 1</td>
<td>Dr. E. Shwageraus</td>
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<tr>
<td>Grand C</td>
<td>Session 2C-1</td>
<td>Accelerators / “Saraf” – 1</td>
<td>Dr. Y. Eisen</td>
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### 10:30 – 12:00

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<tr>
<td>Grand A</td>
<td>Session 2A-1</td>
<td>Radiation Protection and General Issues</td>
<td>G. Haquin</td>
</tr>
<tr>
<td>Grand B</td>
<td>Session 2B-2</td>
<td>Reactor Physics and Technology – 2</td>
<td>Dr. A. Keter</td>
</tr>
<tr>
<td>Grand C</td>
<td>Session 2C-2</td>
<td>Accelerators / “Saraf” – 2</td>
<td>Dr. A. Shor</td>
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### 12:00 – 13:00

**Lunch** (Bazaar – Hotel’s Restaurant)

### 13:00 – 14:30

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<tr>
<td>Grand A</td>
<td>Session 2A-2</td>
<td>Medical Physics-1</td>
<td>Dr. I. Orion, Dr. A. Khatib-Hamed</td>
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<tr>
<td>Grand B</td>
<td>Session 2B-3</td>
<td>Medical Physics Society Assembly</td>
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<tr>
<td>Grand C</td>
<td>Session 2C-3</td>
<td>Materials Science-1</td>
<td>Dr. E. Yahel</td>
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### 14:30 – 15:00

**Coffee Break & Visit the Exhibition** (Atrium)

### 15:00 – 18:00

<table>
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<tr>
<td>Grand A</td>
<td>Session 2A-3</td>
<td>Medical Physics-2</td>
<td>Dr. R. Bar-Deroma, Dr. A. Wygoda</td>
</tr>
<tr>
<td>Grand B</td>
<td>Session 2B-4</td>
<td>Natural Radioactivity</td>
<td>Prof. K. Kovler</td>
</tr>
<tr>
<td>Grand C</td>
<td>Session 2C-4</td>
<td>Materials Science-2</td>
<td>Prof. I. Yaacov</td>
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### 20:45 – 22:30

**Get Together and Amnon Einav Awards Ceremony for the Best Student Works** (for hotel guests) (Grand B)
18:15 Innovation Priorities in Nuclear and Radiation Technologies in Russia. View from Skolkovo
Dr. A. Fertman\textsuperscript{1}, D. Kovalevich\textsuperscript{2}, V. Turtikov\textsuperscript{2}, N. Zaytseva\textsuperscript{2}
\textsuperscript{1}Advisor for the President, Skolkovo Foundation on Nuclear Technology, Russia
\textsuperscript{2}Skolkovo Foundation, Nuclear Technology Cluster, Russia

\textbf{Wednesday, February 22, 2012}

\begin{tabular}{|l|l|}
\hline
\textbf{08:30-10:00} & \textbf{Plenary Session 3: Invited Lectures} (Grand B) \\
& Chairman: Prof. A. Galperin \\
\hline
\textbf{08:30} & The New US Public - Private Partnership to License and Deploy Small Modular Reactors, with Focus on the B&W mPower Reactor \\
& Dr. H. Feinroth\textsuperscript{1}, Douglas Lee\textsuperscript{2} \\
& \textsuperscript{1}Ceramic Tubular Products LLC, USA \\
& \textsuperscript{2}Babcock & Wilcox, USA \\
\hline
\textbf{09:00} & High-Order Metrics for Model Uncertainty Quantification and Validation \\
& Prof. D. Cacuci \\
& Institute for Nuclear Technology and Safety, Karlsruhe Institute of Technology, Germany \\
\hline
\textbf{09:30} & No-Failure Design and Disaster Recovery: Lessons from Fukushima \\
& Prof. Y. Ben-Haim \\
& Faculty of Mechanical Engineering, Technion-Israel Institute of Technology, Israel \\
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\textbf{10:00-10:30} & \textbf{Coffee Break & Exhibition Visit} (Atrium) \\
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\hline
\textbf{10:30-12:00} & \textbf{Accelerators / “Saraf” – 1} (Grand C) \\
& Chairman: Dr. Y. Eisen \\
\hline
\textbf{10:30} & Beam Operations of the SARAF Accelerator in 2011 \\
& Leo Weissman \\
& Soreq NRC, Israel \\
\hline
\textbf{10:45} & Improvements Design for the SARAF Phase I RFQ \\
& Zvi Horvitz, Jacob Rodnizki \\
& Soreq NRC, Israel \\
\hline
\textbf{11:00} & Redesign of Existing SARAF RFQ Electrodes to Enable CW Deuteron Operation \\
& Asher Shor \\
& Nuclear Physics and Engineering Division, Soreq NRC, Israel \\
\hline
\textbf{11:15} & SARAF Beam Energy and Phase Distribution Measurements \\
& Arik Kreisel, Leonid Weissman \\
& Soreq NRC, Israel \\
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\end{tabular}
11:30  **Electron Beam Tests of a High-Power Liquid-Lithium Target as an Intense Epithermal Neutron Source**  
Shlomi Halfon\(^1\)\(^,\) Michael Paul\(^2\), Alexander Arenstam\(^1\), Dan Berkovits\(^1\), Ilan Eliyahu\(^1\), Nir Hazenshprung\(^1\), Daniel Kijel\(^1\), Ido Silverman\(^1\)  
\(^1\)Nuclear Physics and Engineering Division, Soreq NRC, Israel  
\(^2\)Racah Institute of Physics, The Hebrew University of Jerusalem, Israel

11:45  **Control System for SARAF Phase I Beam Line and Targets**  
Ilan Eliyahu, Eyal Reinfeld, Leo Weissman, Isaac Gertz, Nir Hazenshprung, Michael Bisyakoev, Ido Silverman, Daniel Kijel, Alex Arenshtam, Shlomi Halfon, Ashe Grin, Amichy Perry, Arik Kreisler, Dan Berkovits  
Nuclear Physics and Engineering Division, Soreq, Israel

**10:30-12:00  Radiation Protection and General Issues (Grand A)**  
Chairman: G. Haquin

10:30  **The 2011 Revised Basic Safety Standards**  
Jean Koch  
Radiation Safety Division, Soreq Nuclear Research Center, Israel

10:45  **Recommendations on Pregnant and Breastfeeding Women Work with Radioactive Materials**  
Nessia Dukhan, Tuvia Kravchik  
Department of Internal Dose Assessment, NRCN, Israel

11:00  **A New High-Purity Germanium Based Whole Body Counter for Internal Dosimetry Assessments at Soreq Nuclear Research Center**  
Radiation Safety Division, Soreq Nuclear Research Center, Yavne, Israel

11:15  **Reaching Out to the Public - Creating a Nuclear Exhibition in Israel**  
Dov Barak\(^1\), Arie Beck\(^1\), Shlomo Vanunu\(^1\), Yehuda Gabay\(^1\), Ariel Talmor\(^1\), Yael Doron\(^2\), Yair Grof\(^3\), Orit Mittler\(^1\)  
\(^1\)NRCN, Israel  
\(^2\)IAEC, Israel  
\(^3\)Soreq NRC, Israel

11:30  **Egyptian Nuclear Activities at Research Reactors, Laboratories, and Uranium Production (Independent Review)**  
Ashraf Saleh  
AGS, Western Central University in Middle East, Egypt

11:45  **What Affects the Isotopic Composition in Rainfall - A New Interpretation**  
Avraham Dody  
NRCN, Israel
10:30-12:00  Reactor Physics and Technology – 1 (Grand B)
Chairman: Dr. E. Shwageraus

10:30  From W. Rothenstein and J. Rowlands Towards Improved Solutions of the Boltzmann Transport Equation (Invited)
Ron Dagan
KIT, Germany

10:45  Cross Section Tuning as an Adjustment
J.J. Wagschal
Racah Institute of Physics, Hebrew University of Jerusalem, Israel

11:00  Demonstration of the Serpent Monte-Carlo Code Applicability to Few-Group Constants Generation for Existing and Advanced Reactor Concepts
Emil Fridman¹, Eugene Shwageraus²
¹HZDR, Germany
²Department of Nuclear Engineering, Ben-Gurion University of the Negev, Israel

11:15  Optimization of OTTO Fuel Management in Pebble-Bed Reactors using Particle Swarm Algorithm
Barak Tavron¹, Eugene Shwageraus², Alex Galperin²
¹Israel Electric Corporation, Israel
²Department of Nuclear Engineering, Ben-Gurion University of the Negev, Israel

11:30  Proliferation Resistant Fuel for Pebble Bed Modular Reactors
Yigal Ronen, Menashe Aboudy, Dror Regev, Erez Gilad
Department of Nuclear Engineering, Ben-Gurion University of the Negev, Israel

11:45  Innovative Pressure Tube Light Water Reactor with Variable Moderator Control
Reuven Rachamin¹,², Emil Fridman¹, Alex Galperin²
¹Institute of Safety Research, Helmholtz-Zentrum Dresden-Rossendorf, Germany
²Department of Nuclear Engineering, Ben-Gurion University of the Negev, Israel

12:00-13:30  Accelerators / “Saraf” – 2 (Grand C)
Chairman: Dr. A. Shor

12:00  Test of a Solid LiF Target in Preparation for Stellar Neutron Production by a Liquid Lithium Target
Gitai Feinberg¹,², Michael Paul², Dan Berkovits¹, Dan Cohen², Ofer Dudovitch¹, Yossi Eisen¹, Moshe Friedman², Yehoshua Ganon², Arik Kreisel¹, Asher Shor¹, Moshe Tessler², Leo Weissman¹
¹Soreq NRC, Israel
²Racah Institute of Physics, Hebrew University, Israel

12:15  Towards the Production of Semi-Maxwellian Neutron Spectrum using SARAF
Yossi Eisen¹, Gitai Feinberg¹,², Asher Shor¹, Moshe Friedman², Antonin Krasa³, Dan Berkovits¹, Georgio Giorginis³, Tsviki Hirsh¹, Michael Paul², Arjan Plompen³
¹Soreq NRC, Israel
²The Hebrew University, Israel
³JRC/IRMM Geel, Belgium
12:30  **Precision Measurements of the Beta Neutrino Correlation in an Electrostatic Ion Trap**  
Sergey Vaintraub\textsuperscript{1,2}, Michael Hass\textsuperscript{1}, Oded Heber\textsuperscript{1}, Michael Rappaport\textsuperscript{1}, Ofer Aviv\textsuperscript{2}, Israel Mardor\textsuperscript{2}  
\textsuperscript{1}Weizmann Institute, Israel, \textsuperscript{2}Soreq NRC, Israel  
12:45  **Production of Ge-68 at SARAF**  
Ido Silverman, Shlomi Halfon, Efi Lavie, Adi Dahan  
Nuclear Physics & Engineering, Soreq, Israel  
13:00  **LISCAN-Liquid Scintillator Capillary fast- Neutron Imaging Spectrometer**  
David Vartsky\textsuperscript{1}, I. Mor\textsuperscript{1,2}, M. Brandis\textsuperscript{1}, M.B. Goldberg\textsuperscript{1}, D. Bar\textsuperscript{1}, I. Mardor\textsuperscript{1}, V. Dangendorf\textsuperscript{2}, B. Bromberger\textsuperscript{2}, K. Tittlemeyer\textsuperscript{2}, M. Weierganz\textsuperscript{2}  
\textsuperscript{1}Soreq Nuclear Research Center (SOREQ NRC), Israel  
\textsuperscript{2}Physikalisch-Technische Bundesanstalt (PTB), Germany  
13:15  **Detector for Simultaneous Detection of Explosives and Special-Nuclear-Materials**  
Michal Brandis\textsuperscript{1,3}, David Vartsky\textsuperscript{1}, Volker Dangendorf\textsuperscript{2}, Benjamin Bromberger\textsuperscript{2}, Doron Bar\textsuperscript{1}, Mark B. Goldberg\textsuperscript{1}, Kai Tittelmeier\textsuperscript{2}, Eliahu Friedman\textsuperscript{3}, Israel Mardor\textsuperscript{1}, Matias Weierganz\textsuperscript{2}  
\textsuperscript{1}Nuclear Physics, Soreq NRC, Israel  
\textsuperscript{2}Physikalisch-Technische Bundesanstalt, Germany  
\textsuperscript{3}Racah Institute of Physics, The Hebrew University, Israel  

12:00-13:30 **Medical Physics-1 (Grand A)**  
Chairmen: Dr. I. Orion, Dr. A. Khatib-Hamed  
12:00  **Interleaved Carbon Minibeams: An Experimental Method of Radiosurgery Developed at Brookhaven National Laboratory**  
Avraham Dilmanian\textsuperscript{1,2}, Adam Rusek\textsuperscript{1}, Giovanna Fois\textsuperscript{1,3}, John Olschowka\textsuperscript{4}, Nicolle Desnoyers\textsuperscript{1}, Jane Park\textsuperscript{1}, Istvan Dioszegi\textsuperscript{1}, Bari Dane\textsuperscript{1}, Ruiliang Wang\textsuperscript{1}, Dardo Tomasi\textsuperscript{1,5}, Hedok Lee\textsuperscript{1}, Sean Hurley, Patricia Coyle\textsuperscript{2}, Allen Meek\textsuperscript{2}, Kerry O’Banion\textsuperscript{4}  
\textsuperscript{1}Brookhaven National Laboratory, USA  
\textsuperscript{2}Stony Brook University Medical Center, USA  
\textsuperscript{3}University of Cagliari, Italy  
\textsuperscript{4}University of Rochester, USA  
\textsuperscript{5}National Institute of Alcohol Abuse and Alcoholism, USA  
12:15  **A New Beam and Delivery System for Radiotherapy**  
Dror Alezra\textsuperscript{1}, Eran Nardi\textsuperscript{1}, Sion Koren\textsuperscript{2}, Dmitri Bragilovski\textsuperscript{2}, Itzhak Orion\textsuperscript{2}  
\textsuperscript{1}Oncology, Sheba Medical Center, Israel  
\textsuperscript{2}Nuclear Engineering, Ben Gurion University, Israel  
12:30  **Monte Carlo Simulations Study for Small Fields in Stereotactic Radio-surgery Beams**  
Itzhak Orion\textsuperscript{1}, Ruth Dadush\textsuperscript{1,2}, Annette Wygoda\textsuperscript{2},  
\textsuperscript{1}BGU, Israel  
\textsuperscript{2}Hadassah Ein Kerem, Israel
12:45  Feasibility of Hippocampal-sparing Whole-brain Radiotherapy Using ELEKTA Linear Accelerator and CMS Monaco Treatment Planning System
Alexander Nevelsky¹, Nantakan Ieumwananonthachai², Raquel Bar-Deroma¹, Rahamim Ben-Yosef¹, Avraham Kuten¹
¹Oncology, Rambam Medical Center, Israel
²Chulabhorn and Siriraj Hospital, Thailand

13:00  Determination of Volume Size Limits for IMRT QA
Ran Ben Hur, Miriam Sabbah, Yanai Krutman
Radiotherapy, Rabin Medical Center, Israel

12:00-13:30  Reactor Physics and Technology – 2 (Grand B)
Chairman: Dr. A. Keter

12:00  A Simplified Spatial Model for BWR Stability
Yonatan Berman¹,², Yoav Lederer², Ehud Meron¹
¹Physics Department, Ben-Gurion University of the Negev, Israel
²Physics Department, Nuclear Research Center-Negev, Israel

12:15  Implementation of Drift-Flux Model in the THERMO Module
Yaniv Shaposhnik¹, Eugene Shwageraus¹, Ezra Elias²
¹Department of Nuclear Engineering, Ben-Gurion University of the Negev, Israel
²Department of Mechanical Engineering, Technion – Israel Institute of Technology, Israel

12:30  Neutronic Analysis in High Conversion Th-233U Core with ELCOS System
Dan Kotlyar, Eugene Shwageraus
Department of Nuclear Engineering, Ben-Gurion University of the Negev, Israel

12:45  Loss of Coolant Accident Analysis for Israel Research Reactor -1
Reuven Rachamin¹,²,³, Ezra Elias², Alex Galperin³
¹Institute of Safety Research, Helmholtz-Zentrum Dresden-Rossendorf, Germany
²Faculty of Mechanical Engineering, Technion – Israel Institute of Technology, Israel
³Department of Nuclear Engineering, Ben-Gurion University of the Negev, Israel

13:00  Examination of AP-1000 Passive Core Cooling System during Lost of Flow Accident
Bar-Or Liran, Yeal Yasur, Yeshayahu Weiss
Rotem Industries LTD., Israel

13:15  Experimental Study of Critical Heat Flux for Low and Zero Flow of Water in Triangle Rod Bundle at Near Atmospheric Pressure
Y. Aharon, I. Hochbaum
NRCN, Israel

13:30 - 14:30  Lunch (Bazaar - Hotel’s Restaurant)

14:30-15:45  Materials Science – 1 (Grand C)
Chairman: Dr. E. Yahel

14:30  Rate Theory of Defects Induced by Ion Irradiation of Fe-12.5at%Cr Alloy
Aleksandr Gokhman, Frank Bergner
Department of Physics, South Ukrainian Pedagogical University, Ukraine
14:45  
**Study of Neutron Induced Defects in Ceramics with Positron Annihilation Technique Using the GiPS Facility**  
Sharon May-Tal Beck¹, Maik Butterling², Wolfgang Anwand², Arie Beck¹, Andreas Wagner², Gerhard Brauer², Or Hen³  
¹Physics Department, NRCN, Israel  
²Institut für Strahlenphysik, Helmholtz-Zentrum Dresden-Rossendorf, Germany  
³School of Physics and Astronomy, Tel-Aviv University, Israel

15:00  
**Chemically Deposited Thorium-Alloyed PbSe Thin Films: A Novel Technique for Self-Irradiation Damage Study**  
Michael Shandalov¹, Moshiel Biton², Eyal Yahel¹, Yuval Golan², Itzhak Kelson³  
¹Physics Department, NRCN, Israel  
²Materials Engineering, Ben-Gurion University, Israel  
³School of Physics and Astronomy, Tel Aviv University, Israel

15:15  
**Pyrolysis and Gasification of Solid, Organic, Low-Level Nuclear Waste: HDPE as a Case Study**  
Yael Peled¹, Ofra Klein Ben-David¹, Zion Ohaion¹, Dorith Tavor², Gabriela Bar-Nes¹  
¹Chemistry, NRCN, Israel  
²Chemical Engineering, Sami Shamoon College of Engineering, Israel

15:30  
**Immobilization of Low Level Wastes in Geopolymeric Systems**  
Ela Ofer-Rozovsky¹, Gabriela Bar-Nes², Michal Arbel-Haddad², Amnon Katz¹  
¹Faculty of Civil and Environmental Engineering, Technion, Israel  
²NRCN, Israel

14:30-16:00  
**Medical Physics – 2 (Grand A)**  
Chairmen: Dr. R. Bar-Deroma, Dr. A. Wygoda

14:30  
**Influence of Angular Dependence on IMRT QA Point-Dose Measurements: Assessment for Two Small Volume Ionization Chambers**  
Jon Feldman¹, Anne Wygoda¹, Ninel Voskobonik¹, Itzhak Orion²  
¹Radiotherapy Unit, Sharett Institute of Oncology, Hadassah University Hospital, Israel  
²Department of Nuclear Engineering, Ben Gurion University of the Negev, Israel

14:45  
**TrueBeam Commissioning: The Bellinson Experience**  
Amineh Khatib Hamad, Janna Menhel, Yanai Krutman, Demitri Bragilovski  
Radiotherapy Department, Rabin Medical Center, Israel

15:00  
**Biological Responses to Reactive Oxygen Species Forecast the Need for Additional Microdosimetric Considerations in Radiotherapy: “To Be or Not to Be”**  
Brenda Laster¹, Liat Vazana², Rachel Bar-Deroma³  
¹Nuclear Engineering, Ben Gurion University, Israel  
²Biochemistry, Ben Gurion University, Israel  
³Oncology, Rambam Medical Center, Israel

15:15  
**Comparative Prospective Study - Exposure to Radiation during URS vs. ESWL**  
Sigalit Haruz-Waschitz¹, Avi Ben-Shlomo², Yoram Siegel³, Amir Cooper⁴, Yaniv Shilo⁵  
¹Safety and Health, Assaf Harofeh Medical Center, Israel  
²Radiation Protection, Soreq NRC, Israel  
³Urology, Assaf Harofeh Medical Center, Israel
15:30 Hospital Preparedness for a Radiological Terrorist Event  
Raquel Bar-Deroma, Liora Utitz, Kobi Moskovitz-Shir  
Rambam Health Care Campus, Israel

14:30-16:30 Natural Radioactivity (Grand B)  
Chairman: Prof. K. Kovler

14:30 Consideration of Radon Leakage from Marinelli Beakers for Accurate Determination of $^{226}$Ra Activity Concentration in Building Materials  
Konstantin Kovler$^1$, Shmuel Levinson$^2$, Zakhar Prilutsky$^1$, Nathan Lavi$^3$, Zeev Alfassi$^3$, Hisham Naser$^4$, Uzi German$^2$,  
$^1$Technion - Israel Institute of Technology, Israel  
$^2$Nuclear Research Centre Negev, Israel  
$^3$Ben Gurion University of the Negev, Israel  
$^4$Ministry of Environmental Protection, Israel

14:45 Radon Exhalation from Concrete Containing Fly Ash: In-Situ and Laboratory Measurements  
Gustavo Haquin$^1$, Konstantin Kovler$^2$, Rachel Becker$^2$  
$^1$Radiation Safety Division, Soreq Nuclear Research Center, Israel  
$^2$Faculty of Civil and Environmental Engineering, Technion - Israel Institute of Technology, Israel

15:00 Assessment of the Dose to the Population of Israel due to Inhalation of Radon  
Lior Epstein$^{1,2}$, Jean Koch$^1$, Tal Riemer$^1$, Gustavo Haquin$^1$, Itzhak Orion$^2$  
$^1$Radiation Safety Division, Soreq Nuclear Research Center, Israel  
$^2$Department of Nuclear Engineering, Ben-Gurion University of the Negev, Israel

15:15 The Exposure of the Israeli Population to Natural Sources of Ionizing Radiation  
Lior Epstein$^{1,2}$, Jean Koch$^1$, Gustavo Haquin$^1$, Itzhak Orion$^2$  
$^1$Radiation Safety Division, Soreq Nuclear Research Center, Israel  
$^2$Department of Nuclear Engineering, Ben-Gurion University of the Negev, Israel

15:30 Differentiation between the Effect of Temperature and Pressure on Radon Transport within the Subsurface Geological Media  
Hovav Zafrir$^1$, Gustavo Haquin$^2$, Uri Malik$^1$, Susana M. Barbosa$^3$  
$^1$Geological Survey of Israel, Jerusalem, Israel  
$^2$Soreq Nuclear Research Center, Israel  
$^3$IDL, University of Lisbon, Portugal

15:45 A New Laboratory System for the Study of the Effect of Temperature on Radon Transport Processes  
Gustavo Haquin$^1$, Hovav Zafrir$^2$, David Katz$^1$, Yoav Yaffe$^1$, Danielle Ilzycer$^1$, Yonatan Ganot$^3$, Zeev B. Alfassi$^3$, Susana M. Barbosa$^3$, Dov Rendlich$^6$, Eliahu Sayag$^6$, Noam Weisbrod$^3$  
$^1$Radiation Safety Division, Soreq Nuclear Research Center, Israel  
$^2$Soreq Nuclear Research Center, Israel  
$^3$Department of Environmental Hydrology and Microbiology, Blaustein Institutes for Desert Research, Ben-Gurion University of the Negev, Israel  
$^4$Department of Nuclear Engineering, Ben-Gurion University of the Negev, Israel  
$^5$IDL, University of Lisbon, Portugal  
$^6$Workshop Unit, Soreq Nuclear Research Center, Israel
16:00 Radon Diffusion in Open Air
Yakov Shitrit¹,², Avraham Dodi¹, Zeev Alfassi²
¹NRCN, Israel
²Nuclear Engineering, Ben Gurion University, Israel

15:45-16:45 Coffee Break & Exhibition Visit (Atrium)

16:15-18:00 Materials Science – 2 (Grand C)
Chairman: Prof. I. Yaacov

16:15 Water Chemisorption on a Sputter Deposited Uranium Dioxide Film - Effect of Defects
Shai Cohen¹,², Noah Shamir¹, Shimon Zalkind¹, Alice Seibert³, Thomas Gouder³, Moshe Mintz¹,²
¹Nuclear Research Center-Negev, Israel
²Department of Nuclear Engineering, Ben-Gurion University of the Negev, Israel
³European Commission - Joint Research Centre, Institute for Transuranium Elements, Germany

16:30 Crystallographic Structure and Magnetic Properties of HAVAR and HAVAR-H under High – Pressure using Diamond Anvil Cell (DAC)
Itzhak Halevy¹,²,³, Shlomo Haroušh¹,⁴, Yosef Eisen¹, Ido Silberman⁴, Dany Moreno¹, Amir Hen¹,², Amir Broide¹, Mike L. Winterrose³, Zhiqiang Chen⁵
¹Physics, Nuclear Research Center - Negev, Israel
²Nuclear Engineering, Ben Gurion University, Israel
³Materials Science, California University of Technology, USA
⁴Soreq NRC, Israel
⁵NSLS, Brookhaven National Laboratory, USA

16:45 Structural, Electronic and Magnetic Characteristics of Np2Co17
Itzhak Halevy¹,²,³, A. Hen², I. Orion²,⁴, E. Colineau⁴, R. Eloirdi⁴, J.-C. Griveau⁴, P. Gaczyński⁴, F. Wilhelm⁵, A. Rogalev⁶, J.-P. Sanchez⁶, M. L. Winterrose³, N. Magnani⁷, A. B. Shick⁸, R. Caciuffo⁴
¹Physics, Nuclear Research Center Negev, Israel
²Nuclear Engineering, Ben Gurion University, Israel
³W. M. Keck Laboratory, California Institute of Technology, USA
⁴ESRF, European Synchrotron Radiation Facility, France
⁵CEA/UJF, SPSMS, UMR-E CEA/UJF, France
⁶CEA/UMR-CNRS, SPSMS, UMR-CNRS, France
⁷Chemical Sciences, Lawrence Berkeley National Laboratory, USA
⁸Physics, Institute of Academy of Sciences of the Czech Republic, Czech Republic

17:00 What is the Structure of Liquid Bismuth?
Elad Caspi¹, Yaron Greenberg¹, Eyal Yahel¹, Brigitte Beuneu², Guy Makov³
¹Physics, Nuclear Research Centre - Negev, Israel
²Laboratoire Leon Brillouin, CEA/Saclay, France
³Materials Engineering, Ben-Gurion University of the Negev, Israel
16:30-18:00  Medical Physics Society Assembly (Grand B)

16:30  TrueBeam Technology – Pushing Linac Design Forward
Jiri Bocanek
Varian Medical Systems

20:45-22:30  Get Together and Amnon Einav Awards Ceremony for the Best Student Works (Grand B)

Thursday, February 23, 2012

09:00-10:30  Non-ionizing Radiation (Grand A)
Chairman: Prof. T. Schlesinger

09:00  Numerical Analysis of the Microwave Auditory Effect
Nir Mordechay Yitzhak, Ronen Hareuveny, Raphael Ruppin
Radiation Safety, Soreq NRC, Israel

09:15  Occupational Exposure to Electromagnetic Fields from MRI Systems
Nir Mordechay Yitzhak, Ronen Hareuveny, Raphael Ruppin
Radiation Safety, Soreq NRC, Israel

09:30  Temporal Variations of Power Frequency Magnetic Fields in Residential Buildings with Indoor Transformer Stations
Ronen Hareuveny1, Nir Mordechay Yitzhak1, Shaiela Kandel2, Raphael Ruppin1
1Radiation Safety, Soreq NRC, Israel
2The Hebrew University, Israel

09:45  Cognitive Effects of Radiation Emitted by Cellular Phones: A Summary of Three Experiments
Ilan Eliyahu1, Ronen Hareuveny1, Roy Luria3, Menachem Margaliot1,
Nachshon Meiran2
1Soreq NRC, Israel
2Ben Gurion University, Israel
3Tel Aviv University, Israel

10:00  Relative Response of Electric Field Probes to Pulsed Radio Frequency Radiation – Preliminary Results
Itzhak Ben David, Hareuveny Ronen, Ilan Eliyahu, Nir Mordechay Yitzhak
Radiation Safety, Soreq NRC, Israel

09:00-10:30  Radiation Measurements (Grand C)
Chairman: Dr. Y. Shamai

09:00  Ariel Mapping of Soil Radiation Contamination
Eli Eltsufin1, Shlomo Mark1, Ilan Yaar2
1Software Engineering, SCE, Israel
2NRCN, Israel

09:15  Environmental Radiation Monitoring Around the Soreq Nuclear Research Center
Hanan Datz, Jean Koch, Ofir Even-Hen, Zohar Yungrais, Tal Riemer, Lea Broshi,
Nurit Kanyon, Gustavo Haquin
Radiation Safety Division, Soreq Nuclear Research Center, Israel
09:30  Combined Fast-Neutron and Gamma Imaging with a Novel Liquid-Xenon Detector Concept
Amos Breskin¹, Itamar Israelashvili¹,⁵, Marco Cortesi², Lior Arazi¹, Sergei Shchemelinin¹, Rachel Chechik¹, Volker Dangendorf³, Benjamin Bromberger³, David Vartsky⁴
¹Weizmann Institute, Israel
²Paul Scherrer Institut (PSI), Switzerland
³Physikalisch-Technische Bundesanstalt (PTB), Germany
⁴Soreq Nuclear Research Center, Israel
⁵Nuclear Research Center-Negev, Israel

09:45  Fissile Mass Estimation by Measuring the Number of Neutron Signals in Time Intervals
Chen Dubi¹, Tal Ridnik¹, Itamar Israelashvili¹, Janos Bagi², Jozsef Huszti³
¹Physics Department, Nuclear Research Center-Negev, Israel
²Nuclear Security Unit, Institute for Transuranium Elements, Italy
³Institute of Isotopes, Hungary

10:00  Determination of ⁹⁰Sr–⁹⁰Y Activity in Urine Samples Using Cherenkov Counting
Shimon Tsroya¹, Omer Pelled¹, Uzi German¹, Rachel Marco¹, Ester Katorza¹, Zeev B. Alfassi²
¹Nuclear Research Center Negev, Israel
²Ben Gurion University, Israel

10:15  Radioactive Content of EQ “Balance” Bracelets
Shimon Tsroya, Omer Pelled, Adi Abraham, Tuvia Kravchik, Uzi German
Nuclear Research Centre Negev, Israel

09:00-10:30  Radiological Risk Assessment (Grand B)
Chairman : Dr. I. Yaar

09:00  Radiation Control with High Sensitive Gamma Neutron Mobile Systems
Andrei Stavrov
R&D, TSA Systems, USA

09:15  A Recent Development in the Field of Automated Biodosimetry; Overview of the RABiT System
Raphael Gonen, Omer Pelled, Marcello Weinstein, Uzi Gherman
Betihut Krina, Nuclear Research Center, Negev, Israel

09:30  An Experimental Study of Indoor Dispersion of Radioactive Material
NRCN, Israel

09:45  Outdoor Dispersion of Radioactive Material
Avi Sharon, Itzhak Halevy, Zvi Berensetin, Moti Levy, Shimshon Tabib- Zada, Or Chen, Daniel Sattinger, Pinhas Benayim, Ilan Yaar
NRCN, Israel
10:00 Rapid Methods for Screening Potentially Internal Contaminated Individuals in a Radiological Dispersion Event
Eyal Peri, Tuvia Kravchik, Noah Vainblat, Ilan Yaar
NRCN, Israel

10:15 Mikro-Copter in Indoor and Outdoor Dispersion of Radioactive Material, “Green Field” and “Red House” Projects
Itzhak Halevy, A. Sharon, E. Boubil, A. Hen, M. Ghelman, T. Ridnik, I. Yaar
Physics, NRCN, Israel

10:30-11:00 Coffee Break & Exhibition Visit (Atrium)

11:00-13:00 Calculation Methods – 2 (Grand C)
Chairman: Dr. I. Orion

11:00 Radioisotopes Identification Algorithm Based on Generic Database
Vladislav Bronfenmakher¹, Alon Osovizky¹, Vitaly Pushkarsky¹, Dimitry Ginzburg¹, Shlomo Mark², Daniel Khankin², Guy Shilon², Udi Wengrowicz³, Yair Ifergan³, Shmuel Levinson³, Tzachi Mazor³, Yagil Kadmon³
¹Radiation Detection Department, Rotem Industries Ltd., Israel
²Software Engineering, Shamoon College of Engineering, Israel
³Electronics and Control Laboratories, NRCN, Israel

11:15 Simulation of Gamma Spectra for Spent Fuel
S. Levinson¹, B. Sarusi¹, O. Pelled¹, U. German¹, Z. B. Alfassi²
¹Nuclear Research Center Negev, Israel
²Ben Gurion University of the Negev, Israel

11:30 An Application for User-friendly MCNP4 Operation
Benny Sarusi¹, Shmuel Levinson¹, Vitali Pushkarsky², Alon Osovizky¹, Izhak Orion³, Uzi German¹
¹Nuclear Research Centre Negev, Israel
²Health Physics Instrumentation Department, Rotem Industries Ltd, Israel
³Ben Gurion University of the Negev, Israel

11:45 An Ab Initio Calculation of Exclusive ⁴He Photodisintegration
Nir Nevo Dinur, Nir Barnea
Racah Institute of Physics, The Hebrew University, Israel

12:00 Carnallite Density Measurement System for the Dead Sea Evaporation Ponds Based on Gamma Radiation
Yagel Achrak¹, Moti Aharoni², Itzhak Orion¹, Yael Peleg²
¹Ben Gurion University of the Negev, Israel
²Dead Sea Works LTD, Israel

12:15 Analysis of the Electron Average Energy Shift for Layer Thickness Estimation
Alon Givon¹, Eitan Tiferet², Itzhak Orion¹
¹Nuclear Engineering, Ben-Gurion University of the Negev, Israel
²Nuclear Research Center-Negev, Israel
12:30 Monte Carlo Simulations of the Response of a Germanium Detector to Gamma Rays
Aviv Ofer, Yoram Nir-El, Gustavo Haquin
Radiation Safety, Soreq Nuclear Research Center, Israel

11:00-13:00 Dosimetry (Grand B)
Chairman: Dr. U. German

11:00 Kinetic Modeling for LiF:Mg,Ti Incorporating Creation of Defects in the Irradiation Stage
Ilan Eliyahu1,2, Yigal Horowitz2, Leonid Oster3
1SOREQ, Israel
2Ben Gurion University of the Negev, Israel
3Sami Shamoon College of Engineering, Israel

11:15 Experimental Examination of Holt Expression for TLD Response to Electron Beams
Shachar Daniel, Raquel Bar-Deroma, Hai Trieu
Oncology, Rambam Medical Center, Israel

11:30 The Effect of Sample/Planchet Geometry on Precision and Glow Peak Shapes in LiF-TLD-100 Thermoluminescent Dosimetry
Dana Sibony1, Yigal Horowitz2, Leonid Oster3
1Energy Engineering Unit, Ben Gurion University, Israel
2Physics Department, Ben Gurion University, Israel
3Physics Unit, Sami Shamoon College of Engineering, Israel

11:45 Minimizing the Residual Signal of LiF:Mg,Ti Readouts
Adi Abraham1, Marcelo Weinstein1, Omer Pelled1, Uzi German1, Zeev B. Alfassi2
1Research Centre Negev, Israel
2Ben Gurion University of the Negev, Israel

12:00 Glow Curve Prediction Based on the Thermal Behavior of a TLD Detector Heated by a Routine Temperature Profile
Tali Bar-Kohany2, Ravit Rozenfeld2, Marcello Weinstein1, Adi Abraham1, Uzi German1, Zeev Alfassi2, Gennady Ziskind2
1NRCN, Israel
2Ben Gurion University of the Negev, Israel

12:15 Parametric Analysis of Temperature Profiles due to a Nitrogen Jet Impinging on a TLD Detector
Idan Cohen1,2, Uzi German1, Gennady Ziskind2, Tali Bar-Kohany2
1NRCN, Israel
2Ben Gurion University of the Negev, Israel

12:30 The Energy Dependence of the Supralinearity of Peaks 7 and 8 In LiF:Mg,Ti (TLD-100) Can Lead to Incorrect Evaluation of the Radiation Dose to Cancer Patients
Hanan Datz1, Yigal Horowitz2, Lior Epstein1, Leonid Oster3, Menachem Margaliot1
1Radiation Safety Division, Soreq Nuclear Research Center, Israel
2Physics Department, Ben Gurion University of the Negev, Israel
3Physics Unit, Sami Shamoon College of Engineering, Israel

12:45 External Dosimetry of PersonsOccupationally Exposed to Ionizing Radiation in Israel during the Years 2004-2010
Sabi Primo, Gustavo Haquin, Hanan Datz
Radiation Safety Division, Soreq Nuclear Research Center, Israel
11:00-13:00  **Radiation Detectors** (Grand A)
Chairman: Dr. Y. Laichter

**11:00** Detector Design Considerations for a Beta Aerosol Monitoring System
E. Gonen¹, U. Wengrowicz¹, A. Osovizky², J. Nir¹, B. Sarusi¹, I. Levin¹,
T. Mazor¹, Y. Kadmon¹, I. Orion³
¹Nuclear Research Center - Negev, Israel
²Health Physics Instrumentation Department, Rotem Industries, Israel
³Department of Nuclear Engineering, Ben Gurion University of the Negev, Israel

**11:15** Development of Mobile Digital Multichannel Analyzer/Scaling System
Yair Ifergan¹, Udi Wengrowicz¹, Alon Osovizky², Amir Broide¹,
Benny Sarusi², Irad Brandys¹, Tzachi Mazor¹, Yosef Cohen², Yagil Kadmon¹
¹Electronic and Control Laboratories, Nuclear Research Center Negev, Israel
²Radiation Detection Department, Rotem Instrument Ltd, Israel

**11:30** Suppression of Afterglow Effect in CsI(Tl) Scintillation Detector by Heat Treatment
Yair Ifergan¹, Zeev Alfassi², Udi Wengrowicz¹
¹Nuclear Research Center Negev, Israel
²Department of Nuclear Engineering, Ben-Gurion University of the Negev, Israel

**11:45** A Modular High Sensitive Radiation Detector for Homeland Security and Post Event Applications
Dimitry Ginzburg¹, Avi Manor¹, Michael Ellenbogen¹,
Vladislav Bronfenmakher¹, Dima Shmidov¹, Yakov Yehuda-Zada²,
Tzachi Mazor², Yagil Kadmon², Alon Osovizky¹
¹Radiation Detection Department, Rotem Industries Ltd, Israel
²Electronics&Control Laboratories, NRCN, Israel

**12:00** The Compliance of a New PRD with the ITRAP+10 Testing Program
Alon Osovizky¹, Dimitry Ginzburg¹, Avi Manor¹, Rami Seif², Max Ghelman²,
Michael Ellenbogen¹, Yossef Cohen², Vitaly Pushkarsky¹,
Vladislav Bronfenmakher¹, Yakir Knafo², Yair Ifargan², Danny Sadan²,
Ehud Gonen², Udi Wengrowicz², Rami Atias², Tzachi Mazor², Yagil Kadmon²,
Yossi Cohen²
¹Radiation Detection Department, Rotem Industries Ltd, Israel
²Electronics&Control Laboratories, NRCN, Israel

**12:15** Preventing the Spectrum Shift Phenomena in a Portable Radiation Spectrometry System by Stabilizing dT Between the System and the Environment
Irad Brandys, Yeshayahu Levin, Yair Ifergan, Rami Atias, Yagil Kadmon
NRCN, Israel

**12:30** High Efficiency Power Supply with an Analog Temperature Compensation for Silicon Photomultipliers
Vadim Berdichevsky¹, Max Ghelman¹, Alon Osovizky², Tzahi Mazor¹,
Yagil Kadmon¹
¹Nuclear Research Center - Negev, Israel
²Health Physics Instrumentation Department, Rotem Industries Ltd., Israel

13:00-14:00  **Lunch** (Bazaar – Hotel’s Restaurant)
Some Aspects of Introducing a Nuclear Power Plant in Israel Electrical System

Dr. D. Elmakis

Planning, Development & Technology Division – Israel Electric

Corresponding Author: elmakis@iec.co.il

The Israel isolated electrical system emphasizes the need for energy security of supply by diversification of energy sources for electricity production.

Nuclear Power is considered as one of the most efficient technologies for achieving these goals as well as for climate change mitigation efforts. We will present below the benefits and the risks for introducing a Nuclear Power Plant (NPP) in Israel electricity generation system, considering recent repercussions of the Fukushima accident.

Simulations of introduction a NPP in Israel Electric system showed an increase of reliability and a reduction of electricity generation cost in case of long gas shortcuts. In addition, NPP introduction will reduce significantly greenhouse gas emissions without considerable increase of electricity generation cost.

Some NPP safety aspects will also be evoked and the additional risk to the population and the environment will be evaluated, following sitting of a Generation III+ NPP (Westinghouse's AP1000 and Areva's EPR technologies) at the Shivta site in the Negev desert area.
Silicon Carbide Triplex Nuclear Fuel Cladding  
An Alternative to Zircaloy that Avoids Severe Core Damage during Accidents

Herbert Feinroth  
CEO, Ceramic Tubular Products LLC  
Rockville, Maryland Lynchburg, Virginia

Corresponding Author: hfeinroth@gamma-eng.com

The accidents at Three Mile Island in 1979, and more recently at Fukushima in Japan, were caused by different circumstances but had one common underlying cause. The loss of cooling water to the reactor core that occurred shortly after reactor shutdown led to severe core melting and fission product release, primarily because (1) the weak zircaloy clad ballooned thus blocking coolant flow, (2) the zircaloy reacted exothermically with emergency core coolant producing overwhelming quantities of heat, and (3) the zircaloy water reaction released large quantities of flammable hydrogen gas. It was the zircaloy that was the prime cause of the severe melting, and not the core decay heat as presumed by most observers. A recent analysis of the TMI accident by Fauske Associates, sponsored by Westinghouse, evaluated what the TMI scenario would have been with a passively safe fuel cladding such as Silicon Carbide. The peak temperatures within the TMI core would not have exceeded 1170 °C, (as compared to the actual peak temperatures that exceeded 2800 °C), and only some low melting alloys used in some core components would have melted (3000 pounds) as compared to the 68,000 pounds of fuel materials and cladding that actually melted. There would have been no hydrogen explosion, and although the damaged reactor core would need to be replaced, it is possible that the power plant itself would have been recoverable, thus savings billions of dollars. Although it is too soon to say for sure, it is possible that a similar result would have occurred at Fukushima. Certainly there would have been no hydrogen explosions destroying the fuel pool buildings.

This paper will describe efforts in the US to develop an alternative to zircaloy that performs at least as well during normal operation (actually better) but assures minimum core damage even in the event of a severe loss of coolant accident. The new technology is called Silicon Carbide Triplex Nuclear Fuel Cladding. The development work is ongoing, but enough work has been done to make clear that this technology is feasible and potentially economic. With continued support from both the US Government and the private industry, we believe the technology can be demonstrated, and commercialized within the next decade. The following paragraphs describe the history of this development, the scientific basis for our conclusions on passive safety and potential reliability of this new cladding material, and the remaining challenges that must be overcome to achieve successful deployment.

Early History - The idea of replacing metal fuel cladding with a ceramic composite began to emerge after extensive study of the core that essentially melted in the accident at Three Mile Island Unit 2 (TMI-2) in 1979. Early investigations, sponsored by the NRC and by DOE, studied
a ceramic composite made from alumina fibers and an alumina matrix, known as a continuous fiber ceramic composite (CFCC). This clad concept involves the use of a composite material in its central layer that is not brittle, and instead behaves like a metal when subject to mechanical loads. That is, it stretches under load without fracture, and when it does exceed allowable stress, fails in a graceful failure mode very similar to the failure behavior of ductile metallic cladding. These early investigations concluded that alumina composites were not acceptable for two reasons – the composite was permeable to fission gases, and the alumina lost much of its strength during irradiation in the MIT research reactor.

Our research then focused on a multilayered ceramic system that would embody the hermeticity needed to retain fission gas, and the ductile behavior needed for robust in-pile service. The inner layer was a high density monolith to hold fission gases, and the central layer was a composite with the required strength and graceful failure mode. The research also switched from alumina to silicon carbide, based on many years of US Government sponsored fusion research that demonstrated beta phase stoichiometric silicon carbide would retain its strength under irradiation. As shown in the figure below, such silicon carbide composites retain their strength at temperatures of 1500 °C, as compared with zircaloy which loses most of its strength above 500 °C. The high strength at high temperature assures survival of the triplex SiC clad with minimum damage (fission gas release only) during LOCA events. The material is also expected to be resistant to failure during departure from nucleate boiling (DNB) transients, thus allowing an increase in power density, and power output. And because silicon carbide is very hard, it is expected to be resistant to operational fuel failures sometimes caused in zirconium alloy fuel rods by grid fretting and debris.

High Temperature Strength of Various Silicon Carbide Composites vs Zirconium Alloys

2. The Basic Concept - A key innovation introduced by CTP in the early phases of our work was to improve the strength of the multilayered tube with a unique fiber winding architecture and by pre-tensioning the fibers. Such a tube could withstand very high internal fission gas pressure...
prior to breaching the inner monolith tube. Several different fiber architectures were examined, leading to a unique fiber winding architecture and tensioning system. In addition, CTP added a third dense outer layer of monolithic SiC to enhance corrosion resistance. The design, called “triplex ceramic cladding,” is shown in the figure below.

The Three Layer Concept of CTP’s “Triplex SiC Cladding”

3. Key features of the technology

The SiC triplex clad design allows for independent optimization of the properties of the inner monolith for fission gas retention, the fiber-reinforced matrix for overall mechanical performance, and the outer monolith for corrosion resistance.

For example, on one particular triplex clad design, tests at Oak Ridge and MIT has demonstrated that the SiC triplex tube can withstand internal gas pressures of over 5000 psi, as compared to the maximum internal pressure of 2000 psi allowed in a zirconium alloy tube. This is an important capability that may allow higher burnup even with increased fission gas release.

Furthermore, during the mechanical tests at Oak Ridge, the triplex clad tube continued to retain its basic shape even after the inner monolith developed a crack resulting from high internal radial loads. Total strain on the cladding during internal loading exceeded 8% radial strain, while the composite layer retained its basic cylindrical shape, without gross fracture or ballooning.

The key features of the SiC triplex cladding that favor its further development and application to commercial water reactors include the following:

a. Accident tolerance – ballooning - Because the SiC clad would not balloon during design basis, and beyond design basis accidents, (see temperature vs strength characteristics in figure above) it would not block the ECCS flow designed to cool an overheated core.

b. Accident tolerance – oxidation and hydrogen release - In addition to the avoidance of ballooning and flow blockage, the SiC clad would not react exothermically and release
extensive volumes of hydrogen during design basis and beyond design basis accident, thus exacerbating the rapid heatup of the core, and creating the potential for hydrogen explosions. CTP has performed tests of the SiC up to 1400 °C in steam, to verify this characteristic.

c. **Resistance to Fretting and Debris Induced Clad failures** – SiC is one of hardest materials found in nature, and is used as an abrasive in many manufacturing industries. As such it would lead to a significant reduction in operational fuel failures in current LWRs that are caused by fretting of the clad tubes at the grid springs, and debris induced failures at the core inlet.

d. **High Burnup Potential and Extended Fuel Cycle Duration** – SiC cladding has the potential to extend burnup beyond the current regulatory limit of 62 gwd/t peak, because it does not become oxidized and embrittled with extended exposure as does advanced zirconium alloys. Exposure tests at MIT for over 3 full power years indicate the potential for 6 to 10 year exposure, depending on the thickness of the environmental barrier layer.

e. **High Power Density** – Because of its high temperature capability, up to 1500 °C, SiC cladding is likely to survive the brief Departure from Nuclear Boiling (DNB) Events (and similar dryout events with BWRs) that are prohibited with zircaloy cladding, and thus permit higher linear heat ratings and higher power densities. To take advantage of this cladding feature, it will be necessary to couple the cladding with an advanced fuel form, such as voided UO₂ pellets, BeO enhanced fuel pellets, or thoria-plutonia fuel pellets, such that the increased linear heat rating does not lead to excessive fuel temperatures.

4. **Use of Silicon Carbide Composites to Minimize Distortion in Reactor Core Components**

SiC triplex clad fuel rods are much stiffer than zircaloy clad fuel rods, and a fuel assembly of such SiC rods will be less prone to the fuel assembly distortion problems that have plagued many utilities during periodic refuelings. The increased stiffness may also allow reduction in the number of axial grids, further reducing cost, and parasitic neutron absorption in the core.

As a consequence, CTP is evaluating, under a recent DOE small business grant, the option of making the control rod guide tubes (PWRs only) from SiC composite materials. Preliminary evaluation indicates a significant reduction in Fuel Assembly distortion in the case where just the fuel rods have SiC cladding.

Another application we are examining in collaboration with the Electric Power Research Institute (EPRI), is the option of using SiC composites for the BWR channel boxes. Some estimates indicate that over half of the hydrogen gas released by the damaged cores at Fukushima came from oxidation of the zircaloy channel boxes.

5. **Remaining Technical Challenges**

So far, we have identified four major challenges that must be overcome before this technology can be sufficiently proven to move to the next steps involving full exposure in test reactors (such as ATR hot loops, NRU, or Halden), and subsequent testing of lead rods, and lead test assemblies in commercial reactors. CTP has just received a new Phase 2 Small Business grant from the DOE to address these challenges, under a two year program. This DOE sponsored
SBIR program will be done in collaboration with several organizations and consultants, including the MIT, several advanced ceramic firms in the US, the Babcock and Wilcox Company, and Dr. Shwageraus here in Israel. The challenges are:

a. Developing a **robust end joint design** including a bond agent to seal the joint between the fuel rod end cap and the clad tube. During early 2010, a number of different bond agents were irradiated in the MIT research reactor and five of six test specimens failed the test. The specimens used were flat SiC alpha phase blocks. Differential anisotropic swelling of the alpha phase blocks, and bond agent corrosion both contributed to the failures. Alternative bond agents are being formulated, and a new design joint geometry has been developed and will be tested as part of the proposed program.

b. Fabrication and testing of much longer clad tubes. Past development has been limited to triplex tubes about 12 inches long. Existing lab based process equipment are capable of fabricating tubes that are 3 to 4 ft long. Process vessels to fabricate 14 foot long tubes do not exist. In our Phase 2 project we will work with industry collaborators to develop new types of SiC tube fabrication equipment capable of being scaled up to 14 foot long processing equipment.

c. Modifying the fiber architecture of the composite layer to increase the **impact resistance** of the loaded fuel rod during fabrication and shipping. Further optimization of the fiber architecture, composite thickness, and/or the monolith to composite interface design, is required to provide adequate impact resistance. In addition to mechanical impact resistance, which we will test during the project, we will also conduct thermal shock tests to confirm that the triplex clad will continue to contain the solid fuel pellet fragments even after heating up to 1200 to 1600 C (as in a LOCA) and then being exposed to the thermal shock resulting from injection of cold water from the ECCS system.

d. Coupling the SiC cladding with a **modified fuel form** that minimizes the fuel temperatures during normal operation, and during anticipated operational occurrences. Standard UO$_2$ fuel clad with SiC triplex cladding will tend to run hotter than zircaloy clad fuel, and this could lead to increased fission gas release, and exceeding the allowable central temperature (close to melting) during transients and accidents. Although the triplex clad is designed to contain higher fission gas release and higher internal pressures as compared to zircaloy, the higher fuel temperatures will limit the potential benefits. For that reason, our project will examine the performance potential of three advanced fuel forms, when coupled with silicon carbide cladding.

One of the fuel forms being evaluated for use with SiC cladding, with the help of Dr. Shwageraus, is thoria plutonia fuel. Analyses conducted by Dr. Shwageraus in a previous DOE sponsored project indicate the potential for achieving up to 120 GWD/MTHM for such a design in a standard 1000 MW PWR reactor. Some of this improvement derives from the lower parasitic absorption of silicon carbide as compared to zircaloy. Much of it results from the increased fissile loading, and conversion of thoria to fissile U$^{233}$ during reactor operation. In any case, such high burnups would lead to a major reduction in spent fuel from nuclear plants. Such high burnups are not possible with zircaloy clad,
which oxidizes during operation limiting the burnup to about 50 GWD/MTHM batch average.

6. Performance Characteristics

Since 2001, CTP and its partners have tested seven different versions of Silicon Carbide fuel cladding, with different varieties of monolith tubes, fiber compositions, matrix infiltration methods, and environmental barrier coatings. Testing has been done in a prototype PWR coolant environment in the MIT research reactor, with some optimized clad specimens achieving over 30 full power months of exposure so far, and with exposure ongoing as of this date. Room temperature mechanical tests have been performed at the Oak Ridge National Laboratory, and at Ceramic Tubular Products facilities in Lynchburg, VA. In addition to the experimental work, there has been significant modeling and analysis to predict the performance of the cladding and contained fuel in a commercial environment, including work sponsored by the Electric Power Research Institute to determine how best to incorporate the triplex cladding into a typical PWR core design and fuel management cycle. And recently, a test has begun in the Oak Ridge High Flux Isotope Reactor (HFIR) to operate commercially fabricated uranium fuel pellets with silicon carbide triplex cladding under typical average heat ratings found in today’s commercial reactors. The purpose of this test is to evaluate any possible pellet clad interaction that might occur in the fuel rod during operation. This test is supported with DOE funds allocated from DOE’s LWR Sustainability Program, and is planned to continue for several more years.

Some positive output from this development and testing to date include:

- The recession rate of the outer barrier layer during the initial 20 months of exposure in the MIT reactor, extrapolates to a life of 6 to 10 years, enough to support the high burnup objectives of the proposed technology demonstration program.

- The combination of strong monolith and tightly wound and infiltrated central composite layer will support very high internal gas pressures, 2 to 3 times the level that can be achieved with zircaloy clad fuel. Zircaloy cladding creeps under pressure and therefore must operate with a fission gas pressure below plant operating pressure of 2000 psi. Silicon Carbide clad does not creep and therefore can sustain much higher fission gas release which occurs with high burnup.

- Exposure to the high radiation environment in the MIT reactor modestly reduces the strength of the cladding material. Less than 20% for the case of the round 6 Coorstek monolith specimens. A set of tubes briefly irradiated at MIT, similar to the clad tubes used at HFIR, and known as round 7, had a larger strength reduction during irradiation, as reported in a recent MIT CANES report. We believe this was due to the inferior nature of the monolith tubes in the round 7 specimens which contained laminations, and will not be used in future triplex tube production.

- When a triplex clad is mechanically loaded to failure, either due to pellet swelling or high gas pressure, the composite layer maintains its shape even after the inner monolith fails. This would assure, under severe accident conditions, that the solid fuel would be retained
and not released to the coolant, even after the fission gas is released due to the initial failure.

7. Economic Advantages

Silicon carbide retains its strength to very high temperatures, above 1500 C, and thus would be DNB proof allowing much higher linear heat ratings as compared to zircaloy clad fuel. Also, as noted above, a modified fuel form would be required to take advantage of this cladding characteristic – for example voided UO₂, BeO enhance UO₂ or Thoria-MOX. When coupled with these fuel forms, linear heat ratings at least 20% greater than current LWR designs appear feasible.

MIT evaluated the economic implications of this higher heat rating capability in an EPRI sponsored research project, as reported in MIT CANES report “Feasibility and Economic Benefits of PWR Cores with Silicon Carbide Cladding” April, 2011, MIT-ANP-PR-134. One case examined in this study showed that a 20% increase in power rating would lead to a peak linear heat rating of 21 kw/ft, which for a solid UO₂ pellet would lead to centerline melting in certain transient cases. That same scenario, with the same linear heat rating, would have acceptable centerline temperatures for the case of SiC clad UO₂ with a 10% central void. This report went on to analyze various PWR core operating scenarios to take advantage of this capability, and concluded there would be major economic advantages. Because of DNB concerns, this capability would not be achievable with zircaloy clad fuel.

8. Transportation and Storage

Irradiated LWR fuel with SiC clad will be much more durable, and less prone to failure in transportation, storage, and ultimate disposal, than the brittle, irradiated, zircaloy cladding is today. The potential for overheating and severe chemical reaction in the event of a fuel pool loss of coolant accident, as was feared in the recent Fukushima accident, would be largely eliminated. However, the use of higher enriched UO₂ fuel, or thoria plutonia fuel, will require a review of existing license conditions, and in some cases, modifications to the transportation, handling and storage equipment.
INTRODUCTION
In this paper we discuss Canada’s response to the March 2011 events at Fukushima, from a nuclear regulatory point of view, including emergency management. We will also describe the actions taken by the Government of Canada and the challenges with obtaining timely and complete information. Finally, three specific actions taken by the CNSC, namely the establishment of a Fukushima Task Force, the creation of an External Advisory Committee and the results of an IAEA review of the Canadian Nuclear Safety Commission (CNSC) response to the events at Fukushima are summarized.

NUCLEAR EMERGENCY RESPONSE IN CANADA
The management of a nuclear emergency in Canada involves overlapping municipal, provincial and federal jurisdictions:

- Onsite preparedness and response is the responsibility of the nuclear power plant (NPP) licensee
- Offsite preparedness and response is the responsibility of the province, in coordination with municipalities, where the NPP is located
- At the request of the province, the Government of Canada will provide support to the province via the resources of multiple federal agencies
- The CNSC continues to have regulatory oversight of the licensee during an emergency

OFF-SHORE VERSUS DOMESTIC NUCLEAR EMERGENCIES
The primary goals of nuclear emergency management for off-shore nuclear emergencies differ somewhat from the objectives during the management of a domestic nuclear emergency. During the Fukushima events, the primary objectives were:

- Protection of Canadians in Japan
- Protection of Canadians in Canada
- Protection of the Canadian environment
- Evaluating any immediate implications for the Canadian reactor fleet

ACTIONS TAKEN BY THE CNSC
While major accidents such as this happen very infrequently, it is vitally important that all nuclear facility designers and operators, nuclear regulators, and emergency response organizations learn every possible lesson. Recognizing the severity of the accident and the need
for the national regulatory authority to lead, monitor, advise and communicate with both the public, decision-makers and other stakeholders, the CNSC responded immediately to the accident at Fukushima Daiichi with the following actions:

- Activated its Emergency Operations Centre in Ottawa and staffed it 24/7 to monitor the emergency, assess early reports and provide timely, accurate information to Canadians and to other Canadian government departments and agencies
- Requested licensees of Canadian Class I nuclear facilities, under section 12(2) of the General Nuclear Safety and Control Regulations, to review the lessons learned from the Fukushima Daiichi accident
- Performed inspections of all NPPs and other nuclear facilities in Canada to assess the readiness of mitigating systems – these inspections covered seismic preparedness, firefighting capability, backup power sources, hydrogen mitigation and irradiated fuel bay cooling
- Established a Task Force to evaluate the operational, technical and regulatory implications of the accident and the adequacy of emergency preparedness for NPPs
- To aid the Canadian response and advise Canadian citizens in Japan, the CNSC performed source-term calculations and enhanced its public communications, which included daily information updates
- Participated in inter-governmental discussions at all levels
- On the national level, Canada conducted ongoing radiation monitoring across all of its territory and deployed experts to the IAEA
- On the international level, Canada worked closely with other national nuclear regulators (including the United States, the United Kingdom, and France) to share and to validate information.

In addition, the CNSC commissioned an External Advisory Committee to assess the regulatory response and the CNSC was the first nation to request a Fukushima module as part of an International Atomic Energy Agency (IAEA) Integrated Regulatory Review Service (IRRS) mission.

**COMMUNICATIONS**

Fukushima presented some particular information challenges, specifically in getting information early in the event so that modeling assumptions could be verified. There were conflicting reports from media, international agencies and from various sources in Japan. Also, there were challenges in getting information processed, verified and approved so that public communications could be made.

And finally, it is worth noting that Fukushima was the first major nuclear accident to occur during “the internet age”. The speed at which information, and in some cases misinformation, was posted to the internet greatly increased the demands on emergency management and communications staff as they attempted to review and either confirm or refute the information.
**FUKUSHIMA TASK FORCE**

The CNSC Fukushima Task Force was created with the objective of reviewing the capability of NPPs in Canada to withstand conditions similar to those that triggered the Fukushima accident. Specifically, the CNSC Task Force examined the response of NPPs to external events of higher magnitude than have previously been considered. It also examined the licensees’ capability to respond to such events. The focus was on the need to “anticipate the unexpected”: events such as earthquakes, tornadoes or hurricanes that may cause a prolonged loss of electrical power, resulting in operators not being able to continue cooling the reactors. The focus was also on the need for an integrated response capability.

In the process of formulating the safety review criteria, the CNSC Task Force considered all the applicable lessons learned to date from the Fukushima accident and reviewed selected international reports to ensure that all aspects relevant to Canada were addressed. Effectively, the CNSC Task Force has subjected the Canadian NPPs, the existing emergency response measures, and the regulatory framework and supporting processes to a systematic and comprehensive “stress test” to evaluate means to further protect the health and safety of Canadians and the environment. The post-Fukushima review has examined events more severe than those that have historically been regarded as credible and their impact on the NPPs. The CNSC Task Force has proposed changes to designs or procedures, wherever gaps were found, in order to minimize or eliminate their impact. Specifically, the CNSC Task Force made recommendations for:

- Strengthening reactor defence in depth
- Enhancing emergency response
- Improving regulatory framework and licensing

Overall, the CNSC Task Force concluded that Canadian NPPs are safe and that the risk posed to the health and safety of Canadians or to the environment is small. The CNSC staff have also verified that all Canadian NPPs are located far from tectonic plate boundaries and that the threat of a major earthquake at a Canadian NPP is negligible.

The CNSC Task Force is confident that the improvements recommended will further enhance the safety of nuclear power in Canada and will reduce the associated risk to as low as reasonably practicable.

Under the oversight of the CNSC and its staff, Canadian NPPs have been operating safely for over 40 years. As has always been the case, they will only be licensed if the CNSC is satisfied that they will continue to be operated safely.

**EXTERNAL ADVISORY COMMITTEE**

The Canadian Nuclear Safety Commission (CNSC) also established an external advisory committee to assess the organization’s processes and responses in light of the lessons learned from the Fukushima nuclear incident, which has highlighted the importance of nuclear safety around the world.

Committee members reviewed the CNSC’s processes including the immediate response to the Fukushima incident, its connections with the rest of government and international organizations.
and its interactions with the Canadian nuclear sector and its regulated industries. They also reviewed the CNSC’s communications with affected stakeholders, including governments, other nuclear regulators, and the public. Finally the committee assessed the implications on the CNSC’s regulatory approaches resulting from the international response to Fukushima, such as international stress tests and the International Atomic Energy Agency action plan.

IRRS FUKUSHIMA REVIEW\(^{(3)}\)
The IRRS Fukushima review was completed in December 2011. The review found strength in the CNSC response; our response was found to be prompt, robust and comprehensive.

A good practice was identified: CNSC completed a systematic and thorough review of the lessons learned from the accident, making full use of all the available information including the review of actions taken by other international regulators. The CNSC was found to have set up an action plan for addressing all the finding and recommendations of the Fukushima Task Force.

Two recommendations and a suggestion were offered. It was recommended that a national assessment of off-site emergency plans be undertaken, and that periodic full scale exercises are re-instituted (the last one having occurred in 2007). A suggestion was offered that a peer review of the Canadian nuclear emergency preparedness and response be undertaken. All three of these findings were directed at the Government of Canada, as opposed to the CNSC.

CONCLUSIONS
The events at Fukushima have led all international regulators and licensees to review their readiness for external events. While no major changes to the Canadian nuclear regulatory system have been identified, the various findings from the CNSC Fukushima Task Force, along with the recommendations of the External Advisory Committee and the IRRS Mission are being reviewed and incorporated into the ongoing CNSC regulatory improvements program.

REFERENCES
Innovation Priorities in Nuclear and Radiation Technologies in Russia. View from Skolkovo

A. Fertman, D. Kovalevich, V. Turtikov, N. Zaytseva

Skolkovo foundation, Nuclear technology cluster

The direction for the modernization and technological development of “Nuclear Technologies” sector of the Russian economy comprises a group of scientific and engineering subjects (atomic engineering, technologies on the basis of radiation, change of properties of materials, radiation resistant microelectronics, etc.), and serves as the foundation of one of the most high-tech industries. The innovative development of nuclear technologies is an integral condition for the strengthening (and in some directions of conquering) a country’s position as a global technological leader and preservation of defensive capability of the nation. For this reason, nuclear technologies became one of the priority areas for the activity of the Skolkovo Center. The wide opportunities offered by the application of nuclear technologies were already clear at the deployment stage of the “Nuclear Project – 1”. In 1958, at the 2nd International conference on the peaceful use of nuclear energy in Geneva, the USSR presented more than 200 reports and communiqués in all civil use of atomic energy directions.

One of the major results of the development of the nuclear branch have become the developments in the sphere of control of radiation and magnetic fields (radiation technologies). This group of technologies have actively developed in collaboration with design and manufacturing of different types of equipment, including accelerators, neutron generators, lasers, HF-systems, detectors of particles and radiation, microscopes and telescopes, microwave microelectronics, etc. Today these technologies and equipment are used in a variety of other (non-power and not military) markets – and the list of these markets grows constantly. Among the fastest growing ones, we can list the markets of nuclear medicine, sterilization and disinfection, safety and non-destructive testing, ecology and water processing, extraction and the processing of minerals.

Historically, the development of nuclear technologies demands the carrying out of multidisciplinary research. Physicists, chemists, mathematicians and engineers actively co-operated in the “Nuclear Project” for transition to a peaceful use of nuclear energy, later being joined by power engineers. In another example, the rapid development of nuclear medicine in the last decade was a result of the combined efforts of physicists, engineers, biologists, doctors and programmers. Therefore, it is not by chance, that exactly such experts will be involved in Skolkovo’s Biomedical, IT and Nuclear Technologies Clusters.
The New US Public-Private Partnership to License and Deploy Small Modular Reactors, With Focus on The B&W mPower™ Reactor

Herbert Feinroth\textsuperscript{1}, Douglas Lee\textsuperscript{2}

\textsuperscript{1}CEO, Ceramic Tubular Products LLC, Rockville, Maryland; Lynchburg, Virginia
\textsuperscript{2}B&W Nuclear Energy, Inc., Lynchburg, Virginia

Corresponding Author: hfeinroth@gamma-eng.com

On December 16, 2011, The US Congress and the President approved new Fiscal Year 2012 funding for a Government – Industry cost shared program called “Small Modular Reactor (SMR) Licensing Technical Support.” The new legislation appropriates $67 million in 2012 to provide licensing and first-of-a-kind engineering support for small modular reactor designs that can be deployed expeditiously. The legislation requires the Department of Energy to consider applications utilizing any small modular reactor technology. Competitive solicitations are likely to begin shortly and two or three SMR designs will be selected for US Government support. Such support will likely accelerate deployment and operation of at least one such design by 2020. The Congressional language states that the Government portion of the program is expected to total $452 million over five years.

One of the candidates for this competition is the B&W mPower™ reactor being developed by Generation mPower, a company recently formed by the Babcock & Wilcox Company and Bechtel Power Corporation. This presentation will summarize the main features of this design, and explain why it meets the requirements for the Government program, and will be fully developed, licensed and deployed in the US within the next 8 years. Importantly, this design has many features that favor its introduction and use in smaller countries with critical needs for future electric generation capacity, with arid conditions that may require air cooled condensers, and with potential need for a desalination component of the new energy source. The relatively small capacity of the modules (e.g. 320 MWe for an initial two unit plant) will require much lower initial capital investment, as compared to the very large investment of $4 to $6 billion required for the newer 1100 to 1400 MWe plants now being constructed in China, France, Finland, Korea, the US, and the United Arab Emirates.

The presentation will summarize the main features of the mPower plant and project including:

1. Its passive safety design including features to assure core integrity even in the event of extreme natural events such as occurred at Fukushima in March, 2011, or unlikely operator errors such as occurred at Three Mile Island in 1979.

2. Its modularity allowing parallel construction and shipment of the main reactor module, while site construction of balance of plant and containment proceeds in parallel, leading to a much shorter schedule from start of construction to initial operation. Estimates are on
the order of 4 years or less in the US – from issuance of either a construction permit under Part 50, or a design certification and Combined Operating License under Part 52.

3. Its reliance on the same basic fuel technology, and light water coolant, that has proven successful for the vast majority of the world’s current civilian nuclear power plants. Also, the integral reactor design developed by B&W was successfully demonstrated in the merchant ship Otto Hahn.

4. Its enhanced security, low profile plant design maximizes the ability to withstand external events such as a direct impact of a large airplane. The reactor containment structure is located below ground level, and is designed to withstand such challenges without release of radiation.

Of special significance to countries such as Israel, are the optional design features of the generation mPower plant that allow siting of the plant in arid regions (using air cooled condensers), and that include a desalination unit supplied with steam from the reactor plant or electricity from the generator.

Generation mPower, formed by B&W and Bechtel, has been working on the design and licensing of the mPower design for over three years, and has made remarkable progress. A new, 30 tonne, two MW (electrically heated) prototype reactor module has been installed at B&W’s Integrated System Test Facility near Lynchburg Virginia, and is just beginning a test program designed to provide critical data on reactor performance as required for NRC licensing. The facility will also be available for future operator training.

The Tennessee Valley Authority (TVA) has embraced the mPower reactor technology by signing a letter of intent for up to six reactors at its Clinch River site in Tennessee. TVA plans to submit a Construction Permit Application for the first two units to the NRC by the end of 2013.

The author will conclude his presentation with brief historical anecdotes regarding the first commercial reactor, which began operation in 1957 at Shippingport, Pennsylvania, and which had a power rating similar to an mPower module (150 Mwe for the second core). The author worked on the design and initial operation of this reactor, alongside a well known US – Israeli citizen, Alvin Radkowsy, who was largely responsible for conceiving of, and overseeing the nuclear design, including the seed-blanket concept and the first Thorium light water breeder reactor.
High-Order Metrics for Model Uncertainty Quantification and Validation

Dan G. Cacuci
IKR/KIT, Gotthard-Franz-Str. 9, 96131 Karlsruhe, Germany

Corresponding Author: dan.cacuci@kit.edu

INTRODUCTION

It is well known that the true values of measured and computed data are impossible to know exactly because of various uncontrollable errors and uncertainties arising in the data measurement and interpretation reduction processes. Hence, all inferences, predictions, engineering computations, and other applications of measured and/or computed data are necessarily based on weighted averages over the possibly true values, with weights indicating the degree of plausibility of each value. Furthermore, combination of data from different sources involves a weighted propagation (e.g., via sensitivities) of all uncertainties, requiring reasoning from incomplete information and using probability theory for extracting optimal values together with “best-estimate” uncertainties from often sparse, incomplete, error-afflicted, and occasionally discrepant data. The current state-of-the-art data assimilation/model calibration methodologies for large-scale nonlinear systems cannot take into account uncertainties higher-order than second-order (i.e., covariances) thereby failing to quantify fully the deviations of the problem under consideration from a normal (Gaussian) multivariate distribution. Such deviations would be quantified by the third- and fourth-order moments (skewness and kurtosis) of the model’s predicted results (responses). These higher-order moments would be constructed by combining modeling and experimental uncertainties (which also incorporate the corresponding skewness and kurtosis information), using derivatives of the model responses with respect to the model’s parameters. This paper presents explicit expressions for skewness and kurtosis of computed responses, thereby permitting quantification of the deviations of the computed response uncertainties from multivariate normality. In addition, this paper presents a new and most efficient procedure for computing the second-order response derivatives with respect to model parameters using the “adjoint sensitivity analysis procedure” (ASAP).

RESULTS

Consider that the model under consideration is used to compute $N_r$ responses (or results), denoted generically by the vector $r = \{ r_i | i = 1, \ldots, N_r \}$. In general, the model parameters are experimentally derived quantities and are therefore subject to uncertainties. Specifically, consider that the model comprises $N_\alpha$ uncertain parameters $\alpha_n$, which constitute the components of the (column) vector $\alpha$ of model parameters, defined as $\alpha = \{ \alpha_n | n = 1, \ldots, N_\alpha \}$. In practice, the mean values of the model parameters are known together with uncertainties (correlations and standard deviations) computed about the respective mean values. The vector $\alpha^0 = \{ \alpha^0_n | n = 1, \ldots, N_\alpha \}$ of mean values of the model parameters has components denoted as $\alpha^0_i \equiv \{ \alpha^0_{ij} \}$. The angular
brackets, e.g., \( \langle f \rangle \equiv \int f(a, r) p(a, r) \, da \, dr \), denote integration of a generic function \( f(a, r) \) over the unknown joint probability distribution \( p(a, r) \) of parameters and responses. The parameter correlations are defined as \( c_{ij}^\alpha \equiv \left\{ (\alpha_i - \alpha_i^0)(\alpha_j - \alpha_j^0) \right\} \). It is very unlikely that higher-order (i.e., triple, quadruple) correlations among parameters would be available in practice; nevertheless, the skewness \( \mu_3(\alpha_i) \equiv \left\{ (\alpha_i - \alpha_i^0)^3 \right\} \) and kurtosis \( \mu_4(\alpha_i) \equiv \left\{ (\alpha_i - \alpha_i^0)^4 \right\} \) for individual parameters \( \alpha_i \) could be obtained if the forms of the corresponding individual probability distributions are known or can be approximately surmised.

The most general representation of a the vector of responses, \( r \), as a function of the parameters, \( \alpha \), is \( r = a r^c (\ ) \), where \( r^c(\ ) \) denotes the computed response value for a given, but otherwise arbitrary, set of numerical values for the parameters \( \alpha \). The deterministic methods for propagating uncertainties in model parameters to computed responses rely on expanding formally the computed response in a Taylor series around \( \alpha^0 \). In practice, the exact first-order response derivatives ("sensitivities") with respect to model parameters can be computed most efficiently for large-scale nonlinear systems by using the "adjoint sensitivity analysis procedure" (ASAP), as generally shown by Cacuci\(^2\). However, the computation of the second-order derivatives severely strains computational resources, while third- and higher-order response derivatives with respect to model parameters are practically unavailable for large-scale systems. Reflecting these practicalities, only second-order derivatives of responses to model parameters will be explicitly taken into account in this work, to obtain:

(i) the expected value, \( E(r_k) \), of the response \( r_k \):

\[
E(r_k) = r_k^c(\alpha^0) + \frac{1}{2} \sum_{i=1}^{N_x} \sum_{j=1}^{N_x} \frac{\partial^2 r_k(\alpha^0)}{\partial \alpha_i \partial \alpha_j} \text{cov}(\alpha_i, \alpha_j).
\]

(ii) the covariances \( \text{cov}(r_k, r_i) \equiv \left[ r_k - E(r_k) \right]\left[ r_i - E(r_i) \right] \):

\[
\text{cov}(r_k, r_i) = \sum_{i=1}^{N_x} \sum_{j=1}^{N_x} \left( \frac{\partial r_k}{\partial \alpha_i} \frac{\partial r_i}{\partial \alpha_j} \right) \text{cov}(\alpha_i, \alpha_j) + \frac{1}{2} \sum_{i=1}^{N_x} \left( \frac{\partial r_i}{\partial \alpha_i} \frac{\partial^2 r_i}{\partial \alpha_i^2} + \frac{\partial r_i}{\partial \alpha_i} \frac{\partial^2 r_k}{\partial \alpha_i \partial \alpha_j} \right) \mu_3(\alpha_i) + \frac{1}{4} \sum_{i=1}^{N_x} \left( \frac{\partial^2 r_k}{\partial \alpha_i^2} \frac{\partial^2 r_i}{\partial \alpha_i^2} \right) \mu_4(\alpha_i) + \frac{1}{4} \sum_{i=1}^{N_x} \sum_{j=1}^{N_x} \sum_{k=1}^{N_x} \sum_{\mu=1}^{N_x} \sum_{\nu=1}^{N_x} \left( \frac{\partial^2 r_k}{\partial \alpha_{\mu} \partial \alpha_{\nu}} \right) \text{cov}(\alpha_{\mu}, \alpha_{\nu}) \left( \frac{\partial^2 r_i}{\partial \alpha_{\mu} \partial \alpha_{\nu}} \right) \text{cov}(\alpha_i, \alpha_j) \text{cov}(\alpha_i, \alpha_j);
\]

(iii) the skewness \( \mu_3(r_k) \equiv \langle [r_k - E(r_k)]^3 \rangle \)
\[ \mu_3(r_k) = \frac{1}{2} \sum_{i=1}^{N_\alpha} \left( \frac{\partial^2 r_k}{\partial \alpha_i^2} \right) \mu_4(\alpha_i) \left[ 3 \left( \frac{\partial r_k}{\partial \alpha_i} \right)^2 - \frac{3}{4} \left( \frac{\partial^2 r_k}{\partial \alpha_i^2} \right) \sum_{\mu=1}^{N_\alpha} \sum_{\nu=1}^{N_\alpha} \left( \frac{\partial^2 r_k}{\partial \alpha_\mu \partial \alpha_\nu} \right) \text{cov}(\alpha_\mu, \alpha_\nu) \right] \]

\[ + \sum_{i=1}^{N_\alpha} \left( \frac{\partial r_k}{\partial \alpha_i} \right) \mu_3(\alpha_i) \left[ \left( \frac{\partial r_k}{\partial \alpha_i} \right)^2 - \frac{3}{2} \left( \frac{\partial^2 r_k}{\partial \alpha_i^2} \right) \sum_{i=1}^{N_\alpha} \sum_{\nu=1}^{N_\alpha} \left( \frac{\partial^2 r_k}{\partial \alpha_\nu \partial \alpha_i} \right) \text{cov}(\alpha_\nu, \alpha_i) \right] \]

\[ - \frac{3}{2} \left[ \sum_{i=1}^{N_\alpha} \sum_{j=1}^{N_\alpha} \left( \frac{\partial r_k}{\partial \alpha_i} \frac{\partial r_k}{\partial \alpha_j} \right) \text{cov}(\alpha_i, \alpha_j) \right] \]

\[ + \frac{1}{4} \left[ \sum_{i=1}^{N_\alpha} \sum_{j=1}^{N_\alpha} \left( \frac{\partial^2 r_k}{\partial \alpha_\mu \partial \alpha_\nu} \right) \text{cov}(\alpha_\mu, \alpha_\nu) \right]^3 \]

(iv) the kurtosis \( \mu_4(r_k) \equiv \left[ \text{E}(r_k) - \text{E}(r_k) \right]^4 \):

\[ \mu_4(r_k) = \sum_{i=1}^{N_\alpha} \mu_4(\alpha_i) \left[ \left( \frac{\partial r_k}{\partial \alpha_i} \right)^4 + \frac{3}{8} \left( \frac{\partial^2 r_k}{\partial \alpha_i^2} \right)^2 \left[ \sum_{\mu=1}^{N_\alpha} \sum_{\nu=1}^{N_\alpha} \left( \frac{\partial^2 r_k}{\partial \alpha_\mu \partial \alpha_\nu} \right) \text{cov}(\alpha_\mu, \alpha_\nu) \right] \right] \]

\[ - \frac{3}{2} \left( \frac{\partial r_k}{\partial \alpha_i} \right)^2 \left[ \sum_{\mu=1}^{N_\alpha} \sum_{\nu=1}^{N_\alpha} \left( \frac{\partial^2 r_k}{\partial \alpha_\mu \partial \alpha_\nu} \right) \text{cov}(\alpha_\mu, \alpha_\nu) \right] \]

\[ \times \left[ \sum_{i=1}^{N_\alpha} \sum_{j=1}^{N_\alpha} \left( \frac{\partial^2 r_k}{\partial \alpha_\mu \partial \alpha_\nu} \right) \text{cov}(\alpha_\mu, \alpha_\nu) \right] \]

\[ - \frac{3}{16} \left[ \sum_{\mu=1}^{N_\alpha} \sum_{\nu=1}^{N_\alpha} \left( \frac{\partial^2 r_k}{\partial \alpha_\mu \partial \alpha_\nu} \right) \text{cov}(\alpha_\mu, \alpha_\nu) \right]^4 \]

Note that all of the derivatives appearing in the above expressions are evaluated at the nominal parameter values \( \alpha^0 \). Note also that the quantity

\[ q_k(\rho) \equiv \sum_{i=1}^{N_\alpha} \sum_{j=1}^{N_\alpha} \frac{\partial^2 r_k}{\partial \alpha_i \partial \alpha_j} \text{cov}(\alpha_i, \alpha_j) = \sum_{i=1}^{N_\alpha} \sum_{j=1}^{N_\alpha} \frac{\partial^2 r_k}{\partial \alpha_i \partial \alpha_j} \rho_{ij} \sigma_i \sigma_j \]

where \( \rho_{ij} \) denotes the correlation between parameters \( \alpha_i \) and \( \alpha_j \), while \( \sigma_i \) and \( \sigma_j \) denote the standard deviations of the respective parameters, appears repeatedly in the expressions of \( \text{E}(r_k) \), \( \text{cov}(r_k, r_k) \), \( \text{var}(r_k) \), \( \mu_3(r_k) \) and \( \mu_4(r_k) \). In general, the computation the mixed second-order response-derivatives \( \partial^2 r_k / \partial \alpha_i \partial \alpha_j \) would require \( O(N_\alpha^2) \) large-scale computations using the original (forward) model for every response \( r_k(\alpha^0) \). In the special case when the parameters \( \alpha \) are fully correlated (i.e., all \( \rho_{ij} = 1 \)), however, the quantity \( q_k(\rho) \equiv 1 \) can be computed very efficiently by following the following novel procedure. By considering the (column) vector of
standard deviations $\sigma \equiv (\sigma_1, \sigma_2, \ldots, \sigma_{N_{\alpha}})$, and by noting that the Hessian-vector product $\nabla^2 r_{\alpha} \left( \theta^0 \right)$ can be quantified most efficiently by using two computations of the gradients $\nabla_{\alpha} (r_k)$, as follows: $\nabla^2 r_{\alpha} \left( \theta^0 \right) \equiv b^{-1} \left[ \nabla_{\alpha} \left[ \varphi \left( \theta^0 + \phi \theta^0 \right) \right] - \nabla_{\alpha} \left[ \varphi \left( \theta^0 \right) \right] \right] \cdot \varphi \left( \theta^0 \right)$, $k = 1, \ldots, N_r$, where $b$ is a small scalar quantity. As generally shown by Cacuci$^2$, the exact computation of the gradients $\nabla_{\alpha} (r_k)$ can be computed most efficiently using the ASAP (one adjoint model computation per response), yields all of the gradients $\nabla_{\alpha} (r_k)$ with respect to the parameters $\alpha$. Subsequently, the vector $y_{\alpha} \left( \theta^0 \right)$ can be obtained using two adjoint-model computations instead of $O(N_{\alpha})^2$ large-scale computations using the original (forward). Once $y_{\alpha} \left( \theta^0 \right)$ has been obtained, the quantity $q_k (\rho_{ij} = 1)$ can be obtained by an additional vector scalar-product computation of the form $q_k (\rho_{ij} = 1) = \sigma^i y_k \left( \theta^0 \right)$. In practice, however, the model parameters $\alpha$ are extremely unlikely to be fully correlated. More likely, they are partially (positively or negatively) correlated or uncorrelated. In such cases, it will be shown in the oral presentation that is still possible to reduce the number of computations required for obtaining the quantity $q_k (\rho_{ij})$, from $O(N_{\alpha})^2$ to at most $n_k \leq N_{\alpha}$ operations for each response $r_k \left( \theta^0 \right)$, by using Hessian-vector products of the same form as shown in the forgoing.

CONCLUSIONS

This paper has presented, in premiere, explicit expressions for skewness and kurtosis of computed responses, thereby permitting quantification of the deviations of the computed response uncertainties from multivariate normality. Applications to large-scale reactor physics problems, using the peta-flop JAGUAR computer at Oak Ridge National Laboratory (USA) will be discussed in the oral presentation.

REFERENCES


No-Failure Design and Disaster Recovery: Lessons from Fukushima

Yakov Ben-Haim

1Yitzhak Moda'i Chair in Technology and Economics, Faculty of Mechanical Engineering, Technion-Israel Institute of Technology, Haifa 32000 Israel,

Corresponding Author: yakov@technion.ac.il

One of the striking aspects of the early stages of the nuclear accident at Fukushima-Daiichi last March was the nearly total absence of disaster recovery capability. For instance, while Japan is a super-power of robotic technology, the nuclear authorities had to import robots from France for probing the damaged nuclear plants. Fukushima can teach us an important lesson about technology.

The failure of critical technologies can be disastrous. The crash of a civilian airliner can cause hundreds of deaths. The meltdown of a nuclear reactor can release highly toxic isotopes. Failure of flood protection systems can result in vast death and damage. Society therefore insists that critical technologies be designed, operated and maintained to extremely high levels of reliability. We benefit from technology, but we also insist that the designers and operators "do their best" to protect us from their dangers.

Industries and government agencies who provide critical technologies almost invariably act in good faith for a range of reasons. Morality dictates responsible behavior, liability legislation establishes sanctions for irresponsible behavior, and economic or political self-interest makes continuous safe operation desirable.

The language of performance-optimization - not only doing our best, but also achieving the best - may tend to undermine the successful management of technological danger. A probability of severe failure of one in a million per device per year is exceedingly - and very reassuringly - small. When we honestly believe that we have designed and implemented a technology to have vanishingly small probability of catastrophe, we can honestly ignore the need for disaster recovery.

Or can we?

Let's contrast this with an ethos that is consistent with a thorough awareness of the potential for adverse surprise. We now acknowledge that our predictions are uncertain, perhaps highly uncertain on some specific points. We attempt to achieve very demanding outcomes - for instance vanishingly small probabilities of catastrophe - but we recognize that our ability to reliably calculate such small probabilities is compromised by the deficiency of our knowledge and understanding. We robustify ourselves against those deficiencies by choosing a design that would be acceptable over a wide range of deviations from our current best understanding. (This is called "robust-satisficing".) Not only does "vanishingly small probability of failure" still entail the possibility of failure, but our predictions of that probability may err.
Acknowledging the need for disaster recovery capability (DRC) is awkward and uncomfortable for designers and advocates of a technology. We would much rather believe that DRC is not needed, that we have in fact made catastrophe negligible. But let's not conflate good-faith attempts to deal with complex uncertainties, with guaranteed outcomes based on full knowledge. Our best models are in part wrong, so we robustify against the designer's bounded rationality. But robustness cannot guarantee success. The design and implementation of DRC is a necessary part of the design of any critical technology, and is consistent with the strategy of robust satisficing.

One final point: moral hazard and its dilemma. The design of any critical technology entails two distinct and essential elements: failure prevention and disaster recovery. What economists call a 'moral hazard' exists because the failure prevention team might rely on the disaster-recovery team, and vice versa. Each team might, at least implicitly, depend on the capabilities of the other team, and thereby relinquish some of its own responsibility. Institutional provisions are needed to manage this conflict.

The alleviation of this moral hazard entails a dilemma. Considerations of failure prevention and disaster recovery must be combined in the design process. The design teams must be aware of each other, and even collaborate, because a single coherent system must emerge. But we don't want either team to relinquish any responsibility. On the one hand we want the failure prevention team to work as though there is no disaster recovery, and the disaster recovery team should presume that failures will occur. On the other hand, we want these teams to collaborate on the design.

This moral hazard and its dilemma do not obviate the need for both elements of the design. Fukushima has taught us an important lesson by highlighting the special challenge of high-risk critical technologies: design so failure cannot occur, and prepare to respond to the unanticipated.
Beams operations of the SARAF accelerator in 2011


Soreq Nuclear Research Center, Yavne 81800 Israel

Corresponding Author: weissman@soreq.gov.il

INTRODUCTION

The first phase of the SARAF accelerator is in operation at the Soreq Nuclear Center. Figure 1 shows a schematic diagram of component sof SARAF phase I. There is a few years window when the Phase I of the project will be operational before installation of Phase-II of the linac. It was realised quite early that intense but relatively low-energy beams at Phase I will provide some unique possibilities for various types of research. Therefore even given the present accelerator specifications [1] there is strong demand for the accelerator beam time. There are several major experiments demanding major beam time allocation, as well as numerous minor experiments and tests which require modest beam time.

In 2011 for the first time the delivery of beams to the users becomes the primary task. All activities for accelerator commissioning and improvements of its subsystem were performed in the background mode. A temporary beam line has been built (Fig. 1) in the accelerator hall in order to facilitate beam transport for different experiments with broad range of requirements. The new installations and further commissioning of the accelerator were done as secondary tasks.

The main logistic difficulty of beam operation is associated with the fact that the temporary beam line and the experimental setups are situated in the accelerator hall. This significantly reduces the time of accelerator operation due to the required activities in the hall. In addition, poor quality of the electrical power, lack of the technical personal and deterioration with time of some of the accelerator components affected efficiency of the beam operation.

Despite of the difficulties listed above we have used any opportunity to operate the accelerator and gained significant experience in beam operations. In this report the main beam operation activity is summarized and the prospects for the next future are discussed.
SUMMARY OF ACTIVITIES

Experiments with thin foil targets

During 2011, the ~3.5 MeV proton beams were mainly used for tests of thin foil targets cooled by liquid metal. The maximum current that was delivered to these targets was about 0.3 mA. The thin foil targets are delicate. There were several incidents where the targets were damaged and vacuum in the beam line was compromised. Such incidents were foreseen and appropriate protection instruments, including a fast valve, were installed in the beam line. The problems causing the foil targets failure are being analyzed. Improvements of the target setup are under consideration. We expect significant progress on this subject in the near future. In 2011 the tests with the foil targets were the main driving force for improvements in beam operation.

First steps towards astrophysics experiments

One of the main projects for the next future will be production of intense neutron source by impinging intense proton beam on the Liquid Lithium Target (LiLiT [2]). Choosing the beam energy just above the $^7$Li(p,n) reaction threshold (1.880 MeV) allows for production neutron energy spectrum simulating the neutron flux in the stellar environment. The broad experimental program based on such neutron source is being prepared. As a first step in that direction several tests were performed with evaporated LiF target. The target setup was identical to the one used for many years at Karlsruhe [3]. The goals of the tests were: 1. Accurate measurements of beam properties (energy and spread), 2. Study of obtained neutron spectra, 3. Gaining experience with neutron activation methods. Accurate determination of the beam properties is critical for production of astrophysical neutron source. Significant efforts were done in that direction. Different methods (time of flight, Rutherford backscattering, energy scan of radiation yields from some threshold and resonance nuclear reactions) were used for energy calibration. Some nuclear physics instrumentation was used for the first time in on-line measurements. Several systematic problems were discovered during these activities. Improvements in the diagnostics equipment will be implemented in the next future.

Operation of deuteron beam (first step towards radioactive beams)

The production of intense high energy neutrons sources with deuteron beams is the important direction for the SARAF project. In particular, such intense sources will be used for productions of radioactive beams for application and fundamental research. The first step in that direction was a test for production of high energy neutrons via $^7$Li(d,n) reaction. The experiment was performed by bombarding a LiF crystal target with a pulsed deuteron beam. The spectrum of the produced neutrons was measured on-line using a liquid scintillator and an electronic setup, which included a pulse shape discriminator, to separate the neutron signal from the gamma ray contribution. This was the first experiment at SARAF where this neutron detection instrumentation was used. In addition a stack of metallic foils was placed next to the target and activation analysis was performed, in order to obtain the neutron energy spectrum by an alternative offline method. The average current was kept in the order of 1 μA. The neutron spectrum was collected for a few hours. The raw neutron energy spectrum, separated from the gamma signal, is shown in the Fig. 2. De-convolution of the raw spectrum, as well as analysis of the activated foils, is still in progress [4]. An improved version of the setup is under preparation.
CONCLUSION
The first beam operation experience in 2011 showed that even at Phase I SARAF has potential to become a real user facility with intense beam schedule and high beam availability. The SARAF teams will strive to achieve this goal despite of various logistical and technical hindering factors.

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Improvements Design for the SARAF Phase I RFQ

J. Rodnizki, Z. Horvitz

Soreq NRC, Yavne, Israel

Corresponding Author: jacob.rodnizki@gmail.com

INTRODUCTION

The SARAF 4-rod RFQ (1,2) is operating at 176 MHz, designed to bunch and accelerate a 4 mA CW deuteron/proton beam to 1.5 MeV/u. The electrodes voltage for accelerating deuterons is 65 kV, a maximal field of 22 MV/m.

The pre beam dynamics design defined the electrodes modulation to achieve gentle bunching and synchronous acceleration based on flat voltage between the electrodes along the RFQ. 14 tuning plates between adjacent stems are placed along the RFQ, with adjustable height on top of the base plate, enable to reach the RFQ desired resonance frequency, with a flat voltage between the electrodes. Adjusting the tuning plates height varies the RF cells inductance, while the distance between the electrodes governs the RF cells capacity. The 250 kW dissipated power needed to accelerate a deuteron beam are distributed along the 3.8 m RFQ length. This power density is approximately 3 times larger than ever achieved in other 4-rod RFQs prior to this project. At high power, local high surface currents in the RFQ might cause overheating which will lead to out-gassing and in turn to sparking (3). Therefore, there is a vital need for a detailed RF simulation of the RFQ, in combination with thermal simulation, in order to determine a priori the areas that heat up uncontrollably at high power, and for the design of improved cooling. We used CST MWS to simulate the RF currents and fields in a 3D detailed model of the SARAF RFQ. The heat load generated by the simulated surface currents at critical areas along the RFQ was the input for thermal analysis using the thermal solver of CST MWS and ANSYS. These simulations were in good agreement with the findings in the RFQ upstream and downstream flanges zone. High level surface current, up to 2500 A/m, caused sever damage to the RF contacts and the vacuum O-ring seal in this zone. We designed improvements, based on cutting-edge new RF and vacuum sealing components. In this design new upstream and downstream flanges will replace the old ones. In the new upstream flange we also designed Electron Trap to prevent electrons from the LEBT unit to enter the RFQ. This Trap consists of high voltage electrode and a water cooled collimator. Our design is based on similar designs of electron trap that can be seen in the SPES (4) and LEDA (5) works.

THE NEW DESIGN

Our new design is giving answers for two issues, the first one is upgrading the RF and vacuum sealing, and the second is adding an Electron Trap device in the entrance of the RFQ.

Upgrading the RF and vacuum sealing - There are two different RF sealing in each end flange as shown in figure 1. RF sealing 1 seals between the end flange and the tank, and RF sealing 2 seals between the base plate and the end flange. Figure 2A shows the high level surface current around the end flange as was simulated in CST. Figure 2B shows the discoloration of the electroplated copper end flange, which was the result of the failure of RF sealing 1. Figure 2C shows the contact fingers that failed in RF sealing 2, due to the high local surface current between the base plate and the end flange.
The two RF sealing failed due to higher local current than their rated values. The failure mechanism is such that when it just begins, the spring contact is immediately losing its elasticity, which stops it from giving good electric contact, causing additional extra heating. The new designed RF seals are Canted Coil Bal-Spring, which have high conductivity due to highly concentrated forces at numerous contact points, and also having almost constant force along large deflection, as can be seen in figure 3. This advantage can compensate for larger mating tolerances and alignment.
A comparison between the Be-Cu contact fingers and the canted coil RF sealing is presented in figure 4. Thermal simulations results with 2500 A/m as heat source between two 20°C plates is shown in figure 4.

**Contact fingers**
- highest temp. – 10% contact **1240 °C**
- highest temp. – 100% contact **35.3 °C**

**Canted coil spring**
- highest temp. – **20.4 °C**

In order to withstand higher electromagnetic and surface current that might pass the RF sealing, we design for new aluminum U shape metallic vacuum sealing from "Garlock". Although these cutting edge technology components are well familiar in accelerators facilities, we intend to measure their performances in a testing bench.

The Electron Trap device – Preventing electrons entering the RFQ from the LEBT utilized in other accelerators is shown in figure 5.

We designed a similar electron trap device as part of the RFQ end flange. The total length of the collimator is limited to 65 mm, which is the maximum available space. In figure 6 we present the simulated RMS radial beam radius, envelop beam size and the collimator surface. We can learn from it that in normal operation the beam doesn't reach the collimator.
The electron trap device has cooling water and HV lines coming from the RFQ end flange, the collimator consists of two brazed elements, while cooling water are flowing in between as can be seen in figure 7. Thermal simulation results, in case of "off axes" beam directed on the collimator inner surface can also be seen in figure 7. The heat source is projection of 4 mm diameter of 400W, which originates from the beam maximum power with safe factor of 1.5.

CONCLUSIONS

In our proposed new design with the canted coil RF sealing technology we showed that they are more efficient than the two round mesh Monel wire and the contact fingers RF sealing. This new RF sealing along with the new metallic vacuum sealing and the Electron Trap device give us the confidence that the RFQ will function much better.
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Redesign of Existing SARAF RFQ Electrodes to Enable CW Deuteron Operation

Asher Shor

Nuclear Physics and Engineering Division, Soreq NRC, Yavne 81800

Corresponding Author: shor@soreq.gov.il

INTRODUCTION

The SARAF accelerator has been designed for CW acceleration of proton and deuterons at beam intensities up to 4 mA (see I. Mardor ref. [1]). Phase I commissioning has revealed several limitations for routine operation of the existing 176 MHz RFQ. Most notably, the high power demands on the RFQ and the inadequate thermal design does not allow for CW operation at 65 kV required for deuteron acceleration. The heat load required for deuteron acceleration reaches 250 kW, while routine CW operation can be attained for, at best, only 200 kW. As a result, routine deuteron acceleration can be achieved only for a limited RFQ duty cycle (see L. Weissman ref. [2]). Finite element calculation performed by Jacob Rodnizki (reference [3]) have identified locations in RFQ prone to excessive thermal load, and appropriate modifications have been made to alleviate some of the more problematic components. These measures and others [Mardor [1], Weissman [2]] have succeeded in significant improvement of the RFQ performance, but still, at present, the maximum routine CW operation that can be obtained is below 200 kW.

We have embarked in a redesign of the existing SARAF RFQ electrodes to allow for deuteron acceleration at lower RFQ voltages and therefore lower the required RF power with the goal of enabling CW RFQ operation for deuterons. To this end, we have used the General Particle Tracer (GPT) simulation code, along with the RFQ element implementation and 8-term potential procedure as described in reference [4]. We have also used the technique of tail emphasis described in ref [5] to obtain high statistics simulations at reasonable computing time. Results of the simulations with the optimized modulation parameters demonstrate that the RFQ can be redesigned to enable deuteron operation at 170 kW, while providing clean deuteron beam with transmission of 92% for 4 mA beam at 1.32 MeV/u.

Optimization of RFQ modulation parameters for lower fields

In our simulations, we use the RFQ implementation and 8-term potential for the existing SARAF RFQ as described in reference [4]. We use the analytic expression for the 2-term potential only to guide us in guessing a procedure for modification of the existing modulation parameters for obtaining lower emerging deuteron energy.

The 2-term potential is given by:

\[ U(x, y, z, t) = \frac{V_0}{2} \left[ \frac{X}{a^2} (x^2 - y^2) + A I_0(kr) \cos(kz) \right] \sin(\omega t + \phi) \] (1)

where \( X \) represents the (transverse) focusing term and \( A \) represents the acceleration term. The focusing term is given by:

\[ X = \frac{I_0(ka) + I_0(kma)}{m^2 I_0(ka) + I_0(kma)} \] (2)

and the acceleration term is given by:
\[ A = \frac{m^2 - 1}{m^2 I_0(k\alpha) + I_0(k\alpha)} \]  

(3)

\( I_0 \) is the modified Bessel function, \( \alpha \) is the minimum cell aperture and \( m\alpha \) is the maximum cell aperture. \( (m) \) is the modulation strength, \( k \) is the wave number given by \( k = \frac{\pi}{L} \), where \( L \) is the cell length given by \( L = \beta_s \alpha / 2 \), where \( \beta_s \) is the velocity of the synchronous particle and \( \lambda \) is the wavelength corresponding to 176 MHz.

Our procedure for redesign of the existing SARAF RFQ modulation parameters consists of a straightforward scaling down procedure, followed by two correction schemes. The criteria for the redesign is to maintain good bunching and acceleration, but for a lower applied power. We chose a value of 160 kW, rather than the required 250 kW for deuteron acceleration. Assuming the RFQ voltage at 250 kW is the nominal 65 kV for deuterons, then the RFQ power of 160 kW translates to an applied RFQ voltage of (assuming all other things are constant)

\[ V_0 = 65 \text{ kV} \times \sqrt{\frac{160\text{kW}}{250\text{kW}}} = 65\text{kV} \times 0.80 = 52 \text{ kV} \]  

(4)

To enable lowering of the applied voltage, we must modify the modulation cell lengths such that the bunching and phase advance remain the same. From equation (1) we can see that the acceleration term is proportional to the applied voltage. This means that at each accelerating section, the energy gain is proportional to \( \Delta E \sim A \times V_0 \). To keep the bunch at a particular cell at the same phase given the decreased acceleration, the cell lengths are decreased by a corresponding amount. The wave number is then increased according to

\[ (\Delta k)_{52 \text{ kV}} \sim \frac{\pi}{\Delta L} \sim \frac{1}{\Delta \text{vel.}} \sim \frac{1}{\Delta E} \sim (\Delta k)_{65 \text{ kV}} / \sqrt{0.8} \]  

(5)

The cell length is decreased by \( \Delta L \times \sqrt{0.8} \) and the wave number increased by \( \Delta k / \sqrt{0.8} \).

From equation (1), we see that the transverse focusing is proportional to \( V_0 / \alpha^2 \). If we lower the applied voltage to the RFQ, the transverse focusing will decrease correspondingly and we will begin to get significant transverse beam loss. To keep the same transverse focusing strength we need to decrease the quantity \( \alpha^2 \) by the same amount that we decrease \( V_0 \), namely decrease the RFQ cell aperture by \( \alpha \rightarrow \alpha \times \sqrt{0.8} \).

The first correction that is needed is as follows: Following the above procedure, we obtain an outgoing longitudinal beam distribution with a tail. This will give us trouble with beam loss when injecting the beam into the superconducting linac. Actually, the same problem exists with the non-modified SARAF RFQ; there is a tail for nominal deuteron operation at 65 kV. The solution there was to raise the voltage to 67 kV, at which point the tail disappears (see A. Shor ref. [6]). To apply the same correction to the modified electrodes, we simply raise the applied voltage correspondingly to 53.6 kV, and the required RFQ power is then raised to 170 kW.

The second correction is the following. According to the above procedure, the outgoing deuteron energy of the modified RFQ should be \( E = E_0 \times \frac{V}{V_0} = 1.5 \text{ MeV} \times \frac{52}{65} = 1.2 \text{ MeV} \). Since the lengths of the cells are shorter, the total RFQ length now is about 3.4 meters. We can add additional 7 cells to the RFQ to maintain the same length as the original RFQ at 3.73 meters, for a total of 198 cells. This brings up the energy to 1.32 MeV/u.

**Beam dynamics simulations of RFQ with modified modulation parameters**

Table 1 summarizes the results of beam dynamics simulations of existing and of modified RFQ. The transmission calculated by GPT for the non-modified RFQ for deuterons at 4 mA is 86 %, where for the modified RFQ we get transmission of 92%.

Figure 1 shows the longitudinal phase space plot for the emerging beam. The mean energy for the non-modified RFQ at 65 kV is \( E=1.5 \text{ MeV/u} \), with the familiar longitudinal beam spot...
profile which shows development of a longitudinal tail. The beam spot for the modified RFQ at 53.6 kV has a mean energy of 1.32 MeV/u, without any indication of a longitudinal tail.

Table 1. RFQ transmission as calculated with GPT:

<table>
<thead>
<tr>
<th>Old RFQ modulation parameters compared to modified RFQ modulation</th>
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<tbody>
<tr>
<td>Required Power (kW)</td>
</tr>
<tr>
<td>---------------------</td>
</tr>
<tr>
<td>old RFQ @ 65 kV</td>
</tr>
<tr>
<td>new RFQ @ 53.6 kV</td>
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Figure 1. Longitudinal phase space (100,000 macro-particles) at the end of the RFQ

a) old RFQ parameters @ 65 kV  b) modified RFQ parameters @ 53.6 kV

**SARAF superconducting linac tune for 1.32 MeV/u beam with modified RFQ**

We simulate the SARAF linac with the 1.32 MeV/u deuteron beam at 4 mA obtained with the new RFQ with the modified modulation parameters. We simulate the asymmetric version of the SARAF linac consisting of two $\beta=0.09$ cryomodules followed by four $\beta=0.15$ cryomodules. An asymmetric cell consists of one solenoid followed by two HWRs, where the two $\beta=0.09$ cryomodules consist of 3 cells, and the four $\beta=0.15$ consist of 4 cells each. The linac lattice is the default lattice used in most of the SARAF beam dynamics simulations.

Linac simulation were performed with tail emphasis technique (reference [5]) and equivalent to 1,500,000 macro-particles resulted in no loss of particle neither for the 1.5 MeV/u initial beam with old linac tune, nor for the 1.32 MeV/u initial beam with the new linac tune. Figure 2 shows results of linac simulations, including development along the linac of the beam energy, longitudinal emittance, and transverse emittance and envelope.

**CONCLUSIONS**

A redesign of the existing SARAF RFQ electrodes has been initiated to make possible CW operation for deuterons. The existing RFQ design is for emerging energy 1.5 MeV/u and operating voltage of 65 kV for deuterons, thereby requiring RFQ power consumption of 250 kW for CW beam. An re-design for the RFQ modulation parameters is described which will enable applied voltage of 53.6 kV, thereby lowering the power consumption to 170 kW, thus enabling reliable CW RFQ operation. The emerging deuteron beam will have energy of 1.32 MeV/u and a transmission of 92 % for a 4 mA beam. Beam dynamics simulations for the redesigned RFQ show longitudinal phase space plots for deuterons that are well-behaved and exhibit no tailing. A successful linac tune is obtained (for the old SARAF lattice) that takes the
emerging deuterons from the redesigned RFQ at 1.32 MeV/u and accelerates them to a final deuteron energy of 43 MeV, with no beam loss and minimal emittance growth for a simulation with tail emphasis equivalent to 1,500,000 macro-particles for 4 mA deuterons.

REFERENCES

Figure 2: Linac simulations with 1,500,000 macro-particle equivalent
Left: old RFQ parameters @65 kV. Right – new RFQ parameters @ 53.6 kV
a. Energy development b. Longitudinal emittance r.m.s c. Longitudinal emittance 100%
d. Transverse emittance r.m.s. e. Transverse beam envelope
SARAF Beam Energy and Phase Distribution Measurements

A. Kreisel, L. Weissman
Soreq NRC, Yavne, Israel

Corresponding Author: arikk@soreq.gov.il

INTRODUCTION
Knowledge of the beam energy and phase distributions at SARAF are essential for both user experiments and beam dynamics design. While the target experiments need to know the energy distribution after the accelerating module, to design the beam dynamics needed to reach this energy, it is necessary to know the 2D energy and phase distributions at the entrance to the accelerating module. The beam energy distribution is measured after the accelerating module by Rutherford Scattering (RS) of the part of the beam into a silicon detector. This report describes the method used to measure the resolution of this measurement as well as the method used to measure the 2D energy and phase distributions at the entrance to the SARAF accelerating module.

EMITTANCE FORMALISM
With linear focusing forces the trajectory of each particle in the 2D projection phase-space lies on an ellipse. Therefore, the tendency of LINAC beams is to exhibit approximately elliptical isodensity contours. The general equation for an ellipse is,

\[ \gamma X' = 2 \alpha xx' + \beta x'^2 = \varepsilon \]  

(1)

Where \( x \) and \( x' \) are position and direction of the particle and \( \alpha, \beta, \gamma \) are the ellipse twist parameters. The definition of the emittance, \( \varepsilon \), associated with the elliptical phase-space area is,

\[ \varepsilon = \frac{\text{Area}}{\pi} \]  

(2)

The presence of nonlinear forces can produce a considerable departure from elliptical trajectories. To describe an arbitrary particle distribution the rms emittance, \( \varepsilon_r \) is used. The rms emittance, \( \varepsilon_r \), is defined by requiring the projections on the \( x \) and \( x' \) axes to equal the rms values of the distributions which means the variance \( V(x) \) and \( V(x') \) equal,

\[ V(x) = \bar{x}^2 = \beta_r \varepsilon_r, \quad V(x') = \bar{x'}^2 = \gamma_r \varepsilon_r \quad \text{and} \quad \alpha_r = -\frac{x x'}{\varepsilon_r} \]  

(3)

and requiring \( \gamma_r \beta_r - \alpha_r^2 = 1 \) gives

\[ \varepsilon_r = \sqrt{\bar{x}^2 \bar{x'}^2 - xx'^2} \]  

(4)

Equation 1 can be written in a matrix form

\[ \begin{bmatrix} x \\ x' \end{bmatrix} = X \sigma^{-1} X = \varepsilon \quad \text{when} \quad X = \begin{bmatrix} x \\ x' \end{bmatrix} \quad \text{and} \quad \sigma = \begin{bmatrix} \beta & -\tilde{\alpha} \\ -\tilde{\alpha} & \tilde{\gamma} \end{bmatrix} \]  

(5)
If we denote $R$ as the transform matrix from point 1 to point 2 so that

$$X_2 = RX_1, \quad (6)$$

then from equation 5

$$\varepsilon_2 \sigma_2 = R \varepsilon_1 \sigma_1 R^T \quad (7)$$

**LONGITUDINAL EMITTANCE**

In the Longitudinal case one replaces $x$ and $x'$ with the phase $\varphi$ and the kinetic energy $W$.

$$\begin{bmatrix} \varphi_2 \\ W_2 \end{bmatrix} = R \begin{bmatrix} \varphi_1 \\ W_1 \end{bmatrix} \quad (8)$$

In a drift there is no change in $\Delta W$ and the change in $\Delta \varphi$ originates from the existing $\Delta W$ therefore

$$R_{\text{drift}} = \begin{bmatrix} 1 & Y \\ 0 & 1 \end{bmatrix} \quad (9)$$

In the case of a cavity, the change in $\Delta W$ is due only to the $\Delta \varphi$ the particles have when entering the cavity. For the change in $\Delta \varphi$ we assume a thin cavity where there is no change in $\Delta \varphi$. Thus we get that,

$$R_{\text{cav}} = \begin{bmatrix} 1 & 0 \\ X & 1 \end{bmatrix} \quad (10)$$

Transforming from point 1 to point 2 in a series of drift cavity drift,

$$R_{\text{total}} = \begin{bmatrix} 1 & Y_2 & 1 & 0 & 1 & Y_1 \\ 0 & 1 & X & 1 & 0 & 1 \end{bmatrix} = \begin{bmatrix} XY_2 + 1 & (XY_1 + 1)Y_2 + Y_1 \\ X & XY_1 + 1 \end{bmatrix} \quad (11)$$

From equations 3, 7, 11,

$$\varepsilon_2 \bar{\gamma}_2 = V(W_2) = X^2 \varepsilon_1 (\bar{\gamma}_1 Y_1^2 - 2\bar{\alpha}_1 Y_1 + \bar{\beta}_1) + X \varepsilon_1 (2\bar{\gamma}_1 Y_1 - 2\alpha_1) + \varepsilon_1 \bar{\gamma}_1 \quad (12)$$

Equation 12 shows that the energy variance $V(W_2)$ is a parabolic function of $X$ and by measuring $V(W_2)$ as a function of $X$ we can extract the parabolic parameters $p_0, p_1, p_2$. Using equation 12 and the fact that $|\sigma|=1$ it is easy to get the following relations to calculate the ellipse parameters.

$$\varepsilon_1 = \sqrt{p_2 p_0 - \frac{p_1^2}{4}} \quad \bar{\alpha}_1 = \frac{1}{\varepsilon_1} \left( p_2 Y_1 - \frac{p_1}{2} \right)$$

$$\bar{\beta}_1 = \frac{1}{\varepsilon_1} (p_0 - p_1 Y_1 + p_2 Y_1^2) \quad \bar{\gamma}_1 = \frac{p_2}{\varepsilon_1} \quad (13)$$

To calculate what is $X$ we need to calculate the energy spread, $\Delta W$, due to the phase spread, $\Delta \varphi$, which the particles have when entering the cavity. The average energy gain of the bunch passing through a cavity is $E_{\text{gain}} = E_{\text{gain}}^{\text{max}} \cos \varphi_r$ when $\varphi_r$ is the phase of the reference particle. The difference in energy gain between a particle arriving at phase $\varphi_r$ and phase $\varphi_r + \Delta \varphi$ is

$$\Delta W = E_{\text{gain}}^{\text{max}} (\cos(\varphi_r + \Delta \varphi) - \cos \varphi_r) \quad (14)$$

For the case when $\Delta \varphi \ll 1$ we have,
\[ \Delta W = -E_{\text{gain}}^{\text{max}} \sin(\varphi_r) \Delta \varphi \]  

(15)

Inserting \( E_{\text{gain}} = E_{\text{gain}}^{\text{max}} \cos \varphi \), we get,

\[ \Delta W = -E_{\text{gain}} \tan(\varphi_r) \Delta \varphi \quad \Rightarrow \quad X = -E_{\text{gain}} \tan(\varphi_r) \]  

(16)

To calculate what is \( Y \) we need to calculate the phase spread \( \Delta \varphi \) that evolves due to the energy spread \( \Delta W \) in the drift. The time difference after a drift \( D \) relative to the reference particle is

\[ \Delta t = \frac{D}{c\beta} - \frac{D}{c\beta_r} = \frac{D(\beta - \beta)}{c\beta \beta_r} = 2 \frac{D\Delta \beta}{c\beta_r^2} \]  

(17)

(When \( \beta \) and \( \gamma \) are the relativistic velocity and Lorentz factor). Changing \( \Delta t \) to \( \frac{\Delta \varphi}{2\pi f} \) gives

\[ \Delta \varphi = -2\pi f D \frac{\Delta \beta}{c\beta_r^2} \]  

(18)

The difference in \( \beta \) can be related to the difference in energy

\[ E = \gamma mc^2, \quad \Delta E = \frac{\partial E}{\partial \gamma} \Delta \beta, \quad \frac{\partial E}{\partial \beta} = \frac{\gamma^3}{\beta}, \quad \frac{\partial E}{\partial \gamma} = mc^2 \]

\[ \Delta W = \Delta E = mc^2 \gamma^3 \beta \Delta \beta \]  

(19)

Inserting into equation 18 we get

\[ \Delta \varphi = -2\pi f D \frac{\Delta W}{mc^3 \gamma^3 \beta_r^3} \quad \Rightarrow \quad Y = -\frac{2\pi f D}{mc^3 \gamma^3 \beta_r^3} \]  

(20)

MEASUREMENTS OF LONGITUDINAL PHASE-SPACE ELLIPS PARAMETERS

The longitudinal emittance at the entrance to the SARAF accelerating cryogenic module was measured by using equation 12 for a drift cavity drift sequence. The energy variance of the protons was measured at a location downstream from the module by using RS of the protons into a Silicon detector. The measured energy variance was plotted as a function of \( X = -E_{\text{gain}} \tan(\varphi_r) \) (equation 16). The variance as a function of \( X \) is fitted to a parabola when \( X \) is changed by changing the accelerating voltage of a single cavity. The rms emittance parameters at the entrance to the SARAF accelerating cryogenic module were measured and extracted from the fit shown in figure 1 to be,

\[ \varepsilon = 0.89 \pm 0.48 \text{keV} \cdot \text{rad} \]

\[ \bar{\varepsilon} = 4.8 \pm 0.02 \]

\[ \bar{\beta} = 0.04 \pm 0.02 \frac{\text{rad}}{\text{keV}} \]
Figure 1. Energy variance as a function of $X = -E_{\text{gain}} \tan(\varphi_r)$.

Figure 2. Left: phase and energy distributions used in old (upper), and new (lower) MC. Middle: Comparison of experimental RS spectrum (points) with old (upper) and new (lower) MC. Right: Comparison of experimental results with new and old MC.

The same measurements done in real data were repeated using the Track\textsuperscript{2} simulation MC. The measurements were repeated once using the above measured emittance parameters (shown in figures 2 lower plots) and the previous estimation (figures 2 upper plots). The new simulation shows a much improved description of the data.

REFERENCES
Electron Beam Tests of a High-Power Liquid-Lithium Target as an Intense Epithermal Neutron Source

S. Halfon1,2, M. Paul2, A. Arenshtam1, D. Berkovits1, I. Eliyahu1, N. Hazenshprung1, D. Kijel1, and I. Silverman1

1Soreq NRC, Yavne, Israel 81800
2Racah Institute of Physics, Hebrew University, Jerusalem, Israel 91904

Corresponding Author: halfon@phys.huji.ac.il

INTRODUCTION

A prototype of a compact Liquid Lithium Target (LiLiT), which will be able to constitute an accelerator-based intense neutron source with possible application for boron neutron capture therapy (BNCT) in hospitals, was built and tested with high power electron gun at Soreq Nuclear Research Center (SNRC). The lithium target will produce neutrons through the $^7$Li($p,n)^7$Be reaction and it will overcome the major problem of removing the thermal power generated by a high-intensity proton beam (1.91-2.5 MeV, >3 mA), necessary for sufficient therapeutic neutron flux.

The optimization of the neutron flux and energy spectrum for the maximum benefit to the therapy of deep-seated tumors has been studied in the last fifteen years(1). High neutron flux of $10^9$cm$^{-2}$s$^{-1}$ at an irradiation facility beam port and epithermal neutrons energy, lying in the energy range 0.5 eV<$E<$10 keV, have been assessed as best suited for therapy of such tumors for a reasonable therapy duration (30–90 min$^2$).

Worldwide efforts to design a neutron converter for an accelerator-based BNCT facility, which might be more compatible with clinical environment (in hospital), have been focused on the use of lithium through the reaction $^7$Li($p,n)^7$Be(3,4) at proton energies of 1.9–2.5 MeV. The major advantage of this reaction consists in its low-energy neutron spectrum (mean neutron energy in the range of 30–300 keV). Despite the excellent neutronic qualities of the $^7$Li($p,n)^7$Be reaction, a reliable lithium target, working under beam power levels considered for therapy purpose (at least 3 mA, ~2 MeV protons), has been considered as very difficult to build because of the mechanical, chemical and thermal properties of lithium (low melting point of 180°C and low thermal conductivity of 85 W/(m·K) at 300 K), the major problem being to remove the thermal power generated by the high-intensity proton beam. For such high intensity beam a solid lithium target would be destroyed by heat deposited in the target unless specific measures are undertaken to dissipate the heat. Several groups around the world are indeed attempting to design solid lithium target which will be able to dissipate high intensity particle beam(3,5) but so far no such target exists. In this work we present high power electron beam irradiation of a liquid lithium target (LiLiT) simulating the high thermal power density of the proton beam required for such BNCT purposes.

RESULTS

The liquid-lithium loop of LiLiT is designed to generate a stable lithium jet at high velocity on a concave supporting wall with free surface for the incident proton beam. The liquid-lithium flow at a temperature of ~200°C is driven by an electromagnetic (EM) induction pump. The lithium flow is collected into a containment tank where a heat exchanger dissipates the beam power.
Radiological risks due to the $^7$Be produced in the reaction will be handled by a cold trap expected to collect a significant part of the 7Be activity and by appropriate shielding; more details on the target design and operation can be found in ref. 6.

A stand-alone fire resistant laboratory was built near the SARAF accelerator, designated to study the proper working and operation procedures with the liquid-lithium apparatus and to conduct circulation and electron-gun irradiation offline tests. In the preliminary off-line circulation tests, liquid lithium was circulated through the loop and generated a stable jet at velocity higher than 5 m/s on the concave supporting wall\(^6\).

The aim of the electron gun irradiation off-line tests (fig. 1) is to demonstrate that a power equivalent to the ~2 MeV, 2 mA SARAF proton beam (corresponding to power densities of the order of MW/cm\(^3\)) can be dissipated by the circulating lithium. A high-intensity (26 kV, 1 A) electron gun was attached to the system, and the electron beam is directed to the lithium surface. The electron beam (with a mean range of 26 µm in liquid lithium) is simulating the energy deposition of the proton beam. Simulations described in\(^6\) showed that an electron beam, despite it's different energy deposition profile, can provide similar maximum energy deposition as particle beam. The maximum energy deposition density is one of the most significant parameters governing lithium jet disturbances. Hence, we anticipate that studying the behavior of our lithium film, its stability, continuity and shape, under an electron beam, could predict its behavior under the proton beam of SARAF. Video recording were made of the beam-on-target image, while increasing the lithium velocity and the electron beam power.

Figure 1. Electron gun attached to LiLiT system. 1A: electron-gun port view (1- electron-gun), 1B: side view (2- target vacuum chamber, 3- magnetic lens for electron gun directing and focusing.
Four electron beam experiments took place. In the first experiment the circulation of the lithium along with the electron gun was established. In the next two experiments high power electron beam, up to 2.2 kW (85 mA), was applied on the flowing lithium film. The electron beam shape was fitted to a gaussian with sigma of 3.5-4 mm. According to that, the maximum power density applied on the liquid lithium, flowing at an estimated velocity of ~3.15 m/s, was 2.85 kW/cm² and the volumetric power density was ~0.8 MW/cm³ without disturbances or flow instabilities. Throughout these two experiments the irradiation periods were short, limited by the continuous heating of the nozzle edges (“ears”) by the electron beam halo, up to temperatures above 700°C. An excessive heating of the nozzle might damage it shape, disturb the lithium flow and even cause lithium leaks. Another problem was excessive lithium evaporation appearing under high electron beam power, covering the internal parts of the target and blocking the view ports.

In order to overcome these problems two improvements took place before the fourth experiment, the first one was the change of the magnetic lens, creating better focus of the electron beam on the target, now fitted to a Gaussian with sigma of 2.7 mm (fig. 2). The second improvement was the insertion of a cold trap between the target chamber and the view ports, to catch lithium vapors and to avoid the view ports blackening.

During the fourth experiment four long (more than 10 minutes) irradiations took place (fig. 3, irradiations marked 1-4). During these irradiations the electron beam power was increased from an average of 0.7 kW in the first run (1, fig.3) to an average of 1.5 kW in the fourth run (4, fig. 3). The temperatures of the nozzle ears were stable and didn't increase (fig 3). The vacuum at the target chamber was low and even decreased during the last irradiation down to 6·10⁻⁶ mbar. In the last irradiation (4, fig.3) the beam power was alternating between 1.36-1.92 kW (52-74 mA) with average power of 1.5 kW. The maximum power density applied on the liquid lithium, flowing at an estimated velocity of ~4 m/s, was 4.2 kW/cm² and the volumetric power density was ~0.8 MW/cm³. During this irradiation period the maximum nozzle ears temperature were stabilized around 450°C (fig. 3), the system was stopped voluntarily after 45 minutes.

CONCLUSIONS

The LiLiT electron beam irradiation demonstrates that the liquid-lithium target can dissipate power densities of more than 4 kW/cm² and volumetric power density higher than ~0.8 MW/cm³ at a lithium flow of ~4 m/s, while maintaining stable temperature and vacuum conditions. These power densities of the order of 1 MW/cm³ correspond to an epithermal neutron intensity of 1.1·10¹³ n/sec per mA of 2 MeV proton beam. The LiLiT target is planned to be used as a neutron source for accelerator-based boron neutron capture therapy (aBNCT) and astrophysical research.
REFERENCES

Fig 3. Maximum and minimum nozzle edge temperatures along with the electron beam power during irradiation periods marked 1-4.
Control System for SARAF Phase I Beam Line and Targets


Soreq NRC, Yavne, Israel 81800

Corresponding Author: ilan.eliyahu@gmail.com

INTRODUCTION

A new beam line [1,2] was built in addition to the existing beam dump line, in order test new targets while protecting the superconducting LINAC from dirt coming from the target. The beams are delivered to several experiments, Liquid Lithium Target (LiLiT) [3,4,5] and Foils target [6] which have broad range of requirements. The new beam line infrastructures consist of several interconnected parts: magnetic beam optics elements, beam diagnostics elements, vacuum control and machine safety system (MSS).

CONTROL OF BEAM OPTICS ELEMENTS

The purpose of the magnets control is the providing the magnet currents necessary for beam transport. The control system also provides the "magnets ready" interlock for the accelerator MSS. The following magnetic elements were used for control of beam optics: 1) Two 45° dipole magnets for transporting the beam to the new beam line. 2) Five quadruples lenses (two doubles and one single) for containing the beam and preparing the required beam spot for various targets. 3) two x-y steer magnets. All the magnets except the first doublet are water cooled. The thermo switches with 60° limits are placed on the magnet coils. Some of the magnets (dipoles, a quadruple and steers) were DANFYSIK. The two doublets were recuperated from other institutions.

The magnet control hardware located in a mountable rack located in the RF Hall (approximately 30 meters from the magnets). It is containing NI compact reconfigurable Input/output (cRIO) platform control system and the appropriate power supplies for each magnet.

The control system provides "magnets ready" signal for the accelerator MSS. This signal requires the conditions when all the power supplies are enables, the chain of thermo switches is intact and the current values on the dipole magnets in the predetermined range. The power supplies could be enabled only when there is sufficient magnet cooling water flow. The beam transport condition (beam line rather than beam dump) is chosen by signal received from the accelerator control system. In the case when the beam dump line is chosen the current on the first dipole is forced to be zero in order to get "magnets ready" signal.
The main screen of the magnets control application is shown in figure 1. The application allows the user to see whether the beam line or the beam dump is chosen for beam transport. The user enables the magnet supplies and set the desirable currents. The system presents the output current and the deviation from the required current for every magnet. The magnets are ramped to the set values with a ramping speed which also controlled by the user. For the two dipole magnets the upper and lower current threshold can be set thus enable preventing damage to the accelerator. The Information screen presents the status of thermo switchers, cooling water flow, beam line selection switch.

**BEAM DIAGNOSTICS CONTROL**

The purpose of the beam diagnostics system is beam tune and preparation beam on a target within requirements of specific experiment. The control of the beam diagnostics facilitates operation of the diagnostics elements. it also protects beam diagnostics from damage from intense beam. The beam line diagnostics consists of three of X_Y wire-profilers stations and adjustable beam collimators (“4-jaw”).

Each X-Y scanner station consists of two “forks” installed in a 6-way cross. An electrically insulated 150 micron thick tungsten wire is stretched between the ends of the fork. Both ends of the wire are connected to electrical feedthroughs, so the resistance of the wire can be tested externally. The forks are moved vertically and horizontally perpendicular to the beam axis using linear motion mechanisms (Huntington) and stepping motors. On each motion axis there are two limit switches, aimed to restrict the fork movement and thus to avoid damage to the system. The motion of the vertical and horizontal scanners is not performed on the same plane, thus, eliminating the risk of mechanical collision. After introduction into the beam, electrical charge collected on the wire is converted into voltage signal using a 1kΩ resistor. The dependence of the wire signal on its position yields information on the beam profile. These wire scanners can work only in pulsed beam (diagnostic mode) at a relatively low duty cycle. To synchronize the signal from the wires the master accelerator trigger was used as a trigger for digitizer. The wire scanners can't be used for CW beam operation (normal mode).

The control software project includes two parts: the RT controller software (PXI) and the Host computer (PC). The host computer Vi’s handle the graphical user interface (GUI) and data record, while the RT controller loaded to the PXI includes the main operation Vi’s and the Shred...
Variable Library (SV). This configuration of the VI's makes it possible to run the main program on the PXI without the need to operate the host computer, making it much more reliable and robust.

Wire scanners control application is shown in figure 1- The right part of the screen presents the scan results as an X-Y graph and the scan parameters. The profile characteristics: centroid, standard deviation and skewness, are presented in the left middle screen. The upper left corner presents the current wires positions, their velocity and operation status (scan mode). Additionally there is a message field for error notifications and scan status. At the bottom left corner the user sets the operation mode. 1) Scan process: the system starts the scan process based on the defined parameters (range, steps). 2) Returns the wire to the home "out of beam" position. 3) Save data file and plot picture. 4) Emergency stop of the profiler movement.

The "diagnostics out" signal based on the status of the profiler limits is sent to the accelerator MSS. The accelerator cannot operate in full power mode if the profilers are inserted in the beam pipe.

**LiLiT TARGET**

A stand-alone fire resistant laboratory was built to test the liquid-lithium loop, its heat removal capabilities using an attached electron gun as a heater and to establish the safety regulations. The LiLiT control system provides full and safe control of the lithium and its cooling oil loops. The control system is based on a National Instruments compact field point (cFP) 2020 controller. Additionally, the control system includes drivers for the Li and oil motors, vacuum controllers, linear motors for the diagnostic wire scan and for the beam blocker, temperature controllers, etc. The code for the cFP controller is divided in two subsystems. The real time controller program with the critical code controls the most important parameters: the different set points (lithium speed, vacuum safety level, etc.) and the parameters of lithium heating process. The host computer program handles all the other control functions. The Humane Machine Interface (HMI) was built to reflect as closely as possible the system structure, presenting its functionality and parameters in the most intuitively clear way (figure. 2).

The left side of the screen enables operation of the various systems, such as the local temperature controllers, the Li tank and other heaters, the EM pump motor and wire control. Given the safety concerns associated with liquid lithium, the initial heating process of the solid Li must be performed carefully using a proportional temperature feedback algorithm. The Li loop is presented in the centre of the HMI. The diagnostics of the loop include the temperature mapping of the various locations in the loop (in the Li tank, target and pipes) and control of the EM pump. The oil loop is presented on the right side of the screen. This loop handles the operation of the oil motor, cooling fan and the oil heaters in the pipes. On the same side, the different parameters are also shown in the running plot. Before installing the LiLiT experiment at the SARAF proton beam line, we intend to substitute the controller of the system with a new one, a cRIO which has field-programmable gate array (FPGA). This is necessary because the current controller has a relatively low performance and memory and it lacks an FPGA.

**FOIL TARGET**

The foil target is used to examine the effects of proton irradiation of thin foil targets. A 25 microns thick Havar foil of 20 mm diameter is irradiated by an intense proton beam. Another side of the foil is cooled by liquid metal NaK alloy, which is in contact with a water cooled stainless steel plate.
The Foil target control hardware is located in a mountable rack in the RF Hall (approximately 30 meters from the target). It contains an NI compact reconfigurable Input/output (cRIO) 9073 control system. The cRIO includes three components: a real-time controller, a reconfigurable FPGA, and industrial I/O modules. The Foil control systems HMI (figure 2) is divided to two parts, the first of which operates and controls the water cooling systems. This part presents the differential pressure gauge, temperature of the water in different locations along the pipe line, ultrasonic water flow meter, pressure gauge, water valve control and water flow switch. The second part in the HMI presents the target parameters such as target currents, temperatures in several locations (measurement by 4 thermocouples) and target temperature as measured by a pyrometer sensor. The control systems enable the user to direct the pyrometer to a specific point on the target or to perform a two dimensional scan. The scan system consists of two drivers (MicroMaxTM Series 671) and of two accurate mirror positioning systems.

![Figure 2: foil target (left) and LiLit (right) control systems.](image)

The control system provides a "target ready" signal for the accelerator Machine Safety System (MSS). We therefore used the FPGA, which provides high reliability, high speed and determinism. The "target ready" signal is set by the chain of the following signals: water flow value, pyrometer temperature reading and flow switch status. The "target ready" signal ensures that the target cooling system is operating and that the target temperature is in the permitted level.

**SUMMARY**

The SARAF beam lines and target control systems consists of various hardware design by the SARAF engineering team, and are currently in various stages of operation and delivery.

**REFERENCES**

The 2011 Revised Basic Safety Standards

J. Koch*

Radiation Safety Division, Soreq Nuclear Research Center, Yavne, Israel

koch@soreq.gov.il

INTRODUCTION

The BSS (International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources) are published by the IAEA (International Atomic Energy Agency) and other cosponsoring international organizations. This document is the best known publication of the IAEA and it constitutes the basis for national regulations in radiation protection (for occupational, public and medical exposures).

In parallel with the publication of the 2007 Recommendations of the ICRP (International Commission on Radiological Protection)(1), the IAEA – together with the cosponsors of the BSS – began a revision process in light of the new ICRP recommendations and other developments in radiation protection since the publication of the 1996 BSS(2).

The revised version was approved for publication by the Board of Governors of the IAEA in September 2011 and published as interim version in November 2011(3). The final version will be published after approval by the cosponsoring organizations.

MAIN CHANGES

The following main changes were adopted in the revised BSS, as compared to the 1996 BSS.

Adoption of the ICRP revised system of radiological protection

In its 2007 Recommendations, the ICRP proposed to divide the network of events and situations causing exposure to ionizing radiation into three types of exposure situations (planned, emergency and existing exposure situations), which address all conceivable circumstances and replace the former division into practices and interventions, conceived in the ICRP 1990 Recommendations(4). The ICRP further distinguished between three categories of exposures: occupational exposures, public exposures and medical exposures of patients.

The revised BSS adopted the revised system of radiological protection of the ICRP. This is reflected in the structure of the document which includes five sections. After an introductory section describing the ICRP system, section 2 presents generic requirements that pertain to all three exposure situations. The next three sections set out the requirements that are specific to each of the three exposure situations. For each type of exposure situation, the requirements are further grouped into requirements for occupational exposure, public exposure and (for planned exposure situations only) medical exposure.

Exposure due to radon indoors and in workplaces

The ICRP published in 2009 its Statement on Radon(5), in which it presented an updated risk estimate of lung cancer from residential exposure to radon that is almost twice the former

* The author is member (as Israel representative) of the Radiation Safety Standards Committee (RASSC) of IAEA, which was the leading committee overseeing the revision of the BSS.
estimate. This finding is consistent with those of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR)\(^6\) and the World Health Organization (WHO)\(^7\).

The revised BSS requires the establishment of reference levels, which are based on the updated risk estimates, as follows:

- For dwellings and other buildings with high occupancy factors for members of the public, such as schools and hospitals, the government shall ensure that an action plan is established, including a reference level that will not exceed an annual average activity concentration of \(^{222}\text{Rn}\) of 300 Bq/m\(^3\), corresponding to an annual effective dose of the order of 10 mSv.
- For workplaces, the regulatory body shall establish a protection strategy, including a reference level that will not exceed an annual average activity concentration of \(^{222}\text{Rn}\) of 1,000 Bq/m\(^3\), also corresponding to an annual effective dose of the order of 10 mSv.

Moreover, in line with the ICRP approach with regard to reference levels, the revised BSS requires protection optimization below the reference levels.

**New dose limit for the lens of the eye in occupational exposure**

In April 2011 the ICRP published a Statement on Tissue Reactions\(^8\), in which it stated that recent epidemiological evidence suggests that the threshold dose for the lens of the eye is lower than previously considered. It recommended "for occupational exposure in planned exposure situations … an equivalent dose limit for the lens of the eye of 20 mSv in a year, averaged over defined periods of 5 years, with no single year exceeding 50 mSv."

Since the final draft of the revised BSS had already been agreed upon before the ICRP Statement, an extraordinary procedure was applied in order to incorporate the new dose limit into the revised BSS, after a brief consultation of IAEA Member States took place.

**Exemption and clearance values**

The 1996 BSS included exempt activities and exempt activity concentrations for a long list of radionuclides, while it mentioned that use of the exempt activity concentrations is limited to moderate amounts of material, i.e. amounts at the most of the order of a ton. Exempt activity concentrations for bulk amounts of material were developed later and published in an IAEA Safety Guide\(^9\). These data were incorporated into Schedule I of the revised BSS, which includes three tables:

- Activity concentrations and activities of radionuclides for exemption of moderate amounts of material.
- Activity concentrations of radionuclides of artificial origin for exemption of bulk amounts of material and for clearance of material.
- Activity concentrations of radionuclides of natural origin for clearance of material.

**Categorization of sealed radiation sources**

A categorization scheme of sealed radiation sources, including five categories, was devised to provide a simple system for ranking sources in terms of their potential to cause harm. The scheme, which was previously set out in an IAEA Safety Guide\(^{10}\), was incorporated into Schedule II of the revised BSS and its use is therefore required from registrants and licensees.
CONCLUSIONS
The revised BSS is more user-friendly than the 1996 BSS, mainly due to its being structured on the basis of the three types of exposure situations and the three categories of exposures, following the ICRP 2007 recommendations. It is a stand-alone document, due to the inclusion of an extensive definitions section and of a CD-ROM containing tables for calculating doses in external and internal exposure.

REFERENCES
INTRODUCTION
In facilities where workers are exposed to ionizing radiation, a system of radiation protection is implemented which is ultimately based on a balancing of the risks and benefits to the uses of radiation involved. More restrictive dose limits are set for members of the public than for workers and these have been applied also to conceptus.

There are many different industries and practices that involve the use of ionizing radiation and the potential exposure of female workers at fertility age. The radiation sources and levels of exposure may vary significantly in different industries, such as those associated with the nuclear fuel cycle (where the workforce is mostly male), industrial radiography, diagnostic and therapeutic medical applications (where the number of female workers may be greater than males) and others.

The International Commission on Radiological Protection (ICRP) and the International Atomic Energy Association (IAEA) in its Basic Safety Standards (BSS) recommend to regard the conceptus as a member of the public when considering the protection of female workers who are pregnant (1). There is a requirement that women should be informed of the importance of early notification of pregnancy.

In 2001, the ICRP in its Publication 88 (2) published biokinetic and dosimetric models for the calculation of doses to the conceptus from intakes by the mother. Several internal scenarios were considered, which include acute and chronic intakes by inhalation and ingestion for female workers and members of the public. For acute exposures, intakes were taken to occur at 2.5 years and 6 months before conception, at the time of conception, and at the end of weeks 5, 10, 15, 25 and 35 of the pregnancy.

In 2005, the ICRP in its Publication 95 (3) published biokinetic and dosimetric models for the calculation of doses to the infant from intakes by the breastfeeding mother. This work examines the significance of pregnant and breastfeeding women work with radioactive materials on the potential dose to their conceptus or children. The calculations are based on the data from ICRP Publications 88 and 95.

RESULTS

Pregnant women
The dose to the conceptus from inhalation of radioactive materials by the pregnant mother was calculated, based on the assumption that the intake took place at the pregnancy stage in which the dose to the conceptus was the highest.

Table 1 presents the calculated results for radioactive materials that are used in medicine or research laboratories.
Table 1. The dose ratio between the conceptus and the mother from inhalation of radioactive materials by the pregnant mother during pregnancy.

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Absorption Type</th>
<th>Conceptus age at the time of intake (weeks)</th>
<th>Dose ratio between the conceptus and the mother from intake by the mother</th>
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</thead>
<tbody>
<tr>
<td>H-3</td>
<td>Water</td>
<td>10</td>
<td>2.00</td>
</tr>
<tr>
<td></td>
<td>OBT</td>
<td>10</td>
<td>1.88</td>
</tr>
<tr>
<td>C-14</td>
<td>Vapor</td>
<td>10</td>
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<tr>
<td>S-35</td>
<td>F</td>
<td>10</td>
<td>1.11</td>
</tr>
<tr>
<td></td>
<td>M</td>
<td>10</td>
<td>0.068</td>
</tr>
<tr>
<td>Ca-45</td>
<td>M</td>
<td>15</td>
<td>2.39</td>
</tr>
<tr>
<td>Fe-59</td>
<td>F</td>
<td>15-10</td>
<td>0.9</td>
</tr>
<tr>
<td></td>
<td>M</td>
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<tr>
<td>Co-57</td>
<td>M</td>
<td>0</td>
<td>0.213</td>
</tr>
<tr>
<td></td>
<td>S</td>
<td>0</td>
<td>0.097</td>
</tr>
<tr>
<td>Co-60</td>
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<td>0</td>
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</tr>
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<td>S</td>
<td>5-0</td>
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<tr>
<td>Ni-59</td>
<td>F</td>
<td>0</td>
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</tr>
<tr>
<td></td>
<td>M</td>
<td>0</td>
<td>0.149</td>
</tr>
<tr>
<td>Ni-63</td>
<td>F</td>
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<td>M</td>
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<td>Zn-65</td>
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<td>0.607</td>
</tr>
<tr>
<td>Se-75</td>
<td>F</td>
<td>15-10</td>
<td>1.00</td>
</tr>
<tr>
<td></td>
<td>M</td>
<td>15-10</td>
<td>0.706</td>
</tr>
<tr>
<td>Sr-89</td>
<td>F</td>
<td>25</td>
<td>8.57</td>
</tr>
<tr>
<td></td>
<td>S</td>
<td>25</td>
<td>0.048</td>
</tr>
<tr>
<td>Sr-90</td>
<td>F</td>
<td>25</td>
<td>1.07</td>
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<tr>
<td></td>
<td>S</td>
<td>25</td>
<td>0.0096</td>
</tr>
<tr>
<td>Tc-99m</td>
<td>F</td>
<td>25</td>
<td>0.450</td>
</tr>
<tr>
<td></td>
<td>M</td>
<td>25</td>
<td>0.248</td>
</tr>
<tr>
<td>I-125</td>
<td>F</td>
<td>25</td>
<td>0.849</td>
</tr>
<tr>
<td>I-131</td>
<td>F</td>
<td>35</td>
<td>2.36</td>
</tr>
<tr>
<td>Cs-137</td>
<td>F</td>
<td>0</td>
<td>0.522</td>
</tr>
<tr>
<td>Ra-226</td>
<td>M</td>
<td>25</td>
<td>0.114</td>
</tr>
<tr>
<td>U-238</td>
<td>F</td>
<td>10</td>
<td>0.310</td>
</tr>
<tr>
<td></td>
<td>M</td>
<td>10</td>
<td>0.022</td>
</tr>
<tr>
<td></td>
<td>S</td>
<td>10-0</td>
<td>0.0002</td>
</tr>
</tbody>
</table>

*Breastfeeding women*

The dose to the infant from inhalation of radioactive materials by the breastfeeding mother was calculated, based on the assumption that the intake took place 10 weeks after birth (around the time of the mother returning to work after birth).

Table 2 presents the calculation results for radioactive materials that are used in medicine or research laboratories.
Table 2. The dose ratio between the infant and the mother from inhalation of radioactive materials by the breastfeeding mother 10 weeks after birth.

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Absorption Type</th>
<th>Dose ratio between the conceptus and the mother from intake by the mother</th>
</tr>
</thead>
<tbody>
<tr>
<td>H-3</td>
<td>Water</td>
<td>1.22</td>
</tr>
<tr>
<td></td>
<td>OBT</td>
<td>0.83</td>
</tr>
<tr>
<td>C-14</td>
<td>Vapor</td>
<td>0.55</td>
</tr>
<tr>
<td>Ca-45</td>
<td>M</td>
<td>0.41</td>
</tr>
<tr>
<td>Fe-59</td>
<td>F</td>
<td>0.037</td>
</tr>
<tr>
<td></td>
<td>M</td>
<td>0.008</td>
</tr>
<tr>
<td>Co-57</td>
<td>M</td>
<td>0.20</td>
</tr>
<tr>
<td></td>
<td>S</td>
<td>0.043</td>
</tr>
<tr>
<td>Co-60</td>
<td>M</td>
<td>0.23</td>
</tr>
<tr>
<td></td>
<td>S</td>
<td>0.025</td>
</tr>
<tr>
<td>Ni-59</td>
<td>F</td>
<td>0.064</td>
</tr>
<tr>
<td></td>
<td>M</td>
<td>0.034</td>
</tr>
<tr>
<td>Ni-63</td>
<td>F</td>
<td>0.067</td>
</tr>
<tr>
<td></td>
<td>M</td>
<td>0.026</td>
</tr>
<tr>
<td>Zn-65</td>
<td>S</td>
<td>0.15</td>
</tr>
<tr>
<td>Se-75</td>
<td>F</td>
<td>1.29</td>
</tr>
<tr>
<td></td>
<td>M</td>
<td>0.88</td>
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<tr>
<td>Sr-89</td>
<td>F</td>
<td>1.64</td>
</tr>
<tr>
<td></td>
<td>S</td>
<td>0.006</td>
</tr>
<tr>
<td>Sr-90</td>
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<td>0.57</td>
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<tr>
<td></td>
<td>S</td>
<td>0.004</td>
</tr>
<tr>
<td>Te-99m</td>
<td>F</td>
<td>1.00</td>
</tr>
<tr>
<td></td>
<td>M</td>
<td>0.69</td>
</tr>
<tr>
<td>I-125</td>
<td>F</td>
<td>1.15</td>
</tr>
<tr>
<td>I-131</td>
<td>F</td>
<td>2.46</td>
</tr>
<tr>
<td>Cs-137</td>
<td>F</td>
<td>0.22</td>
</tr>
<tr>
<td>Ra-226</td>
<td>M</td>
<td>0.008</td>
</tr>
<tr>
<td>U-238</td>
<td>F</td>
<td>0.007</td>
</tr>
<tr>
<td></td>
<td>M</td>
<td>0.0007</td>
</tr>
<tr>
<td></td>
<td>S</td>
<td>0.000005</td>
</tr>
</tbody>
</table>

**CONCLUSIONS**

A calculation was conducted on the dose ratio between the conceptus or infant and the pregnant or breastfeeding women from inhalation intake of radioactive materials by the women during pregnancy and breastfeeding stage. For some materials it was found that the dose to the conceptus or infant was higher than that to the mother and therefore working with these materials must be forbidden during pregnancy or breastfeeding period. Working with other materials should be allowed with special consideration.
REFERENCES
A New High-Purity Germanium based Whole Body Counter for Internal Dose Assessments at Soreq Nuclear Research Center

G. Haquin\(^1\), H. Datz\(^1\), T. Riemer\(^1\), Z. Yungrais\(^1\), Y. Ben-Hemo\(^2\), D. Katz\(^1\), E. Sayag\(^3\) and D. Rendlich\(^3\)

\(^1\) Radiation Safety Division, Soreq Nuclear Research Center, Yavne, Israel
\(^2\) Applied Physics Division, Soreq Nuclear Research Center, Yavne, Israel
\(^3\) Workshop Unit, Soreq Nuclear Research Center, Yavne, Israel

Corresponding Author: gustavo@soreq.gov.il

INTRODUCTION
The occupational dose assessment of radiation workers in Israel is performed by different means according to the characteristics of the exposure. For external exposure situations the use of thermoluminescence (TLD) dosimetry is the common practice\(^{(1)}\) while for internal dosimetry radio-toxicology and organ or whole body counting (WBC) methods are routinely performed.

Measurements of the radioactivity in urine are performed at the ISO 17025 accredited Radio-toxicology Laboratory of the Radiation Safety Division (RSD) at Soreq Nuclear Research Centre (SNRC).

Whole body counting is a well established method for the measurement of the remaining radioactivity in the body and consequently the internal dose assessment. During more than forty years the measurements at SNRC were performed using low resolution gamma spectrometry system. A new system was designed to improve the performance of the WBC in three aspects: radionuclide identification; lower detection limits and versatility of measurement, i.e. whole body scanning or organ monitor (lung counter, thyroid monitor, etc.). The new high resolution gamma spectrometry WBC system was designed and installed at the RSD in 2010. This paper describes the different components of the new system, the background reduction solution and first measurements performed.

WHOLE BODY COUNTING SYSTEM
The whole-body counting facility at SNRC consists of three main components: four High-Purity Germanium (HPGe) detectors, the lead shielding and a controlled moving bed.

The system configuration consists of two detectors mounted above the bed and two detectors below it, arranged around the same longitudinal and lateral axis. This type of configuration used in several WBCs worldwide\(^{(2)}\) reduce the detection limits of the system and improves the localization of the radionuclides distribution in the body. Top, side and perspective views of the SNRC WBC system are shown in Figure 1.
Detectors
The upper detectors consist of two ACTINIDE-85 detectors (Ortec, Oak Ridge, TN) with cryostat (Model No. CFG-LB-GEM-C-GG) which has a low-level background, a carbon fiber endcap all mounted on 3.5 liter liquid nitrogen (LN$_2$) high-fill dewars. The lower detectors are of identical type as the upper detectors but mounted on a dual neck high capacity (~43 liters) dewar. This type of detectors combine a large frontal area (>56 cm$^2$) with good energy resolution across a wide range of energy from ~ 10 keV to more than 2000 keV. The measured energy resolution of the detectors is 0.44-0.58 keV and 1.70-1.88 keV at 14.4 keV ($^{57}$Co) and 1332.5 keV ($^{60}$Co) respectively. The thin entrance window (~10 microns Ge equivalent) maximizes the low energy efficiency while the large frontal area and thickness (> 30mm) ensure a high relative efficiency of more than 50% (at 1332.5 keV).

The positive high voltage of the detectors is supplied by two ORTEC Model 660 dual 5-kV Detector Bias Supply. The output voltage is turned off when an appropriate signal is delivered by a warm-up sensor in the cooled Ge detector. Nuclear pulses from the preamplifier are amplified and shaped (6 μsec) by ORTEC Model 672 Spectroscopy Amplifier. Spectra of 8192 channels are processed by a two ORTEC ASPEC 927 dual input MCA. The RENP-B32 Renaissance-32 Whole-Body Counter Software control and process the acquired data either by summing the four spectra or by analyzing each spectrum separately.

The LN$_2$ filling is performed by a programmed automatic dispensing system.

Shielding
Several WBC's are installed in a shielded room usually made of old steel. Due to technical limitations of the counting room, the shielding was designed in such way that radionuclides mostly from the body are detected by the HPGe's. The area in the proximity of the detectors was shielded from all sides using regular lead at two longitudinal walls (100 mm x 1000 mm x 2000 mm) and at the lower detectors hole a lead shielding (50 mm x 700 mm x 700 mm) was constructed. Special lead shielding sleeves (∞ shaped) were designed for both the upper and lower detectors. Each lead sleeves was constructed using 150 kg of ultra-low activity lead bricks.
(Plombum, Krakow, Poland), with activity concentration of $^{210}\text{Pb}$ is 2.8±0.3 Bq/kg. The inner part which is in contact with the detectors were lined with OFHC Copper of the C10100 grade and thickness of 5 mm. The purity of copper in this grade is 99.99%. The system structure was made of steel salvaged from a naval ship of World War II, screened by gamma spectrometry to verify they are clean of $^{60}\text{Co}$ contamination (3).

Bed
The bed is made of a thin wooden board, about 5mm in thickness with a minimal distance between the bed and the lower detectors. The bed moving is controlled by a two axis electronic controller (Galil controller model 4020). The scanning speed can be determined using ORTEC Motor Setup program from 5 to 200 mm per min. Static measurements as lung counting or thyroid monitoring can be performed by setting the upper detectors at the shortest distance from the measured organ. A second electronic controller of the same model governs the up-down movement of the upper detectors.

PRELIMINARY MEASUREMENTS

Energy and efficiency Calibration
All four detectors are energy calibrated in the range 14.4 to 1836.1 keV using a multiline gamma source having $^{57}\text{Co}$, $^{210}\text{Pb}$, $^{241}\text{Am}$, $^{109}\text{Cd}$, $^{57}\text{Co}$, $^{137}\text{Ce}$, $^{113}\text{Sn}$, $^{85}\text{Sr}$, $^{137}\text{Cs}$, $^{54}\text{Mn}$, $^{65}\text{Zn}$, $^{88}\text{Y}$ and $^{60}\text{Co}$. The energy resolution of each detector cover a range from 0.45 to 0.58 keV at 14.4 keV ($^{57}\text{Co}$) to 1.70 to 1.88 keV at 1332.5 keV ($^{60}\text{Co}$).

Preliminary efficiency measurements were performed using a multiline gamma source covering the energy range of 46.5 to 1836 keV (filter disk of 70mm diameter and 6 mm height provided by National Physical Laboratory).

The efficiency for the measurement of $^{131}\text{I}$ in the thyroid was evaluated using a small volume liquid source (QCY 58, Amersham Ltd.).

Background measurements
Long background measurements were performed at different configurations: 1) system background without the bed between the detectors, 2) background measurement with static bed between the detectors and 3) moving bed background measurement.

The background count rate for the whole energy range (14 – 2800 keV) was found to be ~ 10.5-13.2 counts per second per detector and ~12 cps/kg Ge for all four detectors.

The major radionuclides found in the background belong, as expected, to the $^{226}\text{Ra}$ and $^{232}\text{Th}$ decay chains and $^{40}\text{K}$ as seen in Figure 2.

Figure 2: Background measurement of detector D1.
First operational measurements
The first operational measurements with the new WBC were performed at the end of March 2011, about 3 weeks after the Fukushima Daiichi Nuclear Power Plant accident. Two type of measurements were performed: a 25 minutes thyroid gland monitoring using the upper detectors positioned just above the examined person for the detection of $^{131}$I, and a 30 minutes whole body counting with a scanning bed for the detection of $^{134}$Cs and $^{137}$Cs. Figure 3 describes a spectrum taken at a static bed mode where traces of $^{131}$I were detected giving a negligible internal dose.

**Figure 3:** Traces of $^{131}$I found in spectrum of Israeli embassy delegate in Tokyo, March 2011.

CONCLUSION
A new WBC facility was constructed and established at SNRC. It consists of four large area broad energy HPGe detectors, having the possibility to perform scanning bed whole body counting and static bed organ counter (such as thyroid gland monitor, lung counter, etc.). Future work on the WBC will include: background reduction methods, calibration using standard phantoms, HPGe simulation using Monte Carlo based codes and source location using the four detectors (4).

REFERENCES
INTRODUCTION
Nuclear technology and nuclear energy in Israel and in other countries as well, are suffering from “bad reputation”, originating from both ignorance and misleading publicity.
In order to provide the public, and especially the young generation, with objective and comprehensive facts and data, the Israeli Atomic Energy Commission (IAEC), and the Nuclear Research Center – Negev (NRCN) decided to create a permanent exhibition in Beer Sheva, covering Nuclear Technology. Similar activity was done in other countries, such as France, Slovenia and others.

THE PROJECT
Our vision:
Providing information and bringing the multifaceted aspects of the Nuclear Technology up close and to the younger generation and the general public.
Creating an enjoyable and interactive experience for the exploration of the world of nuclear science and understanding its multiple applications, in order to promote advancement in this discipline.
The goals we defined for the exhibition:
• Demystification and eliminating the fear associated with the nuclear field
• Inspire and promote curiosity, discovery, and learning through exhibitions and education
• Encouraging the use of nuclear energy as a “green” energy
• Raise the awareness and understanding of this field in the community
• Encouraging young people to study nuclear engineering.
• Creating a virtual “Visitors Center” for the NRCN

We decided to include the following contents areas:
• The history of the nuclear science, including the Israeli history
• The Atom and its Building blocks
• Isotopes and nuclear radiation
• Radiation and matter, properties of various radiation types
• Fission and chain reaction
• The nuclear reactor – for research and energy
• Applications of nuclear technology: medicine, industry, agriculture

While looking for a hosting establishment, we joined forces with the Rashi Foundation, which was in the process of creating the Carasso Science Park in Beer Sheva. Rashi Foundation embraced the idea, and allocated the whole top floor of the old Turkish building to the exhibition. Following an extensive process of conceptual design, we selected a design firm with relevant
experience: Breeze Creative Ltd.\(^{(4)}\) (which engaged Prof. Yigal Ronen as a scientific advisor). The design process is based on team work: our scientific committee, Rashi Madarom experts and the professional team of Breeze Creative.

THE EXHIBITION
The exhibition will occupy the top floor of a restored building from the Turkish era of Beer Sheva (around 1914), with an area of about 400 sq. meters. The planned layout is portrayed in Figure 1. The visitors enters the first hall (A), and proceed from an introduction about the atom (B), through a building blocks game where they create isotopes (C), to the radiation properties, the U\(^{235}\) fission and the chain reaction, and the reactor demonstration (D). On the other side they will learn about nuclear applications, participate in a power stations comparison game (E), learn about nuclear medicine (F) and finally watch a movie at a small auditorium (G), summarizing the visit.

Figure 1 the Exhibition Layout
In *Figure 2* we can see an artist view of the exhibition. Most of the exhibits are interactive, and are meant to attract the visitors and leave a lasting impression. We plan to extend the visit by linking it to a web site where the visitors will be able to watch pictures from their visit, and most important – get more comprehensive information on the subjects covered during the visit. All the explanations and text will be in three languages: Hebrew, Arabic and English.

The exhibition is planned to open towards the end of this year, as part of the new Science Center. We expect a large number (about 160,000 annually) of visitors, including groups of 4th to 9th grade students, and visitors from the general public.

**REFERENCES**


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(4) Breeze Creative Ltd. Moshav Bnai atarot, POB 254, Israel 60991
Egyptian Nuclear Activities at Research Reactors, Laboratories, and Uranium Production (independent review)

Prof. Dr. Ashraf Elsayed Mohamed Mohamed

*Western Central University in Middleeast, AGS. Alexandria, Egypt*

Corresponding Author: ashraf_el@inbox.com

In this paper, I discuss and analysis the Egyptian nuclear activity at research reactors, laboratories and uranium production as follows:

1. **Laboratories:** External beta and gamma radiation is the main source of radioactivity in laboratories. Also, in some cases measurable radon-gas concentration are available as follows; *radon gas measurement* - A portable radioactive gas monitor RGMI/L was used. It is sensitive to alpha particles emitted from radon gas. The unit is picocurie / litre (pci/L) or microcurie per cubic meter (mci/m³). The maximum permissible concentration is 0.03 m ci / m³. I discuss the full activities of Egyptian laboratories in anshass sites (two sites) with the future activities.

2. **Uranium production** - The concentration of a semi-pilot plant for the extractions of uranium from phosphoric acid has been completed and was expected to be commissioned during 1999 and after. The design capacity of the plant is about 15 m³/day (official details) but the fact is different of acid containing about 65 ppm uranium. The process is the adjustment stage. The nuclear materials authority is taking over the responsibility for the exploitation of the black sand deposits at the Rosetta beach on the Mediterranean coast. These deposits contain monazite, zircon and rutile, as well as ilmenite and magnetite. The proposed projects includes wet and dry mills with a capacity for treating 200 m³ / hour (official details) but the fact is different, of wet sand. The area to be evaluated is estimated to contain about six million tonnes of economic heavy minerals at an average grade of 2% (fact is different). This resource contains about 3000 tonnes of monazite whose U content could be classified as EAR-II. The monazite contains 0.46% U (fact is over that rate) and 6.05%Th, As well as 65% REE at location 6 - The nuclear materials authority concentrated its main explorations activities in the development of three mineralized areas discovered in the Eastern desert and Sinai; gabal gattar, El missikat and El erediya, and Abu zeneima as well as Um Ara, G. kadabora, Rosette and Sinai. I discuss this activities in details.

3. **Radon daughters concentration:** Tri met (TM 372-A), portable alpha counter with suitable pump was used. It has a time selector and displays digital reading. The detector is of ZnS (Ag) scintillator type. The instrument was calibrated with radon daughter source (Po 214) and checked before use by Am 241 alpha standard source. Radon daughter were measured using Kuznetz grab sampling method. The maximum permissible radon daughters concentration is 0.3 working level(WL). The total cumulative energy of 1.3. 10 5 Mev as alpha particle energy defines the working level as any combination of short-lived decay product of radon gas (RaA, RaB, RaC and RaD) in one litre of air that will result in the ultimate emission. The working level Month (WLM) is an exposure unit, which is expressed as the product of WL and duration of exposure, normalised to one month which is defined as 170 hours, i.e. 1 WLM = 170 WLH.
4. Monitoring of external gamma radiation: A portable radiation measuring instrument as Geiger counter type, Berthold LB-1200 was used. It gives direct reading for dose rate in millirem/hour. It was calibrated using standard Cs 137 gamma source. It is defined that the maximum permissible value for the dose rate is 2 millirem/hours (Mrem/h) or 20 micro siverts/hour (\(\mu\text{Sv/h}\)) show the whole operations in this field.
What Affects the Isotopic Composition in Rainfall – A New Interpretation

A. Dody
NRCN, P.O.BOX 9001 Beer Sheva84190

Corresponding Author: dodik@bgu.ac.il

INTRODUCTION
Stable isotopes are used as tracers in atmospheric water in order to improve our understanding of the hydrological cycle. There is large number of observations of stable isotopes in precipitation with monthly time resolution all over the globe\(^{(13)}\). It is well known that temperature, altitude, latitude and geographic feature in the path of the air parcels affect the isotopic composition\(^{(5)}\). Other, e.g.,\(^{(4,8)}\) suggested also parameters such evaporation, relative humidity (RH) and sea surface temperature (SST). Levin\(^{(10)}\) stated that the isotopic composition is relatively uniform over a wide area on any particular day, but differs substantially from storm to storm. Later on, Dody\(^{(6)}\) and Adar\(^{(1)}\) showed in their study, done in the same geographical area as Levin\(^{(10)}\), that isotopic composition differs dramatically even in a single rain cell. Phahl\(^{(11)}\) discussed the effect of the trajectory on stable isotopes in water vapor in the eastern Mediterranean. They calculated the synoptic system trajectory based on the RH as measured in the last point. Dody\(^{(7)}\) also mentioned in general the affect of the synoptic-scale trajectory on the isotopic composition. It is obvious today that in order to improve our understanding of fractionation process under atmospheric conditions, sampling in high resolution is needed. Only few measurements are available\(^{(8,3)}\). This study presents high resolution sampling of isotopes in precipitation under desert conditions. The study region is the central part of the Negev Desert, Israel, with average annual precipitation of 86 mm, to 191 mm, distributed during 20 rainy days per season on average\(^{(14)}\). The research ran during four rainy seasons. All of the rain samples were analyzed at the International Atomic Energy Agency (IAEA). Samples of three rain storms were analyzed at both the IAEA Laboratory in Vienna and at the Niedersachsisches Landesmat for Forchung (NLFB) Laboratory in Hannover, Germany. A standard deviation of ±1‰ for δD and ±0.1‰ for δ\(^{18}\)O, are acceptable accuracy for δD and δ\(^{18}\)O analyses, respectively.

METHODOLOGY
A mechanical sequential rain sampling with high resolution in time was used to collect approximately 250 to 500 ml of water equivalent to 1-2 mm of rain, respectively\(^{(1)}\). The novel approach to this method is that once an assigned volume of water is collected, the self-potential energy of the loaded containers is used to operate a mechanical seal for isolating the rain sample from evaporation. The sampler has 20 rain bags allowing the collection of rain water from a storm with 20 to 40 mm.

RESULTS
Eight rainfall (RF) events were sampled in high resolution for isotopic composition during four rainy seasons from 1990-1993, in the Central Negev, Israel\(^{(7)}\). In the all eight storms, the isotopic composition of RF varied dramatically from enriched to depleted and vice versa in each storm. Two modes of δ\(^{18}\)O distribution during the rain were characterized:
a. Wave shape distribution (Fig. 1). The $\delta^{18}O$ is changed gradually during the rain from enriched to depleted values and vice versa.

\[\begin{align*}
\text{Sequential sampling} \\
\begin{array}{cccccccccccccccc}
1 & 2 & 3 & 4 & 5 & 6 & 7 & 8 & 9 & 10 & 11 & 12 & 13 & 14 & 15 \\
\hline
0 & -2 & -4 & -6 & -8 & -2 & -4 & -6 & -8 & 0 & -2 & -4 & -6 & -8 & 0
\end{array}
\end{align*}\]

Figure 1. Wave shape distribution of the isotopic composition during the rain.

b. Constant $\delta^{18}O$ values within each rain cell but significantly different among different rain cells (Fig. 2).

\[\begin{align*}
\text{Sequential sampling} \\
\begin{array}{cccccccccccccccc}
1 & 2 & 3 & 4 & 5 & 6 & 7 & 8 & 9 & 10 & 11 & 12 & 13 & 14 & 15 \\
\hline
-10 & -8 & -6 & -4 & -2 & 0 & 2 & 4 & 6 & 8 & 10 & 12 & 14 & 16 & 18
\end{array}
\end{align*}\]

Figure 2. Step function distribution of the isotopic composition of the rain.

CONCLUSIONS
Synoptic systems combine air parcels with different trajectories and origins to produce rain cells with different isotopic compositions. In a given geographical location a combination of those produce changes in the isotopic composition in the rain as observed. The results suggest that the isotopic composition in precipitation is controlled also by air parcel trajectories where mixing or no, between the air parcel can be occurred. The isotopic composition of each air parcel can be saved along the trajectory with no mixing although the high air turbulence vertically and horizontally (Fig. 3).
Figure 3. Air parcels containing rain cells with different isotopic compositions showing:

a. Partial mixing between the rain cells. b. No mixing between the rain cells.

REFERENCES

Fallout of the Fukushima Accident on commercial products imported to Israel

E. Neenan, B. Magoz, B. Ben Malka

Yavne, Israel System Advanced Laboratories

Corresponding Author: dr_ehudn@netvision.net.il

During May 2011, several gamma spectrometric measurements were performed with nylon samples used as covers of typewriter machines shipped to Israel from Japan. A Geiger meter measurement of the nylon cover showed 50 µRem/h.

Table 1 shows some of the radionuclides measured by gamma spectrometry. These radionuclides are typical signatures to a release in an accident of a nuclear reactor and are therefore assigned to the Fukushima crisis (March 2011).

Table 1: Results of gamma measurements

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>Energy [keV]</th>
<th>Half Life [Years]</th>
<th>Activity [mBq/cm²]</th>
<th>Activity [Bq/g]</th>
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<tbody>
<tr>
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<td>Cs – 134</td>
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<td>2.062</td>
<td>24.52</td>
<td>5.13</td>
<td></td>
</tr>
<tr>
<td>Cs – 137</td>
<td>661.62</td>
<td>30.1</td>
<td>28.87</td>
<td>6.04</td>
<td></td>
</tr>
<tr>
<td>I – 131</td>
<td>364.48</td>
<td>8.021</td>
<td>KeV 364by</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Measurement uncertainty is 5.6% at a 95% confidence level (2 sigma).

CONCLUSIONS

According to ICRP 82 and BSS-115, the use of imported products, should be restricted under the application of the radiation safety standards. The contamination of the Nylon Cover can be treated as PRACTICE. Therefore, the Limit to the general public is 0.3 mSv/y. A person working 24 days, 8 hours per day being exposed to 15 µRem/h, is yearly exposed to 0.034 µSv. There is no risk in working with the contaminated material from Japan.

REFERENCE

1. Dr. Victor Steiner, Ministry of the Environment Communication. - Private
From W. Rothenstein and J. Rowlands towards Improved Solutions of the Boltzmann Transport Equation

R. Dagan

Institute for Neutron Physics and Reactor Technology,
Karlsruhe Institute of Technology Campus North,
Hermann-von-Helmholtz-Platz 1,
76344 Eggenstein-Leopoldshafen

Corresponding Author: dagan@kit.edu

INTRODUCTION
The Boltzmann transport equation is commonly used considerable heterogeneous core design. In such cases the fluctuation in the neutron direction and neutron energy affects to a greater extent the core parameters. The scattering kernel formalism is a part of the source expression in the Boltzmann equation. It deals with those changes in energy and direction of neutrons undergoing a scattering interaction. The exact description of the scattering process is however quiet complicated and for practical reasons several approximations have been “established” in the last 6-7 decades. Some concerns the chemical binding effects (Detailed structure) of the scattering material for light isotopes which are mentioned for example in (1). It is interesting to note the remark of Duderstadt concerning the scattering kernel for light isotopes: Although “one introduces several horrifyingly brutal approximations, at least to the solid state physicist, for nuclear engineer they are acceptable” see (1).

Other approximations concern the handling of the interaction of neutrons with heavy isotopes which have well pronounced resonances. In this case it was common to ignore completely the presence of the resonances and the thermal agitation of the interacting nuclide. Strictly speaking the well known Doppler Broadening effect which is done on the integral scattering Cross section (Loss term) was ignored for the differential part which is the source term on the right side of the very same Boltzmann equation. Evidently an inconsistency arises between the Doppler Broadening treatment within the loss and production terms. This problem was technically solved by the well known processing code NJOY(2). Yet as was stated by the author of the NJOY code - R. Macfarlene- in the manual of the THERMR module (2): “the secondary energy distribution (namely the scattering kernel treatment for heavy isotopes) is still incorrect”.

The third type of approximations concern the numerical available approach namely the use of Legendre moments for the scattering kernel in deterministic solutions and in parallel the stochastic approach known as “Sampling the velocity of the target” in Monte Carlo codes(3). In particular for the stochastic treatment it is worthwhile to quote the warning of D. Cullen in his famous report (4) on thermal scattering: “Aren’t our Monte Carlo codes more or less perfect? Far from it….they are limited by the nuclear data they use …. and ultimately how they interpret data…. The limitation can severely affect the results produced by our Monte Carlo Codes”.

In the last century, indeed, a lot of effort was inserted in better handling of the scattering kernels and in particular for the solid state effects of light isotopes satisfactory results (as far as nuclear engineer are concerned) were achieved.
W. Rothenstein (5) and J. Rowlands (6) devoted in the last 3 decades a lot of work on the improvement of the treatment of scattering by heavy isotopes with pronounced resonances. W. Rothenstein was the first one to suggest a stochastic consistent solution (7) of the transport equation and further developed the analytical expression for the energy and temperature dependent double differential cross section. Both solutions refer to a free gas model as far as the thermal agitation of the nuclei is concerned.

J. Rowlands collaborated with W. Rothenstein and dealt further with an additional important aspect, namely the impact of the solid state effects on the scattering kernel in the vicinity of resonances. In particular he showed (6, JEFF DOC 1402) that the prevailing assumptions on the major importance of solid state effects in the resolved resonance range, based on a unique attempt (14) to solve the word & Trammel (9) expression for solid state –Resonance dependent kernel, was not correct and therefore the above mentioned free gas model is dominating, as suggested by Rothenstein. Further it became evident that the absorption and the scattering part same resonances should be treated separately concerning the impact of the solid state effects. Those fundamental aspect for the treatment of the scattering kernels for heavy isotopes were the essential contribution for the understanding of the physical as well as numerical approaches (some of which are presented below) towards accurate solutions of the Boltzmann transport equation.

RESULTS
The development of fast computer allowed for practical implementation of the idea of W. Rothenstein in its both ways. First the analytical solution was tested (10) and based on Rothenstein (7) stochastic formalism the so called DBRC- “Doppler Broadened Rejection correction” was implemented into Monte Carlo codes (11). The good agreement between the two methods for several test cases confirmed the stochastic DBRC and further allowed for experimental testing. Within the current study the results obtained previously for U238 (12) were repeated for TH232. Figure 1 shows the scattering probability in the vicinity of the 69.2 resonance of Thorium for a backwards angle of 140.8 °. The experimental results of the RPI facility could be modeled to a very good extent by the DBRC model implemented in the MCNP code (3), where as the reference MCNP model under estimates the counts by about 40% at the peak. This result together with the similar results for the main S Resonances of U238 (12) confirmed the numerical assumptions of Rothenstein and the negligible importance of the solid state effect at the vicinity of the resonances as suggested by J. Rowlands. Moreover, the improved DBRC model in MC codes can serve as a reference for the investigation of deterministic solutions of the transport equation.

For deterministic calculation it is common to develop Legendre moments for the anisotropy of the scattering process, which are then embedded in the Flux type solver (Sn, Pn methods etc). Based on the development in (5) and (8) a new numerical program was developed which generates the single Legendre moments for the correct free gas based resonance dependent scattering kernel. More important the program enables a direct calculation of the angular dependent scattering kernel which is practically the sum of all higher Legendre moments namely P∞. Figure 2 shows the estimation of the zero, the zero + first moments compared to the direct calculation of the scattering kernel at 20.00 eV in the vicinity of the 20.87 resonance of U238 at cosine angle of 0.999 (forward angle). It is evident that the use of the first two moments as is done in several codes is insufficient and further moments are needed. However in practice, the use of more then three-four moments evolves usually an impractical numerical endeavor. Thereafter one might think of new ways to handle the scattering kernel which might lead to a
complete different deterministic solver to justify to a greater extent the benefit of the use of a transport equation.

Figure 1: Th232 comparison of experimental results to a DBRC improved MCNP model for neutron counts at scattering degree of 140.8° in the vicinity of the 69.2 eV resonance.

Figure 2: calculation of the scattering kernel at forward angle (μ=0.999), comparing the exact solution with the use of the two first moments for incident energy of 20eV in the vicinity of the 20.87 eV resonance.

CONCLUSIONS
The extensive studies of W. Rothenstein and J. Rowlands lead to a remarkable improvement of the complicated handling of the scattering kernel term in the last century. In particular a consistency was obtained between the integral and differential part of the Doppler broadened scattering cross section. Further the negligible effect of the solid state in the vicinity of the resonances for the secondary distribution were confirmed in dedicated experiments lead by Y. Danon at Rensselaer Polytechnic Institute (12). For solving the Boltzmann transport equation the idea of W. Rothenstein for an additional rejection was verified and applied successfully in various MC codes. For deterministic solver of the Boltzmann transport equation it is evident that the number of Legendre Moments needed is by far higher than can be practically used. In several cases one might need up to 50 moments to achieve comparable results to the new MC-DBRC formalism.(13). Due to the fact that the Legendre Moment treatment is usually strongly embedded within the governing Flux solver further extensive investigations should be inevitable towards the goal of accurate deterministic codes equivalent to the new “Rothenstein based” stochastic Monte Carlo codes.
REFERENCES
Cross Section Tuning as an Adjustment

J. J. Wagschal¹

¹Racah Institute of Physics, Hebrew University of Jerusalem, Edmond J. Safra Campus
Givat Ram, 91904 Jerusalem, Israel

J. J. Wagschal: wagschal@huji.ac.il

INTRODUCTION

The Generalized Linear Least-Squares Cross-Section Adjustment methodology, used in criticality safety problems can handle prior parameter correlations and prior response-parameter correlations (1). In this work the two parameters one response model of the generalized linear least squares parameter adjustment methodology (2) was used in order to analyze the procedure of improving cross sections agreement with integral experimental results by tuning. The tuning procedure can be achieved by modifying only one parameter, for instance the modification of $\nu$ in ENDF/B-VII.0 (3), in order to improve agreement of calculated responses with corresponding integral measurements results. The two parameters one response model formulas will be presented and reduced to a tuning simulation mode, analyzed and illustrated by a criticality numerical example.

TWO PARAMETERS ONE RESPONSE

For a single response the covariance matrix $C_d$ of the deviation $d = \tau(p) - r$ of the calculated response value from its measured response value reduces to a number and its value is:

$$C_d = S_1^2 \sigma_1^2 + S_2^2 \sigma_2^2 + 2S_1 S_2 \rho_{12} \sigma_1 \sigma_2 - 2 (S_1 \rho_{1r} \sigma_1 \sigma_r + S_2 \rho_{2r} \sigma_2 \sigma_r) + \sigma_r^2,$$

where the indices 1, 2, and r denote the two parameters and the response respectively, $S$ denotes the sensitivity of the response to the respective parameter, $\rho$ is the correlation factor and $\sigma$ is the standard deviation of the respective item. The items with an apostrophe are the respective posterior values. The variances, i.e. standard deviations squared, of the adjusted parameters and response are:

$$\sigma_1^2 = \frac{\sigma_1^2}{C_d} \left[ S_1^2 \sigma_1^2 (1 - \rho_{12}^2) + \sigma_1^2 (1 - \rho_{1r}^2) + 2S_1 S_2 \sigma_1 \sigma_2 (\rho_{12} \rho_{1r} - \rho_{2r}) \right],$$

$$\sigma_2^2 = \frac{\sigma_2^2}{C_d} \left[ S_1^2 \sigma_1^2 (1 - \rho_{12}^2) + \sigma_1^2 (1 - \rho_{1r}^2) + 2S_1 S_2 \sigma_1 \sigma_2 (\rho_{12} \rho_{2r} - \rho_{1r}) \right],$$

$$\sigma_r^2 = \frac{\sigma_r^2}{C_d} \left[ S_1^2 \sigma_1^2 (1 - \rho_{1r}^2) + S_2^2 \sigma_2^2 (1 - \rho_{2r}^2) + 2S_1 S_2 \sigma_1 \sigma_2 (\rho_{12} - \rho_{1r} \rho_{2r}) \right],$$

and the posterior correlation of the two adjusted parameters is

$$\rho_{12} = \frac{\sigma_1 \sigma_2}{C_d \sigma_1 \sigma_2} \times \left[ S_1 S_2 (\rho_{12}^2 - 1) \sigma_1 \sigma_2 + (\rho_{12} - \rho_{1r} \rho_{2r}) \sigma_1^2 + S_1 (\rho_{2r} - \rho_{1r} \rho_{12}) \sigma_1 \sigma_r + S_2 (\rho_{1r} - \rho_{2r} \rho_{12}) \sigma_2 \sigma_r \right].$$
It can easily be seen that generally the posterior parameters correlation does not vanish and that it is a function, inter alia, of the prior response uncertainty.

The ENDF/B-VII.0 evaluation process of U\textsuperscript{235} and Pu\textsuperscript{239} included, inter alia, modifying the $\nu$ values, the average number of secondary neutrons emitted from fission, in order to improve agreement with integral experimental results\textsuperscript{(3)}, while keeping the original evaluated values of other cross sections untouched. Since the other parameters (cross sections) have non negligible uncertainties in the ENDF/B files, this procedure is similar to an adjustment process with non vanishing sensitivities to only one parameter. Let us analyze the one response two parameters adjustment model with a vanishing sensitivity to the second parameter, i.e. $S_2=0$. The posterior parameters correlation reduces to

$$\rho_{12} = \frac{\sigma_1 \sigma_2}{C_d \sigma_1 \sigma_2} \times \left[ \left( \rho_{12} - \rho_{1r} \rho_{2r} \right) \sigma_1^2 + S_1 \left( \rho_{2r} - \rho_{1r} \rho_{12} \right) \sigma_1 \sigma_r \right],$$

and obviously vanishes when both the prior parameters correlation and the prior correlation of the second parameter and the response vanish, i.e. $\rho_{12}=0$ and $\rho_{2r}=0$. This result is not surprising since if the response is not sensitive to the second parameter and the second parameter is not correlated to any other parameter or response, the problem reduces to a one response one parameter problem. However, if the second parameter is correlated either to the first parameter or to the response or to both, the posterior correlation does not vanish. Moreover, in this case the posterior second parameter value, to which the response is not sensitive, and its variance are modified by the adjustment procedure,

$$p_2' = p_2 + \left( \rho_{2r} \sigma_2 \sigma_r - S_1 \rho_{12} \sigma_1 \sigma_r \right) \frac{d}{C_d},$$

$$\sigma_2^2 = \frac{\sigma_2^2}{C_d} \left[ S_1^2 \sigma_1^2 \left( 1 - \rho_{12}^2 \right) + \sigma_r^2 \left( 1 - \rho_{2r}^2 \right) + 2 S_1 \sigma_1 \sigma_r \left( \rho_{12} \rho_{2r} - \rho_{1r} \right) \right].$$

The numerator of $\sigma_2^2$ does not depend on $S_2$ and the denominator $C_d$ is now

$$C_d = S_1^2 \sigma_1^2 - 2 S_1 \rho_{1r} \sigma_1 \sigma_r + \sigma_r^2.$$

The posterior variance of the response reduces to

$$\sigma_r^2 = \frac{\sigma_r^2}{C_d} \left[ S_1^2 \sigma_1^2 \left( 1 - \rho_{1r}^2 \right) \right],$$

and the posterior variance of the first parameter reduces to

$$\sigma_1^2 = \frac{\sigma_1^2}{C_d} \left[ \sigma_r^2 \left( 1 - \rho_{1r}^2 \right) \right].$$

Obviously since the response is not sensitive to the second parameter, i.e. $S_2=0$, the uncertainty in the first parameter, the sole parameter to which the response is sensitive, propagates into the uncertainty in the response as $\sigma_r^2 = S_1^2 \sigma_1^2$. 

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When the response is not sensitive to the second parameter, i.e. \( S_2=0 \), the data combination generates a full correlation between the response and the first parameter, i.e. \( \rho_{\gamma\gamma'}=1 \). Similarly, the posterior correlation of the response and the second parameter, i.e. \( \rho_{\gamma'\gamma} \), is now equal to the posterior correlation of the parameters \( \rho_{\gamma\gamma'} \). The posterior uncertainties of the response and of the first parameter are smaller or equal to the respective prior uncertainties. Since the posterior uncertainties are not a function of the prior correlations of the second parameter they are the same as in the case of the one response one parameter case.\(^{(4)}\)

**NUMERICAL EXAMPLE**

Let us look at a bare homogeneous critical sphere of a fictitious multiplying material with \( c=1.55\pm1\% \) average number of secondary neutrons per collision and radius \( R=1.55\pm1\% \) mean free paths. The parameter \( c \), the average number of secondary neutrons per collision, characterizes the physical properties of a multiplying material in the one group approximation of the transport equation and is equal, for instance, to 1.47 for Pu\(^{239}\).\(^{(5)}\) The dimension of the sphere is determined by its mean free paths radius, \( R \). With these two parameters the calculated reactivity (collision multiplication factor) of the sphere, using the universal criticality curve\(^{(5)}\) is \( \gamma=0.99333 \) compared to the "measured" multiplication factor of the critical sphere, \( \gamma=1.0 \). The sensitivity (partial derivative) of the reactivity \( \gamma \) to either parameter is positive. The sensitivity of \( \gamma \) to the first parameter \( c \) is the non escape probability and its value for an \( R=1.55 \) m.f.p. sphere is \( S_1=0.64086 \). The sensitivity of \( \gamma \) to the second parameter, \( R \), is equal to \( c \) times the partial derivative of the non escape probability to \( R \); for \( R=1.55 \) m.f.p. this sensitivity is \( S_2=c*0.2012 \), however in our simulation of the tuning procedure \( S_2=0 \). Increasing the dimension of the sphere or increasing the multiplication per collision increases the reactivity. We apply the Generalized Linear Least-Squares Cross-Section Adjustment methodology to adjust the parameters \( c \) and \( R \) and the response \( \gamma \) assuming various prior \( c-R \) parameters correlations and various prior \( R-\gamma \) parameter-response correlations for several \( \gamma \) response uncertainties. In all cases we vary the other prior parameter-response correlation, \( (c-\gamma) \), over the whole permitted range from -1 to +1. Since the grand covariance matrix of the parameters and the response has to be positive definite the actual permitted range is narrower, and in each case we determine the permitted correlation range.

**Figure 1. Posterior relative response uncertainties**

In Fig. 1 the posterior relative response uncertainties are depicted as a function of the assumed first parameter response correlation for various prior relative response uncertainties. Since a 0.6 parameters correlation was used, the range of \( \rho_{\gamma\gamma'} \) does not span the whole (-1, +1) range.
The maximum is obtained when \( \frac{\sigma_r^2 (1 - \rho_{1r}^2)}{C_d} = 1 \), i.e. \( \rho_{1r} = \frac{S_r \sigma_r}{\sigma_r} \). It can be seen in Fig.1 that the maximum shifts towards the right. For a prior relative response uncertainty of 0.008 it is on the right boundary of the \( \rho_{1r} \) range and for 0.01 there is no maximum in the permitted \( \rho_{1r} \) range. The posterior parameter correlation \( \rho_{12} \) is presented in Fig. 2 as a function of the assumed first parameter response correlation for various prior relative response uncertainties and the corresponding reduced second parameter relative uncertainty is given in Fig.3.

![Figure 2. Posterior parameters correlation](image1)

![Figure 3. Posterior relative uncertainty in the second parameter](image2)

**SUMMARY AND CONCLUSIONS**

The two parameters one response adjustment model formulas were presented and reduced to a cross section tuning simulation mode, analyzed and illustrated by a criticality numerical example. The procedure introduces changes in the covariance data and thus should be avoided in standard general purpose nuclear data files.

**REFERENCES**


Demonstration of the Serpent Monte-Carlo Code Applicability to Few-Group Constants Generation for Existing and Advanced Reactor Concepts

E. Fridman\textsuperscript{1}, E. Shwageraus\textsuperscript{2}

\textsuperscript{1}Helmholtz-Zentrum Dresden-Rossendorf, POB 510119, 01314 Dresden, Germany
\textsuperscript{2}Ben-Gurion University of the Negev, POB 653, 84105 Beer-Sheva, Israel

Corresponding Author: e.fridman@hzdr.de

INTRODUCTION

Serpent is a continuous-energy Monte Carlo (MC) reactor physics code recently developed at VTT Technical Research Centre of Finland\textsuperscript{(1)}. Serpent can be used for 2D fuel lattice calculations as well as for 3D full core simulations. Due to its built-in decay and burnup routine Serpent can perform depletion and decay analysis to provide time-dependent isotopic compositions and spent fuel characteristics including radioactivity and decay heat. Serpent uses matrix exponential method to solve the Bateman decay and depletion equations while the solution of the matrix exponential relies on the Chebyshev Rational Approximation Method (CRAM)\textsuperscript{(2)}. Serpent runs significantly faster than other MC codes due to the two main reasons: 1) the use of the Woodcock delta-tracking\textsuperscript{(3)} in a combination with a typical surface-to-surface ray-tracing in a geometry routine, and 2) the use of the unionized energy grid\textsuperscript{(4)} for all point-wise reaction cross sections. The later, however, considerably increases the memory requirements and can be a bottleneck in simulations with a large number of involved nuclides.

Serpent is especially designed to generate homogenized constants for deterministic 3D core analysis. For any region of interest the code automatically calculates homogenized few-group cross sections, group-to-group scattering matrices, diffusion coefficients, assembly discontinuity factors, kinetics parameters, etc. More details can be found in Serpent User's Manual\textsuperscript{(5)}. Recently some new calculation methods related to the production of homogenized few-group constants were implemented in the Serpent code including homogenization in leakage-corrected criticality spectrum \textsuperscript{(6,7)}, group constant generation in reflectors and other non-fissile regions\textsuperscript{(7)}, and improved treatment of neutron-multiplying scattering reactions\textsuperscript{(7)}. The capability to generate homogenized few-group constants can be considered as one of the most attractive features of Serpent. Being a MC code, Serpent is capable of handling complex geometries without any major approximations and can be used for producing cross section data for virtually any fuel or reactor type. The demonstration of the Serpent capability to generate few-group cross sections for different reactor systems is the main topic of this paper.

DESCRIPTION OF THE REFERENCE CORES

In this study three core models were selected for demonstration of the Serpent modeling capabilities:

- A typical Westinghouse Pressurized Water Reactor (PWR) core. The core was loaded with 193 all fresh UO\textsubscript{2} 17×17 fuel assemblies at different enrichment levels. Some of the assemblies contain Wet Annular Burnable Absorber (WABA) rods. The reflector was modeled as a homogeneous mixture of water and structural materials. More detailed description of the utilized fuel types can be found in \textsuperscript{(7)}. 


A simplified version of the prismatic High Temperature Gas Cooled Reactor (HTGR) core developed by Idaho National Laboratory (INL) in the frame of Next Generation Nuclear Plant (NGNP)\(^8\) project. The core includes control rods channels symmetrically distributed in the fuel blocks. One of the unique features of prismatic HTGR cores is the use of tristructural isotropic (TRISO) fuel particles randomly dispersed in graphite fuel elements which are, in turn, placed in graphite blocks with cooling channels. The information on materials specifications, TRISO coated particle layers dimensions, and TRISO packing fractions can be found in\(^9\).

A Sodium Fast Reactor (SFR) core. The core design chosen for the analysis is a fast reactor with enriched uranium carbide (UC) fuel. It was developed as a part of the effort to design fast reactor with low enriched uranium based once-through fuel cycle, which is also economically competitive with LWRs\(^10\).

General core layouts and some basic operation parameters of three considered systems are given in Table 1.

**MODELLING APPROACH**

For each core the following analysis procedure was applied. Initially, few-group cross section sets for the core components were produced by the Serpent code. Then, few-group cross section sets generated by Serpent were used by multi-group nodal diffusion code DYN3D\(^11\) for the full-core analysis. Finally, the nodal results were compared with the full-core reference solution obtained using Serpent MC calculations. In all examples fixed thermal hydraulic conditions were considered and all core components were assumed to be at the constant temperature of 600K. The few-group constants were generated in 2, 12, and 24 energy group structures for PWR, HTGR, and SFR cores respectively in order to correctly account for particular neutron spectral effect of the each system.

Full-core MC calculations were performed with 5000 active cycles and 200,000 neutron histories per cycle to reduce statistical uncertainties. Fission source convergence was assured by skipping 1000 inactive cycles and monitoring the Shannon entropy behavior.

A large number of TRISO particles, their random distribution in the fuel pins, and so-called double-heterogeneous configuration of the HTGR fuel elements pose a modeling challenge for existing deterministic lattice codes. In Serpent, however, the TRISO particles can be described using incorporated explicit particle fuel model which obtains the particle positions from an auxiliary input file. In this work the distribution files with actual coordinates of the TRISO particles were generated using built-in Serpent command line routine. It should be noted that the explicit modeling of TRISO particles results only in a minor performance penalty.
Table 1. Radial layouts and basic operation parameters of PWR, HTGR, and SFR cores.

<table>
<thead>
<tr>
<th>Core type</th>
<th>PWR</th>
<th>HTGR</th>
<th>SFR</th>
</tr>
</thead>
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<tr>
<td>Core layout* (plotted to scale)</td>
<td>![Image]</td>
<td>![Image]</td>
<td>![Image]</td>
</tr>
<tr>
<td>Thermal power, MW</td>
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<td>600</td>
<td>2400</td>
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<tr>
<td>Active height, m</td>
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<td>7.90</td>
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<tr>
<td>Fuel columns, #</td>
<td>193</td>
<td>102</td>
<td>360</td>
</tr>
<tr>
<td>Reflector columns, #</td>
<td>64</td>
<td>163</td>
<td>150</td>
</tr>
<tr>
<td>Core lattice</td>
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<td>Hexagonal</td>
<td>Hexagonal</td>
</tr>
<tr>
<td>Core lattice pitch, cm</td>
<td>21.5</td>
<td>36.0</td>
<td>16.1</td>
</tr>
<tr>
<td>Fuel matrix</td>
<td>UO2</td>
<td>UO2</td>
<td>UC</td>
</tr>
<tr>
<td>U-235 enrich., w/o</td>
<td>2.1 to 3.1</td>
<td>8.2</td>
<td>10.8</td>
</tr>
</tbody>
</table>

Colors in use: red, green, yellow – fuel regions; light grey – reflector; dark grey – shield; blue – control rods

RESULTS

The results of calculations are summarized in Table 2. For all core models, the results of the Serpent full core MC simulations agree very well with those of the DYN3D calculations performed with the Serpent cross-section sets. The relative difference in k-eff is about 120, 133, and 26 pcm for PWR, HTGR, and SFR cores respectively. For all cases the average difference in the radial power distribution is lower than 1.5%. The maximum difference in the radial power distribution is lower than 3% for PWR and HTGR cores and is about 4.5% for SFR core. It should be noted that for PWR and SFR cores the maximum deviation in radial power distribution was observed for the peripheral fuel assemblies with low power density. More judicious reflector models can potentially improve the accuracy of the power prediction in these regions.

CONCLUSIONS

In this study we demonstrate by numerical examples the feasibility of the MC code Serpent for generating homogenized few-group constants for existing and advanced reactor designs. Extending the use of the continuous-energy MC method to lattice physics and group constant generation is a topic that is gaining more and more attention from the international reactor physics community since the method is not limited to any particular technology. This capability can be considered as particularly valuable for developing advanced reactor concepts for the future.
Table 1. Results summary

<table>
<thead>
<tr>
<th>Core type</th>
<th>PWR</th>
<th>HTGR</th>
<th>SFR</th>
</tr>
</thead>
<tbody>
<tr>
<td>k-eff, Serpent*</td>
<td>1.00027</td>
<td>1.39859</td>
<td>1.098028</td>
</tr>
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<td>k-eff, DYN3D</td>
<td>0.99910</td>
<td>1.40126</td>
<td>1.097719</td>
</tr>
<tr>
<td>Δρ DYN3D vs. Serpent, pcm</td>
<td>-120</td>
<td>133</td>
<td>-26</td>
</tr>
<tr>
<td>Max. diff. in radial power</td>
<td>2.9</td>
<td>0.99</td>
<td>4.6</td>
</tr>
<tr>
<td>Ave. diff. in radial power</td>
<td>0.8</td>
<td>0.42</td>
<td>1.4</td>
</tr>
</tbody>
</table>

*In Serpent calculations $\sigma_{k-eff}$ is about 0.00007

REFERENCES

Optimization of OTTO Fuel Management in Pebble-Bed Reactors Using Particle Swarm Algorithm

B. Tavron\textsuperscript{1}, E. Shwageraus\textsuperscript{2}, A. Galperin\textsuperscript{2}

\textsuperscript{1}Israel Electric Corporation, Haifa, Israel
\textsuperscript{2}Ben-Gurion University, Beer-Sheva, Israel

Corresponding Author: tavron@iec.co.il

INTRODUCTION

Pebble-Bed nuclear reactors feature highly flexible in-core fuel management capabilities due to on-line fueling and thermo-mechanical robust fuel design. Fuel pebbles with various fissile and fertile materials can be loaded into the reactor core at different rates. The fuel pebbles may be re-circulated in the core several times (4-6) until reaching their target burnup, or reach their target burnup in single pass through the core (OTTO- Once-Through-Then-Out fueling Scheme). Pebble-bed reactors have relatively efficient neutron economy since they operate with low excess reactivity and hence minimize the use of neutron poisons and control rods. Moreover, the fuel pebble robust design permits high burnup levels (up to 140000 MWD/THM). The flexibility of the fuel management operations allows enhancing fuel utilization. Traditionally fuel cycle design decisions were made using expert opinions and parametric studies \cite{1}. In this work, we have used the Particle Swarm Optimization (PSO) algorithm to optimize fuel utilization of pebble-bed reactors running OTTO fuel management. Optimization was carried out also for cores with Th232 as fertile material.

Preliminary calculations were performed for a large core \cite{1} with 2 radial fuel loading zones. Results of the optimal fuel utilization performed for cores with UO\textsubscript{2} fuel and cores with (Th-U)O\textsubscript{2} are summarized in Table 1 below. Future work will include optimization of cores fuelled with separate seed (U) and blanket (Th) fuel pebbles and with advanced modular core configuration, like the PBMR400.

DEFINITION OF THE OPTIMIZATION PROBLEM

The optimization objective is to find the fuel management scheme, which yields the highest fuel utilization. Natural uranium fuel utilization is calculated from the following expression:

\[ FU = \sum_{i} m_i \cdot FR_i \cdot FF_i \cdot \frac{Q}{P} \]

\[ FF_i = \frac{F}{P} = \frac{x_{p,i} - x_w}{x_f - x_w} \]

Where: \( m_i \) - Mass of HM per pebble in zone \( i \), kg,
\( FR_i \) - Pebbles feed rate in zone \( i \), #/day,
\( FF_i \) - Pebbles feed factor in zone \( i \),
\( F \) - Mass kg of natural uranium (feed material), kg,
\( P \) - Mass kg of enriched uranium (product material), kg,
\( Q \) – Reactor power, MW,
\( x_{p,i} \) - Weight fraction of U235 in the product
\( x_f \) - Weight fraction of U235 in the feed (natural U - 0.7%)
\( x_w \) - Weight fraction of U235 in the waste (tails, we use 0.2%)

The FU is natural uranium fuel utilization in units MWDth/kgNU. The FU is calculated for the equilibrium core in which the feed fuel enrichment is constant over time.

The VSOP code system is used to simulate the fuel cycle of the reactor \(^{(2)}\). The VSOP is a system of codes for the simulation of pebble-bed reactors with treatment of its special features, such as "double heterogeneity" and on-line continuous fuel loading. Unit-cell spectrum calculations in VSOP are performed by the THERMOS, ZUT and GAM codes for the thermal, the resonance and the epi-thermal energy spectrum regions, respectively. These codes apply various approximations to the transport equation to accommodate special features of pebble-type fuel with coated fuel kernels. Power and neutron flux distributions are then calculated by the 2D diffusion program CITATION. Burnup calculations and fuel shuffling operations are performed by the FEVER code. Thermo-hydraulic and fuel cycle cost calculation may also be performed by THERMIX and KPD codes.

The optimization parameters are HM loading and fissile enrichment per pebble, for each fuel loading zone. The pebble feed rate parameter is dependent on HM loading and enrichment. Pebble feed rate is calculated to maintain equilibrium core conditions using dedicated MATLAB script. In the equilibrium core, the neutron flux, power and material composition do not change considerably over time. The algorithm used in MATLAB script is described below:

Step 1: Initial core loaded with fresh fuel, operates until reaching criticality. Perform the fresh core burnup calculation until criticality is reached.

Step 2: Perform fuel management operations (core refueling with small batch fuel loading increments) and continue the core burnup simulating reactor life:
- Continuously load and discharge fuel pebbles from the core in small batches (in OTTO fuel management scheme load only fresh fuel). Iterate on reload cycle length (feed rate) to maintain core criticality.
- Repeat core burnup cycles until equilibrium condition is reached (no change in fuel feed rate to maintain criticality).

Step 3: Obtain fuel pebbles feed rate \( FR_i \) from equilibrium core and calculate Natural U utilization

PARTICLE SWARM OPTIMIZATION
PSO is an evolutionary optimization algorithm that is inspired by the social behavior of a flock of migrating birds trying to reach an unknown destination (or food)\(^{(3)}\). This stochastic algorithm mimics the flock of birds that communicate with one another as they fly. PSO has been found to have superior performance in several benchmarks \([4]\). In PSO, each solution is a "bird" in the flock and is referred to as "particle". Each bird in the flock looks in a specific direction and also identifies the bird with the best location in the flock. The bird is then speeds to a new location depending on own search (own experience) and a global search (flock experience). The process repeats until it reaches desired destination (convergence).
The process is initialized with generating a swarm of \( N \) random particles (solutions), which, in our case, we generated using the Latin Hypercube\(^4\) sampling technic to ensure an even sampling from the search space. Each particle \( i \) is characterized by a location vector \( x_i \) and a speed vector \( v_i \). The location quality of each particle \( p_i \), is calculated by the objective function, the natural uranium utilization - FU. \( G \) represents the global best location of the swarm. Advancing from time step \( k \) to \( k+1 \), each particle updates its location \( x_i(k+1) \) by the speed vector \( v_i(k+1) \). The updated speed vector depends on previous speed, previous best location and global best location weighted by the algorithm parameters \( w, c_1, c_2 \) and with additional random weighting \( \gamma_1 \) and \( \gamma_2 \). Hence the algorithm formulas are:

\[
\begin{align*}
    v_i(k+1) &= wv_i(k) + c_1\gamma_1(p_i - x_i(k)) + c_2\gamma_2(G - x_i(k)) \\
    x_i(k+1) &= x_i(k) + v_i(k+1)
\end{align*}
\]

Convergence to optimal value is presented in Figure 1 below showing swarm (population) maximum and mean values together with the global maximum of the swam.

![Figure 1. Convergence of the particle swarm algorithm.](image)

**RESULTS**

Several optimization calculations were performed for a large two zone core with OTTO fuel cycle. Reactor thermal power is 3000MW with core dimensions of 5.89m radius and 5.5m height with standard 6 cm fuel pebbles and TRISO type coated particles. Detailed core and fuel design parameters are from reference 1. Optimization was carried out for all uranium fuel (Cases 1 and 2) and for (Th-U)O2 fuel (Case 3 and 4). In cases 1 and 3, HM loading kept constant in the two loading zones, whereas in cases 2 and 4, different HM loading, one in each zone, was allowed. For fuel cost calculation, prices where obtained from the WISE internet site\(^5\) (as for 25/12/2011): Uranium purchase 52 $/lb U3O8; Conversion: 8.5 $/kg U; Enrichment: 140 $/kg SWU; Fabrication 8.5 $/kg U. Plant efficiency of 33% and escalation rate of 5% were used. The results are presented in Table 1.
Table 1. Optimization Results.

<table>
<thead>
<tr>
<th>Input</th>
<th>Case 1</th>
<th>Case 2</th>
<th>Case 3</th>
<th>Case 4</th>
</tr>
</thead>
<tbody>
<tr>
<td>Case 1</td>
<td>Zone1</td>
<td>Zone2</td>
<td>Zone1</td>
<td>Zone2</td>
</tr>
<tr>
<td>HM Loading (g/pebble)</td>
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<td>6.89</td>
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<td>Moderation Ratio</td>
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<td>547</td>
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<td>618</td>
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<td>HM Enrichment</td>
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<td>Ave. Enrichment</td>
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<td>Th fraction</td>
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<td>0</td>
<td>0</td>
<td>0</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Output</th>
<th>Case 1</th>
<th>Case 2</th>
<th>Case 3</th>
<th>Case 4</th>
</tr>
</thead>
<tbody>
<tr>
<td>Case 1</td>
<td>Zone1</td>
<td>Zone2</td>
<td>Zone1</td>
<td>Zone2</td>
</tr>
<tr>
<td>Load rate (Pebble/day)</td>
<td>4470</td>
<td>4449</td>
<td>3753</td>
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<tr>
<td>Power Peak Max/Ave</td>
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<td>3.61</td>
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<td>Discharge Burnup (MWD/kg)</td>
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<tr>
<td>Fuel residence time (days)</td>
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<td>726</td>
<td>942</td>
<td>906</td>
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<tr>
<td>Fuel Utilization (MWDth/kg NU)</td>
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<td>5.849</td>
<td>5.341</td>
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<td>SWU Utilization (MWDth/kg-SWU)</td>
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<td>5.531</td>
<td>4.749</td>
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<td>Plutonium production rate, (kg/GWe-Y)</td>
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<td>41</td>
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<td>22</td>
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<tr>
<td>Fuel Cycle Cost (mills/kWhe)</td>
<td>6.8</td>
<td>6.8</td>
<td>7.5</td>
<td>7.4</td>
</tr>
</tbody>
</table>

**CONCLUSIONS**

OTTO fuel management scheme parameters were optimized for achieving best natural uranium resource utilization, using particle swarm algorithm. It was found that for all-uranium fuel (Case 1) the optimal fuel loading is 6.86 g HM per pebble (moderation ratio of 549) and average enrichment of 8.83%. Mixing Thorium with the fuel (Case 3) does not improve natural uranium utilization since it requires higher enrichment. However, the use of Th does decrease significantly Plutonium production. Allowing different HM loadings in the two radial zones has practically no effect on the fuel utilization (Cases 2 and 4).

Separate Thorium fuel pebbles (the Seed and Blanket concept) is expected to have better fuel cycle performance. This concept will also be optimized using the particle swarm algorithm, together with advanced modular core designs.

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Proliferation Resistant Fuel for Pebble Bed Modular Reactors

Y. Ronen, M. Aboudy, D. Regev and E. Gilad

Department of Nuclear Engineering
Ben-Gurion University of the Negev
Beer-Sheva 84105, Israel

Corresponding Author: gilade@bgu.ac.il

INTRODUCTION

Proliferation of nuclear weapons produced with power reactors plutonium has always been a major problem of the nuclear energy industry. This includes the Pebble Bed Modular Reactor (PBMR), which is a specific design of a GenIV High-Temperature Reactor (HTR), mainly due to its online refueling feature, which may be misused for the production of weapons grade plutonium. A promising approach toward preventing the proliferation of power reactor plutonium is to denaturate the plutonium by increasing the ratio of $^{238}\text{Pu}$ to total Pu in the spent fuel\(^{\text{(1)}}\). The $^{238}\text{Pu}$ isotope is characterized by a high heat rate (approximately 567 W/kg) due to the alpha decay of the $^{238}\text{Pu}$ with half-life of 87.74 yr, in addition to its high spontaneous fission neutron emission, which is higher than that of $^{240}\text{Pu}$. Thus, the presence of $^{238}\text{Pu}$ in Pu considerably complicates the design and construction of nuclear weapons based on Pu, owing to these characteristics of $^{238}\text{Pu}$. Recent papers\(^{\text{(2,3)}}\) show that a Pu mixture is proliferation resistant given that the weight ratio of $^{238}\text{Pu}$ to Pu is larger than 6%. In this paper we have studied a feasible technique for ensuring that the $^{238}\text{Pu}$ to Pu ratio, in the Pu produced in PBMR, is larger than 6% during the entire fuel cycle.

Contamination of the spent fuel with $^{238}\text{Pu}$ may be achieved by doping the nuclear fuel with either $^{241}\text{Am}$ or $^{237}\text{Np}$\(^{\text{(4–13)}}\). The $^{238}\text{Pu}$ isotope is obtained from both $^{241}\text{Am}$ and $^{237}\text{Np}$ by a neutron-capture reaction and the subsequent decay of the reaction products\(^{\text{(13)}}\). The $^{237}\text{Np}$ isotope is by itself a potential weapons grade material. However, its large critical mass of $57\pm4$ kg\(^{\text{(14)}}\) and the difficulty of extracting it from irradiated fuel elements make it impractical for weapons purposes. On the other hand, the critical mass of $^{241}\text{Am}$ is smaller, i.e. 34 to 45 kg. However, with decay heat production of 114 W/kg, the critical mass becomes a heat source of 3.9 to 5.1 KW, which makes $^{241}\text{Am}$ unsuitable for weapons applications\(^{\text{(3)}}\). As a result, it is a non-proliferating material.

Another advantage of introducing $^{241}\text{Am}$ into the fuel is that during the (n,$\gamma$) reaction, another isotope is produces, i.e. $^{242m}\text{Am}$, which is highly fissile with extremely high spontaneous fission neutrons emission rate. Due to its high fission cross-section, the presence of $^{242m}\text{Am}$ reduces the residual poison reactivity penalty resulting from the introduction of $^{241}\text{Am}$ into the fuel. The branching ratio of the $^{241}\text{Am}(n,\gamma)$ reaction generally depends on the neutron energy spectrum in the reactor. However, in the thermal region this branching ratio is independent of the neutron energy. Since PBMR is a thermal reactor, this ratio is averaged and taken to be 10% and 90% to $^{242m}\text{Am}$ (metastable) and $^{242g}\text{Am}$ (ground state), respectively\(^{\text{(15)}}\).
In this paper we study the doping requirements of both $^{241}$Am and $^{237}$Np in order to guarantee a proliferation resistant Pu isotopic composition in the spent fuel during the entire fuel burnup cycle in PBMR. An important result of this study is that the introduction of either $^{241}$Am or $^{237}$Np into the nuclear fuel is accompanied by a penalty with respect to fuel burnup cycle duration.

**PBMR description**

Pebble Bed Modular Reactor (PBMR) has been developed with the intention to improve safety, economics and proliferation resistance. The PBMR has a vertical steel reactor pressure vessel which contains the core barrel and the annular pebble fuel core\(^{(16)}\). The fuel pebbles are 60 mm in diameter, weighting 210 g with graphite inner part encasing coated particles (TRISO) with total of 9 g uranium enriched to 9.6\% $^{235}$U, as presented in Ref.\(^{17}\). They cycle continuously through the reactor, where each pebble cycles through the core about 6-10 times with a single cycle duration of three months, until they are expended after about three years. At this point, the fuel average enrichment of a pebble is 4-5\% and its average burnup is 80 GWday/ton. Operational cycles are expected to last six years between shutdowns\(^{(16,17)}\).

The PBMR has many advantages, such as helium as a coolant (radioactively inert, not corrosive and can be fed directly into a turbine), online refueling, passive cooling mechanisms, and low power density core such that no melt-down scenario could physically occur, even in the case of LOCA\(^{(17,18)}\). In the case of a pebble coating breach, only a small amount of radioactive nuclides would be released, due to the fact that the pebble fuel is divided over 15000 particles, individually coated with ceramic materials. Another advantage, from a proliferation point of view, is that pebbles reprocessing is currently very difficult\(^{(17)}\).

**ANALYSIS METHODOLOGY**

In our analysis we consider a widely used PBMR unit cell approximation\(^{(19)}\). By averaging over the entire core, we set the unit cell packing factor to 0.61\(^{(20,21)}\), whereas the maximum packing factor for bcc lattice is 0.68. Throughout the calculations we used reflective boundary conditions for all six cubic faces, such that the calculated multiplication factor is the infinite multiplication factor $k_\infty$. Whenever beginning of cycle (BOC) was calculated, we used fresh fuel. The parameters of the specific PBMR design of 400 MWth are given in Refs. \(20-24\). The calculations were performed using the BGCore software package\(^{(25)}\), which provides a comprehensive computer simulation of nuclear reactor systems and their fuel cycles.

**RESULTS**

Two different scenarios are considered for doping the fuel. In the first one, Am vector was introduced into the nuclear fuel at BOC. The Am vector studied is composed of 85.6\% $^{241}$Am, 0.08\% $^{242}$mAm, and 14.32\% $^{243}$Am, which corresponds to the discharged fuel of a typical pressurized water reactor\(^{(27)}\). This given composition is used throughout the analyses below. The total amount of Amin in the reactor, corresponding to a concentration of 2100 ppm, is 9.65 kg (0.021 g/pebble). In the second scenario, $^{237}$Np was introduced into the fuel at BOC. The total amount of $^{237}$Np in the reactor, corresponding to a concentration of 3050 ppm, is 14.015 kg (0.031 g/pebble).

The dependence of the PBMR $k_\infty$ (of a unit cell) on the concentration of either Am vector or $^{237}$Np at BOC is illustrated in Fig. 1. The concentration is given in units of ppm – parts per million.
of the UO$_2$ weight. In both scenarios, $k_{\infty}$ exhibits a linear dependence on the concentration. The difference between the two scenarios is due to the high absorption cross section of $^{241}$Am with respect to $^{237}$Np.

Several burnup calculations were carried out in order to find the minimal concentration of the Am vector or $^{237}$Np needed in order to meet the requirement $^{238}$Pu/Pu $\geq$ 6% during the entire burnup cycle. According to Figs. 2 and 3 it was found that the minimal concentrations are 2100 ppm and 3050 ppm of Am vector and $^{237}$Np, respectively.

However, the qualitative behavior of the ratio $^{238}$Pu/Pu with respect to burnup is different for the two scenarios. In the $^{237}$Np doping, this ratio initially decreases on a very fast time scale (approximately 10 FPDs), after which it starts to monotonically increase (in a slower rate), retaining its initial value after approximately 60 FPDs. In the Am vector doping, on the other hand, this ratio exhibits a very sharp initial rise, reaching a maximum after approximately 40 FPDs, after which it monotonically decreases until it reaches an asymptotic value towards the end of the cycle.
From a proliferation point of view, it is possible to use even lower concentrations of $^{237}$Np. Decreasing the concentration of the $^{237}$Np below 3050 ppm results in a decrease of the $^{238}$Pu/Pu ratio below 6%, but only for a short period of time at BOC, as illustrated in Fig. 7b for $^{237}$Np doping at 3000 ppm. This ratio reaches a minimum of 5.94% after 10 FPDs and climbs back up to 6.01% after 30 FPDs. The total amount of plutonium produced during the first 10 and 30 FPDs is 1.98 kg and 5.76 kg, respectively.

The total amount of Pu during burnup is calculated for both scenarios and for a reference PBMR (without fuel doping). The total amount of Pu produced at EOC (after 80 GWd/ton) is rather similar for both $^{237}$Np and Am vector doping (79.1 kg and 78.0 kg, respectively). However, there is an increase of 10.7% ($^{237}$Np doping) and 9.1% (Am vector doping) of the total amount of Pu with respect to the reference PBMR. This excess Pu is mainly due to the increase in $^{238}$Pu amount for the doped fuel.

After determining the minimal concentrations Am vector or $^{237}$Np required for obtaining $^{238}$Pu/Pu $\geq$ 6%, we calculated the penalties in reactor performance due to the introduction of these isotopes. The discharge burnup of fuel from a PBMR reactor is taken to be 80GWd/ton (see Table I). We calculated the behavior of the effective multiplication factor $k_{eff}$ during burnup under the assumption that $k_{\infty}$-$k_{eff}$ equals 5080pcm due to leakage (28), as illustrated in Fig. 4, and found that at 80 GWd/ton, $k_{eff}$ = 0.95940 for the reference PBMR. The requirement that the doped fuel burnup cycle ends at the same $k_{eff}$ as that of the reference reactor results in the shortening of the cycle by 41.2 FPD and 19.9 FPD for the $^{237}$Np and Am vector doping, respectively. This corresponds to reduction of 4070MWd/ton and 1965 MWd/ton in fuel discharge burnup.

CONCLUSIONS
Denaturated plutonium with the ratio $^{238}$Pu/Pu $\geq$ 6% for the entire fuel burnup cycle can be achieved by adding either Am vector or $^{237}$Np to the nuclear fuel with minimum concentrations of 2100 ppm and 3050 ppm, respectively. However, $^{237}$Np is more favorable than $^{241}$Am for nuclear weapon production (27,29) and the penalty it imposes on the fuel burnup cycle length is larger than that of Am vector. Note that the total amount of plutonium produced is increases by 9.1-10.7% compared to a reference PBMR mainly due to an increase in $^{238}$Pu production. We conclude that
$^{241}$Am is a superior choice (to $^{237}$Np) in doping PBMR fuel for its lower minimal concentration, reduced penalty on the fuel burnup cycle and proliferation considerations. We found that the qualitative behavior of the ratio $^{238}$Pu/Pu with respect to burnup is different for the two scenarios. In the case of $^{237}$Np, the fast initial decrease and then a monotonic increase of this ratio enables us to use doping concentration smaller than 3050 ppm. For example, when using 3000 ppm of $^{237}$Np, the production of plutonium with isotopic ratio $^{238}$Pu/Pu less than 6% requires fuel discharge after approximately 10-20 days. The corresponding amount of plutonium produced is 1.98-3.9 kg, which essentially makes proliferation impractical.

The penalty for $^{237}$Np or Am vector fuel doping is the shortening of the fuel burnup cycle, compared to the reference PBMR, by 41.2 FPDs and 19.9 FPDs, respectively. This corresponds to reduction of 4070MWd/ton and 1965 MWd/ton in fuel discharge burnup.

REFERENCES
Innovative Pressure Tube Light Water Reactor with Variable Moderator Control

R. Rachamin¹,², A. Galperin², E. Fridman¹

¹Institute of Safety Research, Helmholtz-Zentrum Dresden-Rossendorf, Dresden, Germany
²Department of Nuclear Engineering, Ben-Gurion University of the Negev, Beer Sheva, Israel

Corresponding Author: r.rachamin@hzdr.de

INTRODUCTION

The features of a reactor based on multiple pressure tubes, rather than a single pressure vessel, provide the reactor with considerable flexibility for continuous design improvements and developments. This paper presents the development of innovative pressure tube light water reactor, which has the ability to advance the current pressure tubes reactors. The proposed design is aimed to simplify the pressure tubes reactors by:

- replacing heavy water by a light water as a coolant and moderator,
- adopting batch refueling instead of on-line refueling.

Furthermore, the design is based on proven technologies, existing fuel and structure materials. Therefore, it is reasonable to expect significant capital cost savings, short licensing and introduction period of the proposed concept into the power production grid.

The basic novelty of the proposed design is based on an idea of variable moderator content in the core and “breed & burn” mode of operation. Both concepts were extensively investigated and reported in the past (²), (³), (⁴). In order to evaluate a practical reactor design build on proven technology, several features of the advanced CANDU reactor (ACR-1000) were adopted. It should be stressed however, that while some of the ACR-1000 mechanical design features are adopted, the core design basics of the reactor proposed here are completely different. First, the inter fuel channels spacing, surrounded by the calandria tank, contains a low pressure gas instead of heavy water moderator. Second, the fuel channel design features an additional/external tube (designated as moderator tube) connected to a separate moderator management system. The proposed design is basically pressure tube light water reactor with variable moderator Control (PTVM LWR) (¹).

This paper presents a detailed description of the PTVM core design and demonstrates the reactivity control and the “breed & burn” mode of operation, which are implemented by the variation of the moderator in the core, from a neutronics point of view.
PROPOSED DESIGN DESCRIPTION

Fuel Channel Assembly Design
The basic unit of the PTVM core design is a fuel channel assembly similar to that of ACR-1000. The PTVM fuel channel assembly is shown schematically in Fig. 1.

![Fuel Channel Assembly Design](image)

Figure 1: Schematic view of PTVM fuel channel assembly design

It should be noted that the main modification of the ACR-1000 fuel channel assembly is an addition of external tube (designated as moderator tube) connected to a separate moderator management system. The moderator management system is introduced to vary the moderator tube content from “dry” (gas filled) to “flooded” by light water injection. The dynamic variation of the moderator is a unique and very important feature of the proposed design. The moderator variation allows an implementation of the “breed & burn” mode of operation. The “breed & burn” mode of operation is carry out by keeping the moderator tube empty (“dry” filled with gas) during the breed part of the fuel depletion and subsequently introducing the moderator by “flooding” the moderator tube for the “burn” part.

The fuel channel assembly is designed to contain 12 fuel bundles and produces average power of 6.129 MWt. The core configuration, which is composed of the fuel channel assembly as a basic unit, is very flexible. Therefore, the total core power may be varied by varying the total number of fuel channel assemblies, essentially from about 600 MWe to 1,400 MWe.

Reactor Core Design
The PTVM core design consists of 504 fuel channel assemblies and 96 water channels arranged in a 14 concentric rings. The 504 fuel channels are divided into four equal regions (each region with 126 fuel channel assemblies). It is assumed that a design of a separate moderator management system, replacing the standard ACR-1000 moderator system, is feasible and capable of changing the moderator content of a collection of fuel channel assemblies, e.g. all channels positioned in a single ring of channels. This assumption is made in order to allow radial symmetry of the moderator distribution in the core. The PTVM core design layout is shown schematically in Fig. 2.
The design described above was considered as the first design iteration. The basic core and thermal-hydraulic parameters adopted for this analysis were similar to those of the ACR-1000 design \(^5\).

**ANALYSIS OF THE PROPOSED CONCEPT**

The proposed PTVM core design was analyzed by simulating core behavior for a total of 7 reload cycles in order to achieve a representative equilibrium cycle. The analysis was performed using the BGCore computer code \(^6\). For the proposed analysis a new feature, which has the ability to verify automatically the moderator content during the burnup calculations, was added to the BGCore code.

For the initial core load at BOL, the \(^{235}\text{U}\) enrichments in the fuel channel assemblies were 3.0wt\%, 4.7wt\%, 3.5wt\% and 2.0wt\% for the inner region to the outer respectively. All following reload enrichments were 4.7%. The total heavy metal mass of the core is 105.04 MT. For a core heat output of 3089 MWt, specific power density of 29.38 MWt/MT HM is obtained.

At each fuel reload, core moderator content is minimal where all (or most) of the fuel channels are in a “dry” state and is varied by sequential (ring by ring) additions of moderator. This process is continued until all fuel channels are “flooded” and core criticality equal 1.00. At this point in time the U fuel is reloaded / reshuffled.

The core criticality of the 7 reload cycles is presented as a function of time in Fig. 3. The criticality curve behavior reflects the addition of moderator in a specific ring of channels at the moment where the criticality is reaching the value of 1.0. It should be noted that maximum excess reactivity of the core is about 2\% and does not exceed 3\%. The average discharged burnup was estimated and found to be about 51 GWd/T.
SUMMARY AND FUTURE WORK

An innovative pressure tube light water reactor has been proposed and analyzed. The main challenges of the proposed design were identified as:

- feasibility proof/preliminary design of a separate and dedicated moderator management system;
- detailed reactivity temperature coefficients for all possible operating conditions;
- design of core reactivity control system and managing the core power distribution.

This paper described and assessed the technical feasibility of the preliminary design from a neutronics point of view. The results and physical observations obtained from the current analysis, will serve for the future optimization and investigation of the Thorium-based fuel potential for the proposed concept.

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Test of a Solid LiF Target in Preparation for Stellar Neutron Production by a Liquid Lithium Target

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\textsuperscript{1}Soreq NRC, Yavne, Israel  
\textsuperscript{2}Racah Institute of Physics, Hebrew University, Jerusalem, Israel

Corresponding Author: gitai@phys.huji.ac.il

INTRODUCTION

Aiming for planned experiments in nuclear astrophysics using a liquid lithium target (LiLiT) for neutron production\textsuperscript{(1)} a set of measurements based on a solid LiF target – SoLiT were performed at SARAF in order to study the systematics and to measure the $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ Maxwellian Averaged Cross Section (MACS) for the SARAF beam to be used as a reference cross section. A dedicated setup was built for the characterization of the proton beam and for gold activation. Data was also collected on the neutron spectrum and background level at the target position and experimental hall.

THE EXPERIMENTAL SETUP

Based on the required beam parameters on target and accelerator constrains the design for SoLiT chamber components and the LiF cup was made. Beam dynamics determined the basic tune for the SARAF proton beam.

In order to produce the neutron cone with a semi-Maxwellian neutron spectrum the proton energy irradiating the LiF target is set ~30 keV above the $^{7}\text{Li}(p,n)^{7}\text{Be}$ reaction threshold. The transversal properties of the proton beam at target position should enable activation of a gold sample of reasonable dimensions. The total proton charge on target required for the activation is of the order of 10 mC and is easily collected for a ~5 µA beam which correspondingly requires to remove ~10 W of beam power.

![Figure 1. Left – SoLiT chamber positioned at the end of the beamline, gold secondary target is attached to the LiF cup, Right – 3D drawing of the LiF target.]

The experimental setup design which fulfill those requirements is based on the design of the solid Li/LiF target used at FZK for MACS measurements\textsuperscript{(2)} (fig. 1). LiF is evaporated on a 1 mm thickness Cu backing to produce a thick target (>1 µm) with a diameter of 8 mm.
The proton beam transversal dimensions were shaped by 3 collimators – a rectangular cooled collimator (4-jaw) at the end of the beamline, a 14 mm cooled collimator inside the SoLiT chamber and a 6 mm collimator very close to the LiF layer. The LiF target was floating and magnetically suppressed in order to enable integration of the proton current. The heat was removed from the target by peripheral water cooling which easily removed the heat while not interfering with the neutron spectrum.

A rotatable stage inside the SoLiT chamber enables choosing between 3 stations – cooled collimator, quartz disk for beam tuning and thin gold target (15 μg/cm² on a 1.3 mg/cm² Ti backing) for proton energy characterization by Rutherford back scattering. For the energy characterization a Si detector with a 2 mm diameter collimator was placed at an angle of 153°.

In order to have the ability to change the proton beam energy at target position while keeping the same tune for the linac and the beamline, the collimator was electrically insulated from the LiF cup. Scanning of the energy in the range of 10-20 keV may enable absolute measurement of radiation yield originated from a nuclei resonance or a reaction threshold. Characterization of the proton beam for relevant energies can be obtained when scanning the gamma yield of the $^{13}\text{C}(p,\gamma)^{14}\text{N}$ reaction or scanning the neutron yield of the $^7\text{Li}(p,n)^7\text{Be}$ reaction below threshold, when neutrons are produce by the high energy tail of the proton beam distribution, and above threshold.

**BEAM DYNAMICS**

Similar to the calculations performed for LiLiT$^{(3)}$, beam dynamics calculations were performed for SoLiT experiments using the TRACK code and included the lattice from the exit of the RFQ until the LiF target (see fig. 2). The cavities amplitudes and phases were tuned to bring the proton energy to 1912 keV and enabled changing the energy by a 1 keV step at the range 1720-1920 keV. Due to problems with cavities 3, 5 and 6 only 3 cavities were operated: cavity 1 was operated as a buncher with a phase of -95° and low amplitude, cavity 2 increased the energy by ~300 keV bringing the beam energy only ~100 keV below the required energy in order to operate cavity 4 at low amplitude for sensitive tuning of the final energy. Table 1 includes the cavities amplitudes and phases resulted from the beam dynamics optimization.

![Figure 2. 3D model of SARAF phase I accelerator and beamline of SoLiT experiments.](image)

Table 1. The cavities tune and related energies.
<table>
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<th>Cavity</th>
<th>Acceleration voltage (kV)</th>
<th>Phase (deg)</th>
<th>Energy at cavity exit (MeV)</th>
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<tr>
<td>5</td>
<td>0</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>6</td>
<td>0</td>
<td>-</td>
<td>-</td>
</tr>
</tbody>
</table>

**NEUTRON AND GAMMA DETECTORS**

During the measurements the neutrons were monitored at forward angle using a neutron dosimeter (Ludlum radiation dose ratemeter) and the spectrum was collected with a $^3$He spectrometer (Seforad), the Seforad detecting volume was shielded by a lead sheet in order to reduce the gamma background level. A shielded NaI detector with a collimator 6 cm in diameter recorded the gamma spectrum during the experiments (fig. 3).

**CONCLUSIONS**

An experimental setup based on a solid LiF target was built and operated at SARAF, the proton beam was characterized and Au cross section measurements were performed in order to get a reference for future experiments planned at SARAF with LiLiT as an intense stellar neutron source. Preliminary results will be presented.

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Towards the Production of Semi-Maxwellian Neutron Spectrum using SARAF

Y. Eisen1, G. Feinberg1,2, A. Shor1, M. Friedman2, A. Krasa3, D. Berkovits1, G. Giorginis3, T. Hirsh1, M. Paul2,*, A. Plompen3

1 Nuclear physics and engineering division, Soreq NRC, Yavne, Israel
2 Racah Institute of Physics, The Hebrew University, Jerusalem, Israel
3 EC-JRC-IRMM, Retieseweg 111, 2440 Geel, Belgium

Corresponding Author: yosef@soreq.gov.il

INTRODUCTION

The 7Li(p,n)7Be reaction plays an important role in the production of neutrons in the keV-energy range. Beer and Käppeler1 showed that for an incident proton energy of 1.912 MeV, the emitted neutrons from a thick Li target, integrated over all kinematically allowed angles, have an energy distribution similar to that of a flux of semi-Maxwellian neutrons at temperature $kT \sim 25$ keV. The neutrons produced by the near-threshold 7Li(p,n)7Be have been widely used to activate stable and more recently unstable nuclides involved in the $s$-process, providing a direct measure of the neutron-capture Maxwellian-averaged cross section (MACS) at $kT \sim 25$ keV. The experimental cross section of neutron capture on gold was carefully measured for the integral spectrum of neutrons from the thick-target near-threshold 7Li(p,n)7Be reaction by Ratynski and Käppeler3 using a Van de Graaff (VdG) accelerator. There currently exists a strong interest in expanding such activation measurements to unstable nuclides, either at branching points of the $s$-process and off the valley of stability or to cases experimentally difficult (i.e., small cross sections, short half-lives and decay radiation hard to detect). For these purposes, higher proton beam intensities (in the milliampere range) are then required. This may be achieved using high current RF LINAC accelerators, such as SARAF5, and high-power liquid lithium targets such as LiLiT6. However the properties of a high-intensity proton beam from a RF-accelerator are different from those of VdG accelerators having a broader energy definition. The purpose of this study is twofold: 1) To investigate whether it is possible to generate at SARAF a neutron spectrum of semi-Maxwellian shape with mean neutron energy around 25 keV. 2) To investigate whether a planar gold foil covering partially the angular spread of the outgoing neutrons is adequate to give a good reference for further MACS measurements. We describe in this paper an experiment using both narrow-energy and broad-energy proton beam, bombarding thick LiF target at energy 1.912 MeV, and the derivation of neutron spectra and gold activation cross sections for both cases. We also describe a comparison with our Monte Carlo code SimLiT7 that simulates the 7Li(p,n)7Be angular and integrated neutron distributions for narrow-energy and broad-energy proton beams.

EXPERIMENTAL CONSIDERATIONS AND RESULTS

The current study was performed in the low-scatter target hall of the VdG laboratory at IRMM, Geel8 Neutron energy spectrawere measured by time-of-flight (TOF) technique. To enable TOF measurements, the VdG was operated in a pulsed mode using a frequency of 625 kHz and a pulse width of 1.5-2 ns FWHM. The targets were made of LiF evaporated on a 1 mm
thick Cu disk. The LiF layer was thick enough to reduce the proton energy significantly below the $^7\text{Li}(p,n)$ threshold. In order to mimic the energy spread expected for a proton beam at SARAF accelerator, we placed a gold foil degrader directly upstream of the target. TRIM$^9$ calculations showed that 2097 keV protons traversing through a 2.06 µm gold foil have a mean energy of 1912 keV and an energy spread of width $\sigma=20$ keV, similar to the energy spread expected at the SARAF. Experimental setup is shown in Fig. 1.

Figure 1.—Schematic drawing of the experimental setup showing the goniometer on the floor and $^6\text{Li}$-glass detectors. Insert shows details of the target area.

A movable $^6\text{Li}$ detector of dimensions 2" diameter×1" thick was positioned at 51 cm from the LiF target at variable angles of 5°. The timing signal from each detector provided the TOF start signal in an allocated time-to-amplitude converter (TAC) channel. The total TOF resolution for the combined system, as measured for the gammas, was ~4 ns for most of the runs. Aplanar gold target of thickness 46.7 µm was utilized for each of the two experiments. The angle subtended by the gold target was 66°. The gold foil activity after irradiations, as well as the $^7\text{Be}$ activity of the LiF targets were measured using a HPGe detector.

Fig. 2a displays a raw TOF spectrum taken from the 2" diameter, 1" thick detector. The prompt gamma peak (off-scale) right and neutron spectra at longer flight times left of the gamma peak are observed. The insert highlighting the gamma peak shows a Gaussian gamma TOF distribution with FWHM ~ 4 ns. Fig. 2b shows a 2D scatter plot of TOF vs. pulse height for the $^6\text{Li}$-glass detector. The pulse height spectrum exhibits a neutron peak with a FWHM of ~15% with a rather long high energy tail. Fig. 3 shows the neutron energy spectra at at various angles for the narrow-energy proton beam and the broad-energy proton beam. Spectrum at each angle is integrated over $2\pi$ radians in azimuth. The spectra taken with the broad-energy proton beam exhibit pronounced spectral broadening compared to the spectra taken with the narrow-energy proton beam, as expected.

Fig. 3 shows the neutron energy distribution for several angles. Fig. 4 shows the integrated neutron spectrum for the narrow-energy and for the broad-energy proton beams. Although the individual spectra at each angle look markedly different for the narrow-energy and for the broad-energy beams, the integrated spectra are remarkably
Figure 2. 2a) TOF spectrum (left) showing gamma (off-scale) and neutrons. Insert shows gamma peak with FWHM~4 ns. 2b) TOF vs. pulse height (right).

similar, except in the high energy tail above 80 keV. Also shown in fig.4 is a fit to a Maxwellian flux distribution. For all the data from 8 to 80 keV, a best fit is obtained with kT=28 keV. The spectrum for the broad-energy beam shows considerably better agreement with the Maxwellian flux distribution at the higher neutron energies.

Figure 3. Measured neutron spectra at various angles for the $^7$Li(p,n)$^7$Be reaction obtained with (a) a narrow-energy and (b) a broad-energy proton beam.

The procedure adopted to determine the neutron capture cross sections by gold for both the narrow-energy and broad-energy proton beams can be divided into three stages: a) Measurement of the $^7$Be activity and extraction of the total number of emitted neutrons during activation runs. The total number of protons (obtained by current integration) was also useful to provide measurement of the n/p ratio. b) Measurement of the gold foil activity and correction for neutrons emitted outside the cone subtended by the planar gold foils. Correction with the aid of Monte Carlo calculations for scattering by the copper target backing is also taken into account. c) Correction with the aid of Monte Carlo calculations the experimental activity of the planar gold foils for the different target thickness sampled as a function of the neutron angle. The activation cross sections for the narrow-energy and the broad-energy proton beams were found to be 587±12 and 607±20 respectively and are in excellent agreement with Ratynski and Käppeler$^3$ for a hemispherical gold target foil. Fig 5 shows a comparison between the experimental data for the integral neutron spectra for the narrow-energy beam and the published spectrum of Ratynski and Kaeppeler$^3$. Also shown is a comparison to SimLiT$^7$ and PINO$^{10}$ calculations. The three spectra exhibit good agreement.
CONCLUSIONS

We have presented in this study a comparison of neutron spectra and gold foil activation studies with a narrow-energy proton beam ($\sigma \approx 1.5$ keV) as is typical of a VdG facility and with a broad-energy proton beam ($\sigma \approx 20$ keV) as expected at the SARAF. The angle-integrated neutron spectra were found to be nearly identical for the two measurements, except for the tail region where the spectrum obtained with the broad-energy beam showed better agreement with the tail of a semi-Maxwellian distribution. The extent to which this factor is significant in astrophysical measurements of the $\pi$-process reactions depends on the contribution of the higher energy neutrons at the tail to the MACS neutron capture cross section. These results are compared to simulations performed with the SimLiT simulation program and are in good agreement. Results for the narrow-energy beam are in good agreement to previous measurements.

It is shown that 1) It is possible to generate at SARAF a neutron spectrum of semi-Maxwellian shape with a mean neutron energy around 25 keV and 2) A planar gold foil covering $66^0$ and partially subtending the angular spread of the outgoing neutrons emerging out of the reaction $^7\text{Li}(p,n)^7\text{Be}$ at SARAF is adequate to give a good reference for further MACS measurements.

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Precision Measurements of the Beta Neutrino Correlation in an Electrostatic Ion Trap

S. Vaintraub$^{1,2}$, M. Hass$^1$, O. Heber$^1$, M. Rappaport$^1$, O. Aviv$^2$, I. Mardor$^2$

$^1$ Weizmann Institute of Science, Rehovot 76100, Israel
$^2$ Soreq NRC, Yavne 81800, Israel

Corresponding Author: sergeyva@weizmann.ac.il

INTRODUCTION
Trapped radioactive atoms present exciting opportunities for the study of fundamental interactions and symmetries. Traps are essential tools in this front-line research field by facilitating the precision determination of the beta decay parameters, and thus enabling probe the minute experimental signal that originates from possible tensor or scalar terms in the weak interaction. Such scalar or tensor terms affect, e.g., the angular correlation between a neutrino and an electron in the beta-decay process, thus probing new physics of “beyond-the-standard-model” nature\(^{(1)}\).

In particular, this article focuses on a novel use of an innovative ion trapping device, the Electrostatic Ion Beam Trap (EIBT)\(^{(2)}\). Such a trap has not been previously considered for Fundamental Interaction studies and exhibits potentially very significant advantages over other schemes. These advantages include improved injection efficiency of the radionuclide under study, an extended field-free region, ion-beam kinematics for better efficiency and ease-of-operation and the potential for a much larger solid angle for the electron and recoiling atom counters. The Electrostatic Ion Beam Trap (EIBT) setup will be thus designed to allow an efficient trapping of light ions such as $^6$He$^+$. In the following we briefly present the theory of beta decay formalism, with special emphasis on the $^6$He beta decay. The subsequent sections include a brief method description and simulations results. We conclude with a brief summary.

THEORY
The $\beta$-decay transition rate $W$ (inverse lifetime) in case of non-oriented nucleus is given by\(^{(3)}\)

$$dW \propto \xi \left( 1 + a_{\beta\nu} \frac{\vec{p}_e \cdot \vec{p}_\nu}{E_e E_\nu} + b \frac{m_e}{E_e} + ... \right) \propto \xi \left( 1 + \frac{\vec{p}_e}{E_e} a_{\beta\nu} \cos \theta_{\nu\nu} + ... \right)$$

Beta-neutrino correlation coefficient $a_{\beta\nu}, b$ and others are the beta decay coefficients (more coefficients also exists in case of polarized nuclei\(^{(3)}\)). In our recent study we concentrate on the beta-neutrino correlation coefficient - $a_{\beta\nu}$, which can be measured completely unbiased in case of $^6$He beta decay, as will be shown below. The following analysis can be very similarly attuned to others beta decay coefficients and radio-nuclei. The $^6$He nuclide has a half-life of 807 ms and is decaying into $^6$Li, electron and electron anti-neutrino $^6$He $\rightarrow$ $^6$Li $+ e^- + \bar{\nu}_e$

The most general way to write the beta decay Hamiltonian\(^{(3)}\) is:

$$H_\beta^{^6\text{He}} = \sum_{i=S,P,Y,A,T} \langle ^6\text{Li} | \hat{O}_i ( ^6\text{He} ) | \epsilon \hat{O}_i (C_i + C_i' \gamma) \nu_e \rangle + h.c.$$
where $S, P, V, A$ and $T$ stands for Scalar, Pseudo-scalar, Vector, Axial-vector and Tensor respectively. Then, the beta-neutrino correlation coefficient then can be expressed by the Gamow-Teller (GT) and Fermi (F) matrix elements along with eight coupling constants $C, C'$ \((3)\):

$$a_{\xi} = |F|^2 \left[ |C_{\gamma}|^2 - |C_S|^2 + |C_{\tau}|^2 - |C_{A}|^2 \right] + \frac{|GT|^2}{3} \left[ |C_{\tau}|^2 - |C_{A}|^2 + |C_{\gamma}|^2 - |C_{A}|^2 \right]$$

$$\xi = |F|^2 \left[ |C_{\gamma}|^2 + |C_S|^2 + |C_{\tau}|^2 + |C_{A}|^2 \right] + |GT|^2 \left[ |C_{\tau}|^2 + |C_{A}|^2 + |C_{\gamma}|^2 \right]$$

The $^6\text{He}$ beta decay is a $J^{\pi} = 0^+ \to J^{\pi} = 1^+$ pure Gamow-Teller decay. In that case the correlation coefficient can be expressed only by Axial and Tensor coupling constants in the Hamiltonian. In the Standard Model (SM) only the V-A interactions can occur, therefore only the axial coupling constants remain and, consequently, the coefficient obtains simply (up to second-order terms), the value $a_{\beta \nu} = -1/3$:

$$dW_{\text{SM}}(\theta_{ev}) \propto \left( 1 - \frac{P_e}{3E_e} \cos \theta_{ev} \right)$$

where the squared GT matrix element becomes a multiplicative constant, not affecting the form of the correlation coefficient. Moreover, in the general case, beta decay may be neither a pure Gamow-Teller nor a pure Fermi decay. As a result any beta decay coefficient in the transition rate equation would be matrix elements depended, thus nuclear structure theory depended. However, even in case of pure GT-decay of $^6\text{He}$ the second order corrections become important when precision measurements are below $1\%$ and should be taken into account in the simulations.

**METHOD**

The main purpose of the experiment is a high precision measurement of the beta decay coefficients. In the following we present a novel approach – that of using an Electrostatic Ion Beam Trap (EIBT). Following the pioneering work at the Weizmann Institute of Science in 1996\((2)\) for storing ions at typical energy of few keV. The principle of operation of an EIBT\((5)\) is based on the analogy to classical optical resonator: it is possible to store a beam of photons between two spherical mirrors separated by distance $L$ if the focal length $f$ fulfills the stability condition, namely: $L/4 < f < \infty$. A simple scheme of the EIBT setup is shown in Figure 1. The basic EIBT consist of a set of eight electrodes both acting as an electrostatic mirror by producing a retarding field which reflects the beam along its path and focuses it on the lateral direction. Thus the ions bounce back and forth between the two mirrors. The typical distance between the electrostatic mirrors is 500 mm.
The trap assembly consists of two mirrors which reflect the beam and focus it on the lateral axis (see Fig. 1). Ions with kinetic energy of ~5 keV are injected through the grounded entrance mirror, when the ions fill the trap; potentials on the entrance mirror are quickly raised (<50ns) so that the ions oscillate back and forth between the mirrors. For $^6$He$^+$, the revolution time is ~2 microseconds. A pickup electrode is used to continuously monitor the bunch location. Therefore, position of the decay is known. The products of the decay (recoil Li nuclei and electron) are detected by a set of dedicated detectors. Two annular position sensitive Multi-Channel-Plate (MCP) detectors are situated along the main trap axis with the purpose of counting the recoil Li. These MCPs have central holes which allow the stable trajectories of the stored ions (such technology was demonstrated before$^{(6)}$). The electrons are counted by two detectors located above and below the optical axis. The position and energy of the electron are detected in a plastic scintillator coupled to multiple photomultipliers with the position being determined by a "center of gravity" algorithm$^{(7)}$.

![Figure 1. Schematics. The radioactive ion $^6$He moves with $E_k=4.2$ keV between the reflecting electrodes. The $\beta$ electrons are detected in position sensitive counters while the recoiling ions, due to kinematic focusing, are detected with very high efficiency in either one (determined by the instantaneous direction) of the annular, position-sensitive Multi-Channel-Plate (MCP) counters.](image1)

![Figure 2. Results of GEANT4 simulations of a realistic geometry for the $\beta$ decay of $^6$He in the EBIT. The slight distortion of the linear cosine dependence around $\cos \theta_{e\beta} = -1$ due to losses out of the MCP and through the annular hole, respectively (see text). The simulated and analyzed $10^6$ events yield an correlation coefficient of $\theta_{e\beta} = -0.3355(18)$.](image2)
SUMULATION RESULTS
Extensive simulations of the relativistic kinematics of the $^{6}$He decay in the EBIT, including all realistic parameters of the experimental set-up, have been carried out using the GEANT4$^{(8)}$ Monte-Carlo code. Simulations are crucial in order to fully and quantitatively recreate the experimental conditions and demonstrate the concept’s validity. Such simulations provide a firm framework for anticipating the experimental results and their errors, both statistical and systematic. The construction of the dedicated set-up of an EIBT for beta-decay studies should naturally follow such simulations and take advantage of the gained insight. In the Figure 2 we present briefly the most pertinent simulation results.

SUMMARY
This research direction is also closely related in turn to the intensive R&D efforts that are currently under way in our laboratory to facilitate the production and extraction of such radioactive nuclei ($^{6}$He and $^{8}$Li, for example) in record yields, using neutron-induced reactions$^{(9)}$. We are planning to use an intense d+t commercial neutron generator that provides ~14 MeV neutrons at a fluxes reaching $10^7$-$10^{10}$ n/s. The delivery of such a generator at WI is planned by the next month. At a later stage, we envisage taking advantage of uniquely high yields of radioactive nuclei that will become available at the new and modern SARAF accelerator at Soreq. A preparatory setup designed to explore the experimental conditions en-route to the full experimental system has now been constructed. This includes a beam-line and electrostatic trap where stable $^{4}$He$^+$ and $^{4}$He$^{++}$ are being trapped and investigated to optimize experimental conditions such a trapping time, bunch size and injecting efficiency.

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Production of Ge-68 at SARAF

Shlomi Halfon, Ido Silverman, Efi Lavie, Adi Dahan
Soreq, Road 4111, Yavne 81000, Israel

Corresponding Author: ido@soreq.gov.il

ABSTRACT
Germanium-68 (T1/2 = 270.8 d) serves as a generator parent for short-lived 68Ga (beta+, 88%, 67.6 min), a positron emitter utilized in PET. The high potential of 68Ge/68Ga generator for PET application was recognized in the last few years.

68Ge can be produced in accelerator through the reactions 69Ga(p,2n) or with natural gallium natGa(p,*)68Ge (natural gallium consist of 60 % 69Ga). The 68Ge yield from gallium-69 thick target was calculated from published cross section data. According to this calculation the proton energy for efficient 68Ge production should be around 30 MeV, with 66 mCi/(mA*h). Generators marketed today have maximum activity of ~2000 MBq (~55 mCi).

SARAF (Soreq Applied Research Accelerator Facility) is a high intensity proton/deuteron super-conducting linear accelerator currently in construction at Soreq NRC. The target consists usually of gallium encapsulated in niobium. The low melting point of gallium poses a challenge for target design. SARAF target development group works for several years on development of target for high power accelerators. One of the target designs has been modified to enable 68Ge production by the high current proton beam of the SARAF. This is almost one order of magnitude higher than current state of the art. The design has undergone a comprehensive test program with a high-power electron-gun and is expected to be tested with the low energy high current Phase-I SARAF accelerator later this year.

The manuscript will describe the target and preliminary results of tests done with the high-power electron gun.
LISCAN-Liquid Scintillator Capillary Fast- Neutron Imaging Spectrometer

D. Vartsky\textsuperscript{1}, I. Mor\textsuperscript{1,2}, M. Brandis\textsuperscript{1}, M.B. Goldberg\textsuperscript{1,2}, D. Bar\textsuperscript{1}, I. Mardor\textsuperscript{1}, V. Dangendorf\textsuperscript{2}, B. Bromberger\textsuperscript{2}, K. Tittlemeyer\textsuperscript{2} and M. Weierganz\textsuperscript{2}

\textsuperscript{1}Soreq Nuclear Research Center (SOREQ NRC),Yavne 81800, Israel
\textsuperscript{2}Physikalisch-Technische Bundesanstalt (PTB),Braunschweig 38116, Germany

Corresponding Author: david@soreq.gov.il

INTRODUCTION

High resolution imaging of fast neutrons combined with energy spectroscopy is required in a variety of applications, ranging from Fast Neutron Radiography and Tomography, nuclear material monitoring, to solar and atmospheric physics.

Over the last decade several groups have been developing fast detectors based on particle tracking using scintillating plastic fibers or capillary tubes filled with liquid scintillator, mainly in order to determine the direction of the incident neutron and for high resolution imaging \cite{1-5}.

Our principal interest is Fast Neutron Resonance Radiography (FNRR) \cite{8}, which requires detectors with the following properties: high neutron efficiency (> 10\%), large area or long linear arrays for high resolution radiography of voluminous objects, sub-mm position resolution capabilities, neutron spectroscopy, insensitivity to gamma-rays. The imaging neutron detectors we have developed for FNRR to-date \cite{9,10} perform fast neutron spectroscopy by the method of measuring neutron time-of flight (TOF). TOF methods require operating with a nanosecond-pulsed neutron source, such as a particle accelerator.

In this paper we describe development of a micro-capillary bundle detector filled with liquid scintillator, that will permit high resolution imaging and medium-quality energy spectroscopy of non-pulsed fast neutron sources, such as CW accelerators, isotopic neutron sources or reactor beams.

THE CONCEPT OF LISCAN DETECTOR

Fig. 1 shows schematically the concept of the capillary bundle detector. The detector is based on a capillary array filled with high refractive-index liquid scintillator. The principal fast neutron reactions (in the energy range 0.8 – 14 MeV) within the liquid scintillator are elastic scattering on hydrogen and, to a somewhat lesser extent, carbon, as both elements have comparable atomic density in such substances. The proton moves within the bundle, creating scintillation light inside the capillaries it traverses. A fraction of this light will travel to the end of these capillaries via total internal reflection and be registered in the optical readout system, thereby creating a projection of the proton track.

Assuming the incident neutron flight direction relative to the capillary bundle axis is known (this is true for radiographic systems) and there are no multiple neutron interactions in the bundle, one can calculate the incident neutron energy using two parameters that characterize...
the proton track: the total amount of light in the track and the track projection length. Detailed capillary detector simulations were performed in order to find relations necessary for reconstruction of the incident neutron energy, namely: recoil-proton energy and recoil-proton track-length vs. total created light.

Fig. 2 shows the reconstructed spectra of neutrons for incident energies 4, 10, 13 and 20 MeV. The FWHM resolution is 1.7-2 MeV. Noticeably, the peaks are quite broad at the base of the distribution.

The main reason for this effect is the inaccuracy in counting the number of capillaries in the track and converting them into projection length (in μm), especially for short projections. In order to reduce this error it is important for the diameter of the capillaries and the thickness of the walls to be as small as possible. The above spectra show all tracks above a projection length of 3 capillaries. The distributions can be improved if we reject the short projections. However, rejecting the short tracks reduces the counting efficiency.

**EXPERIMENTAL SETUP AND RESULTS**

The capillary detector, seen in Fig. 3, consists of the following components: capillary array filled with liquid scintillator, tandem reversed stacked lens configuration, time-gated image-intensifier, cooled CCD camera.

In order to achieve high energy-and-spatial resolution, the capillary cross-sectional dimensions should be as small as possible. Fig. 4 shows a
small capillary array made from silica glass (1.4 cm in width, 3 cm in length, refractive index \( n = 1.4632 \)) made by the XOS company [12]. The dimensions of a single capillary are: inner average diameter 11 µm, wall thickness 1 µm, the usable free area (excluding the triangular patterns in Fig. 10) is about 72%. The array was filled by capillary action with EJ309 scintillator manufactured by ELJEN [11], presenting the following characteristics: refractive index \( n = 1.57 \), light yield = 11,500 photons/MeVee, density: 0.965 gr/cm³, H/C ratio = 1.25, light output: 75% of anthracene.

The detector was tested in a neutron beam using the PTB cyclotron. The capillary array was irradiated from the side, i.e., perpendicular to its central axis, by neutrons created in the \(^{10}\)Be(d,n) reaction using a 12 Mev deuteron beam that impinges on a thick Be target. Neutron energy from the broad energy spectrum was selected using the Time-of-Flight (TOF) technique. This was accomplished by time-gating the image-intensifier such that it acts as a very fast shutter (on a ns time-scale), collecting light after a pre-determined time relative to the creation time of the neutron pulse. The selected neutron energies were 8.6 and 15.5 MeV. Protons generated by neutrons in the scintillating liquid exhibit bright continuous tracks with a Bragg peak at their end. Gamma-ray-induced electrons generate small, faint blobs of light, that appear as a multitude of specks. The separation of electron from proton events can be performed automatically by using light and track length thresholding, pixel connectivity analysis and the existence of a Bragg peak.

Fig. 5 shows 3 examples of proton tracks obtained for 15.5 MeV neutrons. In all of them the Bragg peak is clearly discernible.

At this stage of the project the identification of the proton tracks has been performed by visual inspection of the CCD images. In this experiment about 60 proton tracks, with projection length above 7 capillaries, have been collected for the two neutron energies and used in the energy reconstruction.
Fig. 6 show the reconstructed energy spectrum. Clearly the event statistics is rather poor; nevertheless there is a strong evidence for the existence of two neutron groups at about 8.6 MeV and at 15.5 MeV.

![Fig. 6 Reconstructed spectrum of 8,6 and 15.5 MeV neutrons](image)

**CONCLUSIONS**

A novel imaging neutron detector consisting of micro-capillaries filled with a high refractive index liquid scintillator has been developed. The important properties of the detector are high position resolution (tens of microns), good rejection of gamma-ray events and the ability to construct efficient, large-area detectors.

The energy reconstruction method presented here, based on the determination of light yield and the projection length of each proton track, results in 10-15% energy resolution in the range 4-20 MeV. This energy resolution is inferior by a factor of 2-4 to that obtainable with TOF spectrometers or with non-imaging organic spectrometers that use unfolding algorithms. However, its’ energy resolution is much better than that of the Bonner multi-sphere method. It is planned to investigate the standard unfolding reconstruction procedures with the capillary detector to achieve better energy resolution, and yet maintain the high position resolution.

**REFERENCES**

Detector for Simultaneous Detection of Explosives and Special-Nuclear-Materials

M. Brandis¹, D. Vartsky¹, V. Dangendorf², B. Bromberger², D. Bar¹, M.B. Goldberg¹, K. Tittelmeier², E. Friedman¹, I. Mardor¹ and M. Weierganz²

¹Nuclear Physics Division, Soreq Nuclear Research Center, Yavne 81800, Israel
²Physikalisch-Technische Bundesanstalt (PTB), Braunschweig 38116, Germany
³Racah Institute of Physics, The Hebrew University of Jerusalem, Israel

Corresponding Author: michal_brandis@yahoo.com

INTRODUCTION

Israel has been a prime target for international terrorism for many years. Given the recent availability of nuclear technology in countries hostile to Israel, the threat of nuclear material being smuggled-in, calls for vigorous and effective countermeasures. The recent deployment of a radioactive bomb detection system(¹) at Haifa port is an acknowledgment of this threat, but can be merely an initial response to it.

The development Time-Resolved Event-Counting Radiation (TRECOR) detector is part of the Automatic Contraband-(Explosives and Special Nuclear Materials)-in Cargo Interrogation System (ACCIS) project. The ACCIS project, proposed by Soreq NRC and PTB, Braunschweig, is financed by a bi-national German/Israel Fund for Civil Security research project.

The ACCIS concept combines high-spatial resolution Fast-Neutron Resonance Radiography (FNRR) and Dual Discrete-Energy γ-Radiography (DDER) for automatic detection of standard and improvised explosives as well as special nuclear materials (SNM) concealed within inspected items ranging in size from small parcels and passenger baggage to aviation cargo containers and passenger vehicles (²).

Unlike conventional X-ray radiographic systems, FNRR and DDER do not rely on the skills of a human operator to identify the shape of the threat. Instead, they rely on automatic identification of the concealed contraband by its elemental composition. An operational system combining these two state-of-the-art inspection techniques will permit reliable and automatic detection of small quantities of explosives and SNM, respectively.

Dual Discrete Energy γ-Radiography

DDER (⁶) makes use of the discrete gamma-ray lines, 15.11 and 4.43 MeV, copiously produced in the ¹¹B(d,n) reaction with 3–6 MeV deuterons. In the conventional Dual Energy Bremsstrahlung (DCER) method the two gamma spectra are predominantly overlapping. In contrast, the very clean and well separated gamma lines in the ¹¹B(d,n) spectrum allow for significantly higher contrast sensitivity and will provide for much lower limits on the detectable amount of SNM material. Taking several radiographic views permits effective discrimination against innocuous high-Z materials, such as Hg, Pb and Bi thereby reducing false alarms to very low levels.

Fast Neutron Resonance Radiography

FNRR is an imaging method that exploits the characteristic cross-section structure (resonances) of different isotopes in the energy-range E₀=1-10 MeV. The inspected object is irradiated with neutrons in the above energy range. The transmission spectrum carries information about the
elemental composition of the object. A pre-requisite for FNRR is the precise knowledge of the neutron energy which is achieved by TOF spectroscopy \(^{3,4}\).

**Combined FNRR/DDER - FNDDER**

DDER utilizes the 15.11 and 4.43 MeV gamma-rays produced in the \(^{11}\text{B}(d,n)\) reaction with 3–8 MeV deuterons to counter the threat of improvised nuclear devices. The same \(^{11}\text{B}(d,n)\) reaction produces neutron spectrum with energy between 0-17 MeV that can be used for FNRR, to detect standard and improvised explosives in solid or liquid form.

The gamma-rays can be differentiated from the neutrons by TOF, allowing for simultaneous FNRR and DDER \(^{2}\). Using the same reaction for both detection techniques will permit employing the same accelerator, target and detectors, thus reducing costs and facility size. The combined FNRR/DDER detection system is denoted FNDDER.

The schematics of such an inspection system are presented in Fig. 1. It is configured around a compact ns-pulsed deuteron accelerator. Fast neutrons and \(\gamma\)-rays produced in the \(^{11}\text{B}(d,n)^{12}\text{C}\) reaction are collimated into fan-beams and detected by corresponding detector arrays. To generate multi-view data in a single pass and to shorten screening times of an object, several columnar detector arrays will be employed.

![Figure 1. Schematic of Combined FNRR/DDER System, showing how few-view inspection can be achieved in a single pass by means of several detector arrays.](image)

**TRECOR description**

The TRECOR detector is based on a time resolved event-by-event counting optical technique. TOF is required to distinguish the neutrons from the gamma-rays emitted in the same nuclear reaction. The time resolution achievable in this method is in the order of \(<1\) ns (provided the deuteron beam can be pulsed in sufficiently short bursts).

The TRECOR concept, as illustrated schematically in Fig. 2, starts with incoming particles being converted into light by the scintillator screen. The light is then bent 45° and focused by the lens onto the Event-Counting Image-Intensifier (ECII) that registers the TOF and position for each event \(^{6,7}\).
The TRECOR concept makes for a flexible neutron/gamma detector. When a plastic scintillator fiber screen is in place, TRECOR is a neutron detector. By replacing it with a high-Z scintillator, TRECOR becomes an efficient gamma detector due to the gamma/high-Z atoms electromagnetic interactions, specifically the pair production which is the dominating process in the 4-15 MeV gamma-rays energy range. This property makes TRECOR ideal for the dual-purpose detection system discussed earlier.

In the measurements described here, TRECOR was used in its gamma-detector form. A LYSO block, 20×50×100 mm³, was the high-Z scintillator. In order to readout the pulse height out of the LYSO a photomultiplier tube (PMT) was attached to the side of the scintillator by a perspex light-guide.

RESULTS

In previous reports (6), results of TRECOR as a neutron detector have been shown. In the present report, the latest results of TRECOR as a gamma-ray detector are to be presented.

Fig. 3 shows a TRECOR pulse height spectrum acquired with no attenuator. The LYSO scintillator yields 23,000 photons/MeV, allowing easy separation between the 15.1 and the 4.4 MeV gamma-rays. The difference between the two energies is clear in the spectra, as seen in Fig. 3.

The experimental transmission is compared to the expected transmission, calculated from cross-sections(8) in Tab. 1 for gamma-rays at both 4.43 and 15.1 MeV energies. The experimental transmission agrees with the expected value within 10%, which is consistent the experimental accuracy.
### Table 1. Transmission compared between experiment and expected from cross-sections (8)

<table>
<thead>
<tr>
<th>X (cm)</th>
<th></th>
<th>4.43 MeV</th>
<th>15.1 MeV</th>
<th></th>
<th>4.43 MeV</th>
<th>15.1 MeV</th>
<th></th>
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<td>C</td>
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<td>0.64</td>
<td>0.79</td>
<td>0.60</td>
<td>0.74</td>
<td>0.94</td>
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<tr>
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<td>1</td>
<td>0.65</td>
<td>0.53</td>
<td>0.62</td>
<td>0.53</td>
<td>0.96</td>
<td>0.99</td>
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<tr>
<td>Pb</td>
<td>2</td>
<td>0.42</td>
<td>0.31</td>
<td>0.39</td>
<td>0.28</td>
<td>0.92</td>
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<tr>
<td>Fe</td>
<td>1</td>
<td>0.81</td>
<td>0.83</td>
<td>0.78</td>
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<tr>
<td>Fe</td>
<td>2</td>
<td>0.61</td>
<td>0.64</td>
<td>0.60</td>
<td>0.61</td>
<td>0.98</td>
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<tr>
<td>Fe</td>
<td>5</td>
<td>0.30</td>
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<td>0.30</td>
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</tr>
</tbody>
</table>

### CONCLUSIONS

Results are presented from the latest experiment with the TRECOR detector in a gamma-ray field. TRECOR’s concept was presented as well as its’ intended application in explosives and SNM detection, as a multipurpose detector for neutrons and gamma rays.

### REFERENCES

Interleaved Carbon Minibeams: An experimental Method of Radiosurgery Developed at Brookhaven National Laboratory

F.A. Dilmanian¹,4,5, A. Rusek², G.R. Fois¹,6, J. Olschowka⁷, N.R. Desnoyers¹, J.Y. Park¹, I. Dioszegi³, B. Dane¹, R. Wang¹, D. Tomasi¹,8, H. Lee¹, S.D. Hurley⁷, P.K. Coyle⁵, A.G. Meek⁴, and M.K. O’Banion⁷

¹Medical Department, ²NASA Space Radiation Laboratory, and the ³Nonproliferation and National Security Department, Brookhaven National Laboratory, Upton, NY 11973, USA
⁴Department of Radiation Oncology, Stony Brook University Medical Center, NY 11794, USA
⁵Department of Neurology, Stony Brook University Medical Center, NY 11794, USA
⁶Physics Department, University of Cagliari, Sardinia 09042, Italy
⁷Department of Neurobiology & Anatomy, University of Rochester, Rochester, NY 14642, USA
⁸National Institute on Alcoholism and Alcohol Abuse, Bethesda, MD 20892, USA

Corresponding Author: dilmanian@bnl.gov

INTRODUCTION

Despite recent major innovations in radiation therapy and radiosurgery, there are limitations in treating certain tumors and neurological targets while assuring acceptable damage to the surrounding tissues. These encompass certain tumors in the brain, spinal cord, spinal column, and head-and-neck. This paper describes an experimental radiosurgery, “Interleaved carbon minibeams”¹ being developed at the NASA Space Radiation Laboratory (NSRL), Brookhaven National Laboratory (BNL). The method uses arrays of parallel, thin (~0.3 mm incident beam thickness) planes of carbon ion beams, called carbon minibeams, that “interleave” to produce a solid radiation field at the target (Figures 1 and 2). Therefore, the method delivers a solid field of carbon radiation to the target while exposing the surrounding tissues to single arrays of carbon minibeams, which are well tolerated by tissues.

Fig. 1. Schematic views of two-directional and four-directional interleaved carbon minibeams.
The rationale for the use of incident minibeams arrays is that minibeams are tolerated by tissues, including the central nervous system (CNS) much better than solid beams. The rationale for the use of carbon beams are the following. First, as other heavy charged particles carbons slow down in the matter gradually and deposit much of their energy at the end of their trajectory in a structure called the Bragg peak (Fig. 2, left). This dose-deposition mode allows better dose confinement to the target using the Bragg-peak spreading method. The result is low tissue entrance dose with very little target exist dose (Fig. 2, right). The second rationale for the use of carbons is their high relative biological effectiveness (RBE), which particularly helps controlling radioresistant tumors. Finally, carbons produce a very sharp dose fall-off at the target’s edge, with a typical 80%-to-20% falloff of less than a millimeter compared to 3-5 mm for high energy x rays and for protons.

The tissue-sparing effect of arrays of parallel, thin planes of radiation was first established in the early 1990s with arrays of parallel 25-37 μm planes of synchrotron x-ray beams, called microbeams, in the rat brain at the National Synchrotron Light Source, BNL. The study showed that exposure of the rat cerebellum to 250 Gy in-beam in-depth dose did not produce any damage visible in H&E staining three months later.

It was shown in the early 2000s that arrays of synchrotron microbeams as thick as 0.68 mm still retain much of their tissue-sparing effect, and that to arrays of such thick minibeams can be interleaved (or interlaced) to produce a solid radiation field at the target.

The basis for the tissue-sparing effect of arrays of thin planes of radiation is two underlying effects (see, for example, ref. 3). The first phenomenon is the “dose-volume effect”, which means that the smaller the target the larger its dose tolerance. The second effect is the so-called “prompt biological repair effect” that occurs only for beams with sub-millimeter dimensions and seems to include fast repair of the microvasculature.

We evaluated the efficacy of 4-directional interleaved carbon minibeams as a radiosurgery method by applying it to damage a 6.5-mm target in a rabbit brain and by measuring the resulting damage in the target and in the surrounding tissues in six months using contrast MRI and histological methods.

RESULTS

The carbon minibeams were composed of 0.3-mm incident carbon beams spaced 1.05 mm on-center. The value of beam spacing was determinate using MCNPX Monte Carlo simulations,
with the criterion that there should be no gaps inside the target between the adjacent interleaving minibeams\textsuperscript{4}. Carbon minibeam arrays were produced by a tungsten multi-slit collimator positioned in the path of a broad, uniform carbon beam. The anesthetized rabbit was positioned vertically in the beam on a motorized stage capable of rotating around a vertical axis. Bragg-peak spreading was implemented by using a binary polyethylene filter and by adjusting the beam intensity at each step accordingly (Fig. 2, right). The energies used in Bragg-peak spreading were 124-135 MeV/nucleon. The 6.5-mm target volume received 40 Gy physical dose in a single dose fraction, which would be 120 GyE using an average RBE of 3.0 for the spread-out Bragg-peak region. This is a very high dose for an exposure given in a single session (i.e., single dose fraction).

The rabbit behaved normally during the six-month observation period. Contrast MRI and H&E histology at 6 months showed substantial focal target damage of about 4 mm in diameter at the target, with very little damage to the surrounding brain. The smaller size of the focal tissue damage compared to the target size could be the result of inaccuracies in aiming at the target with carbon minibeam arrays from the four directions.

CONCLUSIONS
These results raise the hope that interleaved carbon minibeams will turn into a new modality of carbon therapy that will produce less damage to the non-targeted tissues and could possibly be administered in a single dose fraction. Because of the method’s requirement for a stationary target, its main applications would be in treating tumors of the brain, spinal cord, spinal columns, head-and-neck, breast, and extremities. The possible low-impact of the method on the non-targeted brain also raises the hope that the method could be used for re-irradiation of recurrent or new CNS tumors.

REFERENCES
A New Beam and Delivery System for Radiotherapy

D. Alezra\textsuperscript{1}, E. Nardi\textsuperscript{1}, S. Koren\textsuperscript{2}, D. Bragilovski\textsuperscript{2}, I. Orion\textsuperscript{2}

\textsuperscript{1} Oncology Dept, Sheba Medical Center, Tel Hashomer, Israel
\textsuperscript{2} Nuclear Engineering Dept, Ben Gurion University, Beer Sheva, Israel

Corresponding Author: dalezra@sheba.health.gov.il

INTRODUCTION

The new beam delivery system consists of an electron accelerator and a system of magnets (one or more). Introducing a transverse magnetic field in and near the tumor, causes the electrons to spiral in this region, thereby producing an effective peak in the depth dose distribution, within the tumor volume. Although the basic idea is not new, we suggest here for the first time, a viable as well as a workable, magnetic field configuration, which in addition to focusing the beam does not interfere with its propagation to the target.

Prototypes were successfully tested by means of computer simulation.

The electron accelerator: can be a linear accelerator or any other type electron accelerator, capable of producing different electron energies for different depths and dose absorption accumulation. The Field size can be as small as a pencil beam and as big as any of the other standard field sizes that are used in radiotherapy. The scatter filter can be used or removed. The dose rate accumulation can be as higher as possible.

The magnets are able to produce magnetic fields. The order, direction, width, place, shape and number of the magnetic fields define the shape and the Percentage Depth Dose (PDD) curve of the electron beam.

RESULTS

Our results suggest that by using an electron beam at different energies, combined with magnetic fields, we could modify the delivered dose. This is caused by manipulating the electron motion via the Lorentz force. The applied magnetic field, will focus the electron beam at a given depth and deposit the energy in a given volume and depth, where otherwise the electron energy will have spread deeper.
The direction and magnitude of the magnetic fields will prevent the scattering of the electron beam and its absorption in remote volumes. In practice, we get a **pseudo Bragg peak depth dose distribution**, applying a relatively low cost system. The therapeutic efficiency induced by the system is of similar efficiency as the ion beam therapy techniques.

FIG.’s 3a, 3b, demonstrate in PDD’s the ability of the technique to localize the energy so as to deliver a sufficiently high energy-dose to a sufficiently small internal target location.

**CONCLUSIONS**

Our novel concept demonstrates treatment that is almost similar to proton therapy and in some parameters even better performance.

Unlike the current high-energy electron therapy, our system's beam deposit almost all of its energy on its target, with a low amount of radiation deposited in tissues from the surface of the skin to the front of tumor, and almost no "exit dose" beyond the tumor. This property will enables to hit tumors with higher, potentially more effective radiation doses, while being considerably less expensive.
Monte Carlo Simulations Study for Small Fields in Stereotactic Radio-Surgery Beams

Ruth Dadush¹,², Annette Wygoda², Itzhak Orion¹

1. Nuclear Engineering Department Ben-Gurion University of Negev, Israel.
2. Radiotherapy Hadassah Ein Kerem hospital, Israel.

Corresponding Author: iorion@bgu.ac.il

INTRODUCTION

Along with the progress of technology in computerization and medical imaging, techniques for radiotherapy treatments were developed, in which small radiation fields of even less than 2 cm were used. Among the most advanced treatments in which small fields in use today are: the Stereotactic Radiosurgery (SRS), Intensity Modulated Radiation Therapy (IMRT), and Stereotactic Body Radiation Therapy (SBRT).

All these treatments are planned by treatment planning systems (TPS) that are based on advanced mathematical algorithms. TPS systems should match each specific accelerator, therefore precise measurements of different parameters of radiation fields should be conducted, such as: FSD (Field Size Dependence), and PDD (Percentage Depth Dose). The selection of detector for the measurements becomes quite a challenge, due to problems in case of small radiation fields. In order to face the difficulties, several researchers compared the measurements of small fields to calculations using ESG4 code Monte Carlo simulations, or older codes. The last one was published in Medical Physics July 2009 by J. Scott et al [1] which used the Monte Carlo EGS4 system. In addition, in Medical Physics January 2008 by J. Das et al [2], that if the entire source cannot be viewed from the center of the field than the geometrical penumbra is extended all over the field cross section. Under such conditions traditional methods for field size determination such as full width at half maximum (FWHM) break down, resulting in overestimated field sizes as Gaussian.

In the latest years, the company Varian has developed a new accelerator "Trilogy", which contains 3 types of photons beams, including a new beam, "SRS beam", with energy of 6 MV with high dose rate of 1000 MU/min. A newer code for Monte Carlo simulation called EGS5 was developed during the past five years [3]. The EGS5 is more advanced and allows significantly longer electron transport step size, and hence shorter computation time than required for identical problems in the EGS4 version.

METHODS

Performing Monte Carlo simulation in new code EGS5 for SRS beam with a 6 MV energy spectrum using small fields "Trilogy" accelerator.

RESULTS

Full simulation of the CC01 ionizing chamber was performed. One example of the simulated cases is to be presented. At first, comparison between CC01 ionizing chamber simulation and CC01 ionizing chamber measurement by PDD has been performed. The results of the simulation and of measurement were approximately identical.
The next step was examining the assumption that small sized fields have a shape of a Gaussian. Comparison between measurements and simulations on a 1 X 1 cm$^2$ field size has been obtained. As can be seen in Figure 2, the results match in all the tested depths. Simulations on 0.5 X 0.5 cm$^2$ and 2 X 2 cm$^2$ size fields have also been performed and will be presented.

CONCLUSIONS

The use of the EGS5 Monte Carlo code system enabled us to inspect the assumption that small sized fields profile behave like a Gaussian. The Monte Carlo simulation results were compared for several field sizes in different depths with ion chamber measurements. The comparisons showed a good agreement.
Figure 2: Measurements and simulations results comparison for field of 1 X 1 cm² in several depths.

REFERENCES


Feasibility of Hippocampal-Sparing Whole-brain Radiotherapy Using ELEKTA Linear Accelerator and CMS Monaco Treatment Planning System.

A. Nevelsky¹, N. Ieumwananonthachai², R. Bar-Deroma¹, R. Ben-Yosef¹, A. Kuten¹
¹ Rambam Medical Center, Haifa, Israel
² Chulabhorn and Siriraj Hospital, Bangkok, Thailand

Corresponding Author: a.nevelsky@gmail.com

INTRODUCTION

Sparing the hippocampus during cranial radiotherapy represents a significant challenge with respect to contouring and treatment planning. Several publications reported about hippocampal-sparing whole-brain radiotherapy (HS WBRT) using Tomotherapy and Varian equipment. The purpose of this work is to evaluate the feasibility of HS WBRT using the ELEKTA Infinity linear accelerator and CMS Monaco treatment planning system (TPS).

Ten patients previously treated with whole-brain radiotherapy (WBRT) underwent repeated planning in attempt to spare the hippocampus region. MRI-CT fusion sets created for every patient were used for delineation of hippocampus, optic nerves and chiasm. Hippocampus avoidance region was generated by expanding the hippocampus volumetrically by 5 mm. RTOG 0933 recommendations were applied for treatment planning. The whole-brain dose was prescribed to 30 Gy in 15 fractions. IMRT plans for the ELEKTA Infinity linear accelerator (MLC leaf width 1.0 cm) were created using CMS Monaco 2.04 TPS based on a 9 field arrangement and step-and-shoot delivery method. The quality of the plans was evaluated using D2% and D98% for whole-brain PTV (defined as whole brain excluding hippocampus avoidance region), D100% and Maximum Dose to the Hippocampus and Maximum Dose to Optic nerves and Chiasm. Homogeneity Index (HI) defined as (D2%-D98%)/Dmean was used to quantify dose homogeneity in the PTV. The RTOG per protocol compliance criteria applied: Whole-brain PTV D2% < 3750 cGy (D2% < 4000cGy is an acceptable variation), D98% > 2500 cGy; hippocampus D100% < 900 cGy and Dmax < 1600 cGy (Dmax<1700 cGy is an acceptable variation); optic chiasm and optic nerves max dose < 3750 cGy.

RESULTS

For the whole-brain PTV mean D2% was 3722 cGy (range 3693-3757cGy) and mean D98% was 2568 cGy (range 2540-2589 cGy). The hippocampus mean D100% was 838 cGy (range 768-892 cGy) and the mean max dose was 1569 cGy (range 1428-1606 cGy). The max dose to optic nerves and optic chiasm for all patients did not exceed 3750 cGy. HI mean value was 0.35 (range 0.33-0.36). Mean number of segments was 105 (range 90-125) and mean number of monitor units was 1155 (range 1081-1279).

CONCLUSION

We show the feasibility of planning HS WBRT using ELEKTA equipment (CMS Monaco TPS and ELEKTA Infinity linac) according to the very demanding compliance criteria defined by the RTOG 0933 protocol.
Determination of Volume Size Limits for IMRT QA

R. Ben Hur, M. Sabbah and Y. Krutman

Medical Physics Unit, Rabin Medical Center, Petah Tikvah, Israel

INTRODUCTION

Ion Chamber (IC) measurement on a solid water slab phantom provides the absolute dose verification of a patient quality assurance (QA) program for intensity modulated radiotherapy (IMRT). A comparison of the planned and measured dose should agree within 3% of the prescribed dose. In this study the expected dose deviations with respect to different planning target volumes (PTV) are measured. The uncertainties of the IC measured error is analyzed and compared with the use of two cylindrical chambers, IBA CC13 (0.13cc) and Farmer (0.6cc). Furthermore, different calculation grid sizes are compared and analyzed. A quantitative limit for small and large fields is defined.

METHODS AND MATERIAL

A solid water slab phantom (CIVCO medical systems, USA) measurements were conducted on a 0.6cc Farmer type (30013 PTW, Freiburg, Germany) and a CC13 0.13cc (IBA) ion chamber at 5cm depth.

All plans were optimized using Eclipse AAA version 8.9 algorithm (Varian Medical Systems, Palo Alto, CA).

A number of 6 to 12 PTV volumes were drawn surrounding the chambers, i.e. 2.17, 4.27, 5.42, 7.35, 9.15, 12.3, 18.76, 26.63, 36.56, 48.21, 63.55 and 80.46 cc. A seven field IMRT plans were optimized for each of the target volumes (PTV). The optimization parameters were kept constant for all volumes i.e. 500 iterations, lower PTV limit at 100% of the volume receiving 70 Gy, upper PTV limit at 0% of the volume receiving 74 Gy and an optimization weighting of 150 for each of the constraints. Two different grid sizes 0.25cm and 0.125cm were calculated, measured and analyzed.

Verification plans were irradiated where the gantry angle was set to 0. Measured and calculated doses were compared to both, the point dose calculation based on a central reference point and a mean dose calculation based on the chambers dose volume histogram (DVH) for the segmented chambers volume.

RESULTS AND DISCUSSION

It is shown that dose deviation between calculation and measurements for all volumes were below 3%. This means that in both small field sizes and large field sizes the chambers used are reliable for IMRT measurements. It is also shown that the use of a smaller grid size reduces significantly the expected measured error to below 1% for all volumes. The average modulation index for all data is 1.285±0.045.
Further analysis of the measured data reveals that large field sizes can be defined for volumes above 18cc. This relates to a field width of 5cm. For these volumes a stable mean deviation value of about 1.44±0.15 from measured dose is expected for 0.25cm grid size and 0.25±0.16 for 0.125cm grid size.

For volumes smaller than 25cc the measured errors are irregularly distributed. Repeated measurements are consistent both in distribution and the deviation from calculated dose. This means that some other parameters that are involved in the dose calculation are responsible for these irregularities (see figure 1).

![Figure 1: Deviation from measured dose for 2.17, 4.27, 5.42, 7.35, 9.15, 12.3, 18.76, 26.63, 36.56, 48.21, 63.55 and 80.46 cc PTV volumes and 0.125cm grid size using a 0.6cc Farmer chamber and 0.13cc IBA chamber](image)

**CONCLUSIONS**

A measurement of a range of IMRT target sizes was conducted and evaluated. Two different IC’s used and compared. Two calculation grid sizes were also assessed.

It is concluded that both chambers are suitable for QA measurements and their deviation from the calculated dose is within the prescribed tolerance. Moreover, a smaller grid size reduces the measured error.

Further investigation will be carried out; well defined geometries will be designed and an increase in modulation index will be assessed.

**REFERENCES**


INTRODUCTION

The stability of boiling water reactors (BWRs) has been a major field of research in nuclear science ever since BWRs were in use. Since the late 1970’s, experiments have indicated the possibility for BWRs to lose their stability under certain operating conditions, characterized by a low coolant flow and high power\(^{(1-3)}\). The source of these instabilities is the strong coupling between the neutronics and the thermalhydraulic properties of the reactor due to the significant void fraction in BWRs\(^{(4-5)}\).

To date, two kinds of such instabilities have been observed in BWR plants\(^{(4)}\):

1. **Global** (corewide) oscillations, where the power produced by the fuel rods in the core oscillates uniformly in phase.
2. **Regional** (first azimuthal mode) oscillations, where the power in half of the core oscillates out-of-phase with respect to the other half while the average power remains essentially constant.

Two main methodologies have been developed to simulate the coupled thermalhydraulic and neutron kinetic behavior of the BWR core. The first, is the use of system codes and coupled thermalhydraulic and neutronic codes\(^{(6)}\) (e.g. RAMONA-3\(^{(5)}\), the LAPUR code\(^{(4)}\), RELAP-PARCS coupled codes\(^{(7)}\)). These codes, however, are computationally expensive for exploring the entire BWR parameter space. A second methodology has therefore been developed that trades accuracy for simplicity and better physical insight. This methodology has resulted in reduced order models (ROMs) that contain a relatively low number of equations and are simple enough for analysis. An important example is the work of March-Leuba *et al.*\(^{(3)}\), who proposed the first simple qualitative model that is able to simulate limit cycle oscillations in BWRs. While different types of ROMs has been developed in the past years\(^{(3,6,8-10)}\), most of these ROMs do not consider continuously space dependent variables, but are rather “point” or “2-point” models\(^{(3,8-9)}\).

In this article we propose a new simplified space-dependent reduced order model, based on (and essentially reducible to) the model proposed by March-Leuba *et al.*\(^{(3)}\), with continuous space dependence of the dynamic variables. Analytical and numerical analyses of this spatially explicit model may shed additional light on the underlying physics which govern instabilities in BWRs.

RESULTS

**Model Derivation**

Our objective is to formulate a minimal spatially dependent extension of the phenomenological model proposed by March-Leuba *et al.*\(^{(3)}\). Our starting point is the space-time dependent one-group neutron transport equation under the diffusion approximation, with one delayed neutron precursors group\(^{(11)}\). In general, all parameters in the transport equation can be space and time dependent. We will assume, however, that the neutron velocity and the diffusion coefficient are constant in time and space. This assumption can be justified by considering only thermal neutrons and assuming that the change in density of the moderator.
water does not affect the diffusion coefficient, and will be accounted for by the other cross sections. To determine the dependence of the nuclear cross sections on time and space we assume the following relation:

\[ (1 - \beta) \nu \Sigma_f(x, t) - \Sigma_a(x, t) = \chi_1 T(x, t) + \chi_2 V(x, t) \]  

where \( \Sigma_f \) is the fission cross section, \( \Sigma_a \) is the absorption cross section, \( \nu \) is the average neutron velocity, \( \beta \) is the delayed neutron fraction, \( T \) is the excess fuel temperature (relative to nominal steady state), \( V \) is the excess void reactivity feedback (relative to nominal steady state) and \( \chi_{1,2} \) are free parameters.

Next we add two equations for \( T \) and \( V \), based on the works of March-Leuba et al. and of Farawila and Pruitt\(^{3,8}\):

\[ T_t(x, t) = -\frac{1}{\tau} T(x, t) + \frac{a}{\tau} (\phi(x, t) - \phi_0) \]  

\[ V_{tt}(x, t) + \xi V_t(x, t) + \alpha^2 V(x, t) = -\kappa T(x, t) \]  

where \( \tau, a, \phi_0, \xi, \alpha \) and \( \kappa \) are free parameters and \( \phi \) is the neutron flux in the reactor.

Equation (2) is simply an energy conservation equation - the temperature increases due to the fissions term and decreases due to the flow term. Equation (3) describes forced damped harmonic oscillations. The temperature functions as the forcing and the minus sign is due to the negative reactivity feedback of the void. By appropriate choice of parameters and the removal of the space dependence, the resulting equations can be brought to a form identical to the model of March-Leuba et al.\(^{3}\). By applying the relation (1) above, we assume that the change in cross sections is due to the fuel temperature feedback and the void feedback only. We further assume that the fraction of the delayed neutrons is negligible and that the processes associated with change in delayed neutrons generally have a significantly smaller time scale than the time scale of the observed oscillations (0.3-0.5 Hz\(^{12}\)). Adding the prompt jump approximation\(^{11}\), we dismiss the equation for the delayed neutron precursors and change the equation for the flux to get our model. We conclude the derivation of the model by transforming the resulting equations to the following non-dimensional form:

\[ \phi_t(x, t) = \phi_{xx}(x, t) + \left( \chi_1 T(x, t) + \chi_2 V(x, t) \right) (\phi(x, t) + 1) \]  

\[ T_t(x, t) = -T(x, t) + \phi(x, t) \]  

\[ V_{tt}(x, t) + V_t(x, t) + \omega^2 V(x, t) = -\eta T(x, t) \]  

where \( \chi_{1,2}, \omega \) and \( \eta \) are free parameters and all the dynamic variables are dimensionless.

**Model Analysis**

Equations (4-6) have two uniform steady state solutions, \( (\phi_0, T_0, V_0) = (0,0,0) \) and \( (\phi_0, T_0, V_0) = (-1,-1,\eta/\omega^2) \). The zero solution represents the normal operation state of the reactor. The second solution represents a state where the neutron population is zero and the reactivity is not zero, and therefore represents a shutdown reactor. Naturally, the second solution is of less interest when searching for instabilities of a normal operation mode of a reactor. First, we analyze the zero solution stability after neglecting the effect of Doppler broadening on the reactivity (taking \( \chi_1=0 \) and \( \chi_2=\chi_2 \)). The effect of the Doppler broadening is discussed later on.

Linear stability analysis of the zero state regarding the different free parameters as control parameters was performed. Of special interest is the reaction of the system to uniform perturbations \( (k=0) \) and to non-uniform perturbations \( (k=1) \), which may account for global and regional instabilities. The three control parameters define a three-dimensional space \( P \). For each point in \( P \) we determine (by linear stability analysis) whether the zero solution is
stable, loses stability to homogeneous \((k=0)\) perturbations, to non-uniform \((k=1)\) perturbations or to both. Of more interest to us are the stability properties of the zero state with respect to actual physical parameters, specifically the nuclear power of the reactor and of the coolant flow. The reactor power and the coolant flow are proportional respectively to the parameters \(\phi_0\) and \(\mu = 1/\tau\), that appear in equation (2). These parameters form a two dimensional space, \(Q\). Each point in \(Q\) can be associated through linear stability analysis with its stability properties, illustrated in the phase diagram in figure 1 for the domain \(\mu \in [0,1.2]\) and \(\phi_0 \in [0,0.8]\). The other parameters used were the parameters calculated by LAPUR code for Vermont Yankee Test 7N\(^{(5)}\).

Figure 1 illustrates that the system becomes more stable as the flow increases. A large flow prevents a significant excess void fraction to appear, since it manages to remove the heat produced in the fuel before it is transferred to the water. However, increasing the reactor power can make the system unstable by allowing for a large excess void fraction to appear. In addition, the regional instability exists only at low power and flow values. Regional oscillations are a result of asymmetries in the pressure drop and the flow in the core\(^{(5)}\), which develop at low flow and power values. These qualitative results, achieved by linear stability analysis, generally agree with known results\(^{(3-6,10)}\). Another behavior of the model illustrated in figure 1 is the transition between \(P_2\) and \(P_3\) in low power and flow values. For power values higher than a critical value, there is no point in \(Q\) for which the zero solution is unstable under regional perturbations and stable under global perturbations and vice versa, i.e. for power values lower than this critical value there is no point in \(Q\) which shows instability under non-uniform perturbations without being unstable under uniform perturbations, as well. The general behavior depicted in figure 1 agrees with known results. However, according to March-Leuba and Blakeman\(^{(5)}\), the system cannot be stable under non-uniform perturbations and unstable under uniform perturbations for power values higher than the transition point power value. This is due to Doppler broadening, which was not taken into account in the calculation presented in figure 1.
The results presented above do not provide information about the asymptotic behavior of the model. Therefore, numerical integration of the model was performed, taking into account a one-dimensional slab geometry with reflective boundary conditions. The non-dimensional slab width was chosen to be equal to $\pi$, so that the eigenfunctions satisfying the boundary conditions are of the form $\cos kx$, where the wavenumber $k$ is integer. The numerical calculations were performed using a code that implements the predictor-corrector method (13). The numerical integration results confirm the linear stability analysis predictions presented in figure 1, and furthermore, it illustrates that all the instabilities are oscillatory, i.e. the model variables oscillate in time. The numerical integration also allows studying the late stages of the system’s dynamics, where nonlinear saturation leads to constant oscillations amplitudes. The asymptotic behaviors associated with the different domains in figure 1 are:

- $P_1$ - Convergence towards the stable zero solution.
- $P_2$ - Formation of constant amplitude global (uniform) oscillations.
- $P_3$ - Formation of constant amplitude regional oscillations.
- $P_4$ - Bistability of two pure state solutions, for which global or regional oscillations appear, depending on the initial conditions.

The dynamical behavior of the system for fixed reactor power and various flow values is summarized in the bifurcation diagram shown in figure 2. For a sufficiently large flow values, the normal operation solution $(0,0,0)$ is stable. Decreasing the flow, a supercritical Hopf bifurcation emerges. For the particular power value used, the bifurcation is to global oscillations, while for lower values instability to regional oscillations can take place. The other fixed point of the model - the shutdown reactor solution, is unstable for all values of $\phi_0$ and $\mu$, since it represents a state of zero flux and positive reactivity. Therefore, adding even the smallest neutron population will cause the neutron population growth.

**The Effect of the Doppler Reactivity Coefficient**

We will repeat the linear stability analysis of the zero state, this time, considering a negative Doppler reactivity coefficient. For small absolute values of $\chi_1$ (see equation (4)), the stability characteristics of the model are essentially identical to what is depicted in figure 1.
Decreasing the value of $\chi_1$ (i.e. more negative), the system is stabilized gradually in terms of the flow, meaning the system is stable for lower flow values. This result was expected, since decreasing the Doppler reactivity coefficient is equivalent to increasing the absorption cross section of the moderator in high temperatures. Therefore, when instabilities occur and the temperature grows, its feedback will be large enough to restrain the neutron flux. When the Doppler reactivity feedback is sufficiently strong, the void fraction effect is negligible and the zero solution is stable. The results for the case $\chi_1=-100$ are presented in figure 3. The nominal physical relevant values are in the range $\chi_1=-200 \text{ - } -10$, according to the values specified by March-Leuba et al.\(^{(3)}\).

It is possible to identify a transition point between $P_2$ and $P_3$ in figure 3, but this time the behavior is similar to the behavior presented by March-Leuba and Blakeman\(^{(5)}\). The source of this behavior is that the stabilizing effect of the Doppler broadening is mode dependent, i.e. decreasing the Doppler coefficient generally stabilizes the global mode more than it stabilizes the regional mode. This effect is a direct result of the different mechanisms characterizing the instabilities to global and regional oscillations. In the instability to global oscillations, the neutronic feedbacks are large comparing to the flow feedbacks, because the flow oscillations are damped (due to changes in the pressure inside the coolant channels). In the instability to regional oscillations, the situation is reversed\(^{(5)}\). In addition, the asymmetry associated with the regional instability, reduces the effect of the Doppler broadening effect on reactivity, since it affects in opposite directions the two halves of the core. Therefore, changing the Doppler coefficient will obviously have a greater effect on the corewide instability than on the regional instability. The conclusion drawn from this result is that the Doppler reactivity feedback cannot be omitted when studying regional instabilities, since it affects distinct modes differently.

**CONCLUSIONS**

In this paper we described the derivation and analysis of a one dimensional space-dependent reduced order model for studying oscillatory instabilities in BWRs. The linear stability of the model was analyzed both analytically and numerically and the results obtained resemble past...
known results. The global and regional instabilities observed in BWRs were identified and characterized by the stability analysis. The significance of the Doppler broadening effect was discussed. It was found that the Doppler reactivity feedback has, in general, a stabilizing effect on the system. However, it affects differently the distinct spatial modes of the neutron flux. Therefore, neglecting this effect may lead to inaccurate qualitative results when studying regional instabilities.

ACKNOWLEDGEMENTS
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Implementation of Drift-Flux Model in the THERMO Module

Y. Shaposhnik\(^1\), E. Shwageraus\(^1\), E. Elias\(^2\)

\(^1\)Ben-Gurion University of the Negev, P.O.B. 653 Beer-Sheva 84105, Israel
\(^2\)Technion – Israel Institute of Technology, Technion City - Haifa 32000, Israel

Corresponding Author: shaposh@bgu.ac.il

INTRODUCTION

The BGCore\(^1\) system has been upgraded by implementing a drift-flux model to accurately simulate boiling water reactor operation under the full range of standard steady-state operation, along with the capability to fully track the spatial isotopic distribution. The drift-flux model extends the more widely used homogeneous equilibrium model (HEM) by accounting for slip between the vapor and the liquid phases. This has been shown to affect pressure drop and the void fraction distribution along the core, which in turn affect the moderator density, the neutron spectrum and the isotopic composition of the fuel.

METHODOLOGY

The basic concept of the drift-flux model is to consider the mixture as a whole, while relying on a set of constitutive relations for evaluating the velocity ratio between the two phases. Here, we have adopted the Zuber-Findlay\(^2,3,4\) correlation for void fraction, which considers the slip ratio as given in (1) independent of the flow regime.

\[
\alpha = \frac{X_t}{C_0 \left[ X_t + \frac{\rho_g}{\rho_f} \left(1 - X_t\right)\right] + \frac{\bar{u}_g \cdot \rho_g}{G}} \tag{1}
\]

Where \(\rho_f\) and \(\rho_v\) are the liquid and vapor densities, respectively, \(\bar{u}_g\) is the drift velocity, \(G\) is mass velocity and \(X_t\) is a pseudo-mass vapor quality. The parameter, \(C_0\), in (1) represents an empirical factor correcting the one-dimensional homogeneous theory to account for the fact that the concentration and velocity profiles along the channel can vary independently. For using eq. (1) Zuber\(^5\) suggested,

\[
\bar{u}_g = 1.41 \cdot \left[\frac{\sigma \cdot g \cdot (\rho_f - \rho_g)}{\rho_f^2}\right] \quad \text{and} \quad C_0 = 1.13 \tag{2}
\]

where \(\sigma\) is surface tension and \(g\) acceleration due to gravity.

The drift flux model is useful mainly when the drift velocity is significant compared with the total volumetric flux. Thus, limiting its application to bubbly, slug and churn flow patterns only. To extend its utility to other flow patterns encountered in BWR core and to introduce continuity in the void fraction, we have chosen to use a pseudo-mass vapor quality, \(X_t\), in eq. (1) instead of equilibrium vapor quality, \(X_e\). \(X_t\) was defined by the following quality profile fit of Ahmad\(^6\).
\[
X_d = \frac{X_v - X_d \cdot \exp \left( \frac{X_v}{X_d} - 1 \right)}{1 - X_d \cdot \exp \left( \frac{X_v}{X_d} - 1 \right)}
\]

where \( X_d \) is an equilibrium vapor quality at the point of bubble detachment, given by,
\[
X_d = \frac{C_{pf} \cdot (\Delta T_{SUB})_b}{h_{fg}}
\]

where \( C_{pf} \) is the specific heat of liquid phase, \((\Delta T_{SUB})_b\) is the degree of subcooling at the point of bubble detachment, and \( h_{fg} \) is the latent heat of vaporization.

MODEL VERIFICATION

In order to verify the implementation of the drift flux model into THERMO\(^7\), a code to code comparison was performed. The DYN3D\(^8\) code was used as a reference code to simulate a hypothetical generic PWR 3×3 lattice with 22 axial nodes each of 15 cm height. DYN3D is based on four balance equations for mixture mass, momentum and energy and vapor mass balance. The two-phase model is closed by the following additional assumptions:
- One of the phases (vapor or liquid) is in saturation conditions.
- The relation of phase velocities is described by a quasi-stationary phase slip model, which means, an algebraic equation for the phase slip ratio is applied.

Table 1 summarizes the main parameters of the lattice configuration and Figure 1 depicts the axial fuel pin power profile. The axial power distribution was chosen to have multiple high power regions interspersed by low power regions. Such heterogeneous fuel configuration was proposed to increase fertile to fissile conversion ratio in LWRs\(^9\).

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
<th>Units</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mass flow rate</td>
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<td>[kg/sec]</td>
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<td>Inlet pressure</td>
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<td>[MPa]</td>
</tr>
<tr>
<td>Inlet temperature</td>
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<td>[°C]</td>
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<tr>
<td>Flow area</td>
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<tr>
<td>Heated diameter</td>
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<tr>
<td>Wetted diameter</td>
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<td>[cm]</td>
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<table>
<thead>
<tr>
<th>Parameter</th>
<th>Dimension [cm]</th>
</tr>
</thead>
<tbody>
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<tr>
<td>Gap Diameter</td>
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<tr>
<td>Clad Diameter</td>
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<tr>
<td>Assembly pitch</td>
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<tr>
<td>Unit cell Pitch</td>
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<tr>
<td>Control rod inner radius</td>
<td>0.6120</td>
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<tr>
<td>Control rod outer radius</td>
<td>0.5715</td>
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</table>
In order to simulate boiling conditions, the inlet pressure was set to 7.5 MPa. Two main parameters of interest have been compared: vapor quality and void fraction along the axial direction. Figure 2 shows the THERMO predictions for three types of void correlations; the first (THERMO - Basic) refers to HEM model and neglecting subcooled boiling, the second (THERMO - SUB) refers to HEM with subcooled boiling region, and the third one (THERMO - Drift) includes drift flux model and subcooled boiling.

As shown in Fig. 2, the exit vapor quality results are in good agreement for all cases. As expected, the drift-flux model yields lower void fraction along the fuel. Since void formation in the subcooled region is not considered in the case of THERMO–Basic calculations, large deviations are noticed at the lower core section before the coolant reaches its boiling temperature. Subcooled boiling starts at about 70 cm from the inlet and develops into nucleate boiling at ~140 cm. Obviously, neglecting subcooled boiling in the THERMO–Basic model leads to large deviations in that region. The differences between the DYN3D and the THERMO–Drift results are due to different correlations adopted in the codes for vapor quality in the subcooled region. Nevertheless, once the nucleate boiling is established the codes give similar results.

CONCLUSIONS
A drift flux model was implemented in the THERMO module. The model implementation was verified by comparison with DYN3D predictions of a hypothetical 3×3 lattice. It is
shown that using the drift flux model and considering a subcooled boiling in the lower section of the core yield good agreement with DYN3D results for vapor quality and void fraction. The THERMO module is currently used to study the effect of various void correlations on the neutronic behavior and breeding performance of a heterogeneous BWR assembly.

REFERENCES


Neutronic Analysis in High Conversion Th-233U Core with ELCOS System

D. Kotlyar, E. Shwageraus
Ben-Gurion University of the Negev, P.O.B. 653 Beer-Sheva 84105, Israel

Corresponding Author: kotlyar@bgu.ac.il

INTRODUCTION

This paper reports on preliminary core analysis of a PWR loaded with Th-233U fuel and operating in close to self-sustainable fuel cycle. Achieving such self-sustainable with respect to fissile material fuel cycle would practically eliminate concerns over nuclear fuel supply hundreds of years into the future. High conversion ratio can be achieved in LWRs through careful optimization of the fuel assembly design. Moreover, utilization of LWR technology and its associated vast experience would allow faster deployment of such fuel cycle without immediate need for development of fast reactor technology, which tends to be more complex and costly. Previously reported (1,2) strategy for optimization of fuel assembly design was based on understanding and exploiting basic physics effects that lead to high conversion ratio. Possible strategies included, for example, transition from homogeneous to heterogeneous seed-blanket (SB) fuel geometry. The results of these neutronic analyses performed for an infinite fuel assembly lattice showed that some of the SB assembly designs (3) are capable of achieving net breeding at least for a portion of the fuel cycle.

However, previously derived conclusions were based on an infinite assembly lattice calculation which relied on several assumptions, such as: leakage reactivity worth of 3% and equal power sharing between the batches. In addition, since the assembly reactivity exhibited highly non-linear behavior, the fuel in-core residence time was evaluated by assuming integral neutron balance over the fuel lifetime.

The main objective of this paper is to confirm the assembly level findings by performing full core analysis with regards to breeding potential, fuel cycle length and in-core power distribution. The results presented in this paper indicate that the assumptions used in the assembly lattice level calculations are fairly accurate. The calculations in this study, which included few group cross section condensation and neutronic-depletion core analysis, were performed with ELCOS (4) system. The results obtained in the current study will be used to determine maximum achievable core power density which assures that all major thermal margin requirements are met. These thermal design considerations may suggest modifications to the assembly design which was optimized only with respect to breeding capabilities.

In the future, a number of iterations may be required between assembly and core level calculations which will include thermal hydraulic feedbacks.

METHODOLOGY

In this study, the nodal diffusion SILWER (4) code was used to perform 3D neutronic analysis. This code is a part of ELCOS (4) system, which is designed to perform a complete LWR core analysis. SILWER code requires a set of pre-generated few group homogenized parameters, that were prepared by BOXER (5) - a fuel assembly transport and burn-up code. Its cross section library is based on JEF-1 nuclear data (6). Previously reported strategy (1,7) for optimization of the fuel assembly design was based on understanding and exploiting basic physics effects that lead to high conversion ratio. These strategies included transition from homogeneous to heterogeneous SB fuel geometry, optimization of the seed and blanket region dimensions to fit the neutron migration length in the corresponding regions, optimization of
blanket fuel pin diameter and the use of heterogeneous duplex seed pin. The most favorable design was 11x11 pins fuel assembly setup. Then, the Monte Carlo simulated annealing algorithm was applied to confirm this solution. Later on, the simulated annealing algorithm was used\(^7\) to identify the most favorable seed and blanket fuel pins arrangement in a traditional 17x17 pins fuel assembly. Relative volume fractions of seed and blanket regions remained the same as in the 11x11 assembly case in order to maintain the same core average power density, which happens to be limited by the seed pins linear power. Nearly identical breeding performance was achieved in both lattices. In this study, a seed blanket 17x17 pins fuel assembly was used in the 3D core modeling simulation as presented in Fig.1.a.

The objective of 3D core analysis is to confirm the findings of the previous assembly lattice design stage, mainly with regards to the cycle length and discharge burn up. Since reactivity as a function of time curve for the infinite fuel assembly lattice exhibit highly non-linear behavior, the traditional approach of estimating fuel discharge burnup and other core average characteristics on the basis of assembly lattice calculations using Linear Reactivity Model\(^8\) are no longer valid. Instead, the fuel in-core residence time was evaluated using Eq. 1

\[
2 \times \int_{0}^{T} (\rho(t') - \rho_L) dt' = \int_{T}^{T_e} (\rho(t') - \rho_L) dt'
\]

where, \(\rho(t)\) is the reactivity curve obtained from the neutronic calculations and \(\rho_L\) is the assumed leakage reactivity worth (taken as 0.03), \(T\) is the time point at which the assembly becomes super-critical. The parameter \(t\) in Eq. 1 is the fuel residence time, during which a 3-batch core can in principle sustain criticality. Eq. 1 effectively represents the integral neutron balance over the fuel lifetime time, where twice as many excess neutrons are produced by the fuel than required to sustain criticality (in a 3-batch core). However, from this assumption the net neutron balance at any given point in time during the fuel cycle is unknown. The validity of such assumptions can only be verified by performing a full core burnup analysis, which is the main objective of this work. Therefore, a full core model was developed and its layout is presented in Fig.1.b. One of the major design challenges associated with the SB concept is high power peaking in the seed region within the assembly. The greatest power imbalance between seed and blanket regions of individual fuel assemblies occurs in the seed region at the beginning of life. Therefore, the fresh assemblies were arranged in the periphery of the core in order to reduce the assembly average power peaking factors.

a. Quarter of 17x17 SB assembly  b. Quarter of the SB core

Figure 1. ELCOS model of a Th-\(^{233}\)U core and fuel assembly
FULL CORE NEUTRONIC RESULTS

The core analyses included several stages:

- Assembly level calculations and preparation of 2-group parameters for SILWER calculations. At this stage, the condensed parameters were prepared only as a function of burnup. This procedure was done for various assembly average enrichments in the range of 2.9\(^\circ\)/0 till 4.3\(^\circ\)/0.
- Then, the assembly cycle length was evaluated according to the described neutron balance presented in Eq.1.
- For each of the enrichment values, the core cycle length was calculated by performing multiple cycle re-load calculations until equilibrium burnup distribution between the batches was obtained.

Fig.3 presents the core cycle length (blue curve) and fissile inventory ratio (FIR) as a function of enrichment. FIR is the ratio of instantaneous fissile nuclides inventory to that at the beginning of fuel irradiation.

The FIR values (red curve) were evaluated at the obtained core discharge burnup (Fig.3) for each of the enrichment values.

It can be seen from Fig.3 that for an annual cycle, the FIR value is about 0.96. Traditionally, the cycle length is determined by identifying the time where the core reactivity (reactivity average of all batches) is equal to zero. However, in the case of Th-U233 fuel, the assembly is subcritical at BOL and its reactivity gradually raises and becomes positive later on in the cycle. Therefore, in this case, the cycle length for a given enrichment is determined by requiring BOC rather than EOC core reactivity to be zero. As a result, higher enrichment values yield shorter cycle length because less burnup (less positive reactivity) is required to compensate the initial fresh batch sub-criticality. This fact is illustrated in Fig. 3, where the calculated cycle length periods decrease with enrichment.

Fig. 4 presents the obtained cycle length from 3D calculations (blue curve) against the evaluated 2D cycle length (red curve). It can be seen that there is a considerable difference between the cycle length values obtained in the 2D and full core calculations. However, the differences are relatively small (below 7%) for the cycle lengths around 12 months.
CONCLUSIONS
This work presented results of a neutronic 3D simulation of a Th-233U fuel core. Possibility of achieving self-sustainable Th-233U fuel cycle was investigated in previous reports and proven to be possible in principle. However, the calculations reported previously were performed for 2-dimensional infinite fuel assembly lattice. These analyses included calculation of cycle length and evaluation of the maximum achievable core power density, based on linear reactivity model assumptions. In order to confirm these assumptions, realistic 3D core neutronic calculations were performed in this study. The obtained results show that critical core can be designed with seed-blanket fuel assemblies, which has net breeding at least for a part of the fuel residence time. The results indicate that higher burnup is achieved in 3D calculations than assumed in the assembly level calculations. This fact slightly reduces the breeding capabilities.

Although the calculations were performed with a 3D core model, they are still should be considered as preliminary since no loading pattern optimization was performed which resulted in somewhat high power peaking. Such optimization will be performed in the future.

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INTRODUCTION

One of the main objectives of reactor safety systems is to keep the reactor core in condition that does not permit any release of radioactivity into the environment. In order to ensure this, the reactor must have sufficient safety margins during all possible operational and accident conditions. This paper focuses on the analysis of loss of coolant accident (LOCA), which is one of the most severe scenarios among other hypothetical events such as reactivity induced accidents, loss of flow accident, etc.

The analysis was carried out for the Israel Research Reactor 1 (IRR-1), which is a 5MW swimming pool type research reactor. The IRR-1 core consists of MTR highlyenriched uranium (HEU) fuel type, and is reflected by Graphite elements. During normal operation, the reactor core is cooled by downward forced flow of light water circulated by a primary cooling circuit pump. But during shutdown stage, the reactor core is cooled by upward natural convection flow through a safety flapper valve.

There could be several primary causes to initiate a LOCA in research reactors, such as breaks in the piping system, ruptures of the beam tubes, and concrete wall failures of the reactor pool. Although probability of large break accident in research reactors is very low, once the accident occurs, it may cause major core damages, so it must be considered.

ACCIDENT SCENARIO

The present analysis assesses the possibility of passively cooling the IRR-1 core and determines the clad wall temperature in case of complete LOCA event. The following accident scenario was considered following an initiating of LOCA event:

- At time = 0 the reactor SCRAMoccurs as a result of LOCA.
- About 20 min later the reactor core is completely uncovered and exposed to the ambient air. The core uncover time was minimized by assuming that the failure occurs at the largest pipe at the bottom of the pool, i.e. at the 10 inch diameter outlet water pipe.
- The uncovered core is cooled passively by upward natural convection flow of air and thermal radiation to containment chamber walls.

METHODOLOGY

A mathematical model for the complete LOCA analysis has been developed. The reactor core was assumed to lose heat only by natural convection and thermal radiation. Conduction in the fuel plates only serves to equalize the element temperatures across the reactor core. Conduction losses to the grid plate were ignored.
Based on theory and previous analysis\(^{(2)}\), it was assumed that both the fuel plate temperature and residual heat generation are uniform. A schematic view of adjacent fuel elements is shown in the Fig. 1.

![Schematic view of adjacent fuel elements](image)

**Figure 1:** A schematic view of adjacent fuel elements

Under the above assumptions, the average temperature transient was evaluated by the following energy balance equation

\[
\left[q_{\text{residual}} - q_{\text{radiation}} - q_{\text{convection}} - q_{\text{conduction}}\right] = m \cdot c_p \cdot \frac{dT_w}{dt}
\]

with initial condition \(T_w = T_{w0}\) at \(t = t_0\).

Evaluations of individual terms of this equation are presented in reference\(^{(1)}\).

Three mechanisms of natural convection heat losses were considered in the model:
1. from the channels between the fuel plates,
2. from the channels between adjacent elements, and
3. from the element’s free faces on the sides of the reactor core.

Radiation heat transfer effects are often significant relative to natural convection. Three mechanisms of thermal radiation heat losses were considered:

1. from the element’s free faces to the containment chamber walls,
2. from the top opening of the elements, and,
3. between surfaces of opposing elements.

It should be noted that the third mechanism does not remove any heat from the reactor core, but only tends to equalize the elements temperature.

The model was validated by performing LOCA analysis for the Ford Nuclear Reactor (FNR) and comparing the obtained results with the Science Applications Inc. (SAI) analysis \(^{(2)}\). The
validation showed that the developed model is quite suitable for analysis of LOCA in pool type research reactors.

RESULTS AND DISCUSSION
The IRR-1 core is an assembly of standard and control fuel elements mounted on a grid plate with 30 holes in a 5×6 pattern. The allocation of the elements on the grid plate is shown in Fig. 2. The decay heat generated in these elements are attributed to the assumed operational schedule of the reactor\(^3\), which are

- 580 fuel plates and steady-state power of 5 MWt,
- 8 hours of operation in a working shift per week,
- Average fuel depletion = 30% of initial fissile content.

![Figure 2](image)

Figure 2: Typical core layout in IRR-1 with graphite elements reflector

The reactor core is reflected by graphite elements. Since the graphite reflector elements surrounded the core are typically solid that does not generate heat, they have the potential to act as a heat sink. Therefore, the LOCA analysis was performed in two modes

1. Neglecting thermal coupling between the core and graphite reflector elements.
2. Considering perfect thermal coupling of the core with the graphite reflector elements.

In both analyses, the natural convection and radiation heat losses from the AC and BC element were ignored. The core temperature as a function of time is presented in Fig. 3, assuming the core is completely uncovered within 1200 sec following SCRAM.
It is shown that assuming no thermal contact with the graphite reflector elements, the IRR-1 core is predicted to reach melting point after about 3200 sec from SCRAM. On the other hand, for a core with perfect graphite heat sink, the core temperature is predicted to remain indefinitely below the clad melting point.

**SUMMARY**
Detailed LOCA analysis has been performed for the 5MWth swimming pool type Israel Research Reactor 1 (IRR-1), which utilizing highly enriched uranium fuel. The analysis was aimed to assess the possibility of passively cooling the exposed reactor core by natural convection of air and thermal radiation. The core uncover time was estimated by assuming that the LOCA was induced by a guillotine break of a 10 inch water pipe at the bottom of the pool, causing the core to uncover about 20 min after reactor SCRAM.

The effect of the reflector on the core cooling was studied by comparing the total heat transfer from the core with and without considering the thermal contact between the core and the graphite reflector elements. It has been shown that for uncover time of 20 min the core may reach melting point if thermal contact with the graphite is neglected. On the other hand, considering perfect thermal contact between the core and the graphite reflector, the core temperature is predicted to remain below the clad melting point.

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Examination of AP-1000 Passive Core Cooling System During Lost of Flow Accident

L. Bar-Or¹, Y. Yasur¹, Y. Weiss¹

¹Rotem Industries LTD, Mishor Yamin, D.N Arava, 86800, Israel

Corresponding Author: liranbo@rotemi.co.il

INTRODUCTION

The AP – 1000 nuclear power plant is a III+ generation nuclear power plant, having two steam generation loops with 3400 MW thermal power produced by 41,448 UO₂ fuel rods assembled in 157 fuel assemblies and 1170 MW electric power.

As a III+ generation nuclear power plant it utilizes passive cooling systems for the core and for the containment. These systems depend solely on natural physical mechanisms such as gravity, natural circulation and compressed gases to generate the driving forces for passive cooling during an accident. These passive systems do not utilize active components such as pumps, fans, chillers and generators to satisfy safety requirements. One of these existing passive systems which are placed in the first cooling loop is called Passive Residual Heat Removal (PRHR), which takes a crucial role during a Loss of Flow Accident (LOFA).

AP-1000 power plant has three major cooling loops, while the first loop is related to the main core cooling system, the second loop is related to the electricity generation loop and the third loop exchanges heat with the environment.

During nominal working conditions the first cooling loop includes the core and the steam generator. Through the hot legs pipes, the hot water (coming out of the core) enters the steam generator, exchanges heat with the second loop and returns through the cold legs pipes to the core entry.

While LOFA occurs, the electrical feeds of the main cooling pumps stop and the fluid is still circulated, for a short time, as a result of the pumps inertia, without any cooling water loss. Forced flow cannot exist and natural circulation flow (thermo syphon flow), which is driven by the difference in density along the first cooling loop is being built. At this moment, the PRHR-HX, which is immersed in the core storage tank, is passively being connected to the hot leg and is improving the heat removal from the core.

The US Nuclear Regulatory Commission (US NRC) licensing defines the safety requirements document for the AP-1000 nuclear power plant. The document defines "Cold shutdown" which sets the maximal cooling water temperature to 93.3 degC, cooled solely by safety related systems (passive systems). The time needed to reach this temperature is 36 hours since the beginning of the accident.

Since the low driving force in the passive cooling systems, it is expected that the PRHR will not fulfill the US NRC cold shutdown requirements. Hence, a modified definition "Hot shutdown" was declared and sets the maximal cooling water temperature to 215.6 degC after 36 hours, while using safety related systems. During the following 36 hours, the maximal cooling water should decrease to 93.3 degC, while using non safety-related systems is allowed.

This work examines the ability of the AP-1000 first cooling loop, to comply with the US NRC "Hot shutdown" requirement.
The work was conducted in an ongoing corporation program with the Israeli Electrical Company, in objective to develop a source of knowledge for possible future electrical power plant in Israel.

MODEL AND RESULTS
A preliminary one phase analysis of the AP-1000 natural circulating flow in the first loop is performed by dividing it to 4 separate components, as shown in figure 1. Segment 1-2 represents the core, segment 3-4 represents the heat exchanger and segments 2-3 & 4-7 represent the hot leg and the cold leg respectively. A model describing the natural circulating flow, through the first cooling loop, was built using the Engineering Equation Solver (EES) software. The model calculates the temperature distribution, the pressure drop and the mass flow rate along the flow path of the loop components.

To analyze the core behavior during LOFA, a representative open channel based on the gap between four adjacent fuel rods was used. In each segment along the loop presented in fig.1, a 1D, coupled momentum and energy equations were solved, assuming one phase S.S approach. The time was implemented in the model as an outcome from the residual heat time dependent function (the well-known W.V function).
Since the height of the fuel rods is 4.26 m, the temperature distribution was calculated using a total of 426 elements along the representative open channel. The output of the last element was considered to be the core exit temperature, which is the maximal cooling water temperature along the first loop.

Modeling of the heat exchanger for exit temperature and total pressure drop requires exact technical data of its components (flow separator, number of tubes and geometrical features of the tubes). Since the exact data was not fully available, the value of the minimal "heat exchanger parameter- UA" needed for operation was used, as was published in the core technical data\(^1\). The pressure drop along the heat exchanger was estimated, from a representative geometry found in the available data.

The solution method was an iterative process by guessing initial mass flow rate value in the entrance regime of the core, heat balances on the mentioned above segments can be conducted so temperature and density can be calculated along the loop. With these calculated parameters the momentum balance can be conducted to calculate a new mass flow rate value until the iterative procedure is converged.
Three separate time regimes were identified:
- 60 seconds after shutdown where two phase flow is possibly generated.
- From 60-100 seconds after shutdown a transition regime, which the fluid is on saturation limit, is detected.
- After 100 seconds, the flow is certainly one phase flow and the results are acceptable from the preliminary analysis conducted herein, as shown in figures 2-4.

The results show that after 36 hours since the accident the cooling water temperature is less than 150 degC and that the maximum temperature of the fuel rod is almost equal to the water temperature.

**CONCLUSIONS**

The results show that although many assumptions were taken, the AP-1000 is compliance with the US NRC requirements and hot shutdown meets the needs of the first loop cooling. It was identified that the current EES model does not fit with two phase flow and modification is needed to be done.

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Intrduction

In the last half century, extensive experimental and theoretical works have been performed on boiling Critical Heat Flux (CHF) phenomena. CHF is a condition in which a small increase in heat flux leads to abrupt wall overheating, caused by the transition or Departure from Nucleate Boiling (DNB) to film boiling. Much effort was devoted to understand the phenomena and to develop design predicting correlations. Since the foregoing studies are mostly related to nuclear power plants, most of the existing experimental data is at a high pressure (say 2.5-20.0 Mpa) and high flow velocity and so are the correlations which have been developed. Other works in the literature tried to understand the pool boiling CHF mechanism which characterizes by a natural convection driving force and very low velocities.

The last nuclear accident in Fokushima which was a result of electrical supply failure due to the Tsunami was a trigger for researches on cooling accidents in nuclear reactors. In case of Loss Of Coolant Accident (LOCA) or Loss Of Flow Accident (LOFA) in nuclear power plains or in a research reactors the flow velocity and/or the pressure decreases. In this case the CHF in the fuel channels is occurring due to another mechanism which differs from the pool boiling or forced convective flow boiling CHF. The understanding of the fundamental nature of that kind of CHF in vertical flow channel under stagnant or low flow conditions is important for reactor safety.

In a case of vertical channel with a large liquid volume the CHF mechanism is similar to the Departure from Nucleate Boiling (DNB) in pool boiling. When the vertical heated channel becomes narrow and long like in a fuel channel, the CHF mechanism which limiting the residual heat removing from the fuel calls Flooding CHF. In this mechanism a countercurrent flow is formed in the vertical flow channel by the vapor which flows in the upper direction and the water which flows in the other direction due to the gravitation. Several investigators (Barnard et al. 1974, Mishima and Nishihara 1987, Chang et al. 1991, Park, J. W. et al. 1997 and S-Y. Chun et al. 2001) have conducted experiments under those conditions of stagnant or low flow rate between zero to several hundred kg m$^{-2}$ s$^{-1}$. Mishima conducted his experiment with one side heated annular channel and with rectangular channel and round tube. He compared his results with several correlations and found that the best prediction of the experimental results was achieved with his correlation which was a modification of Wallis correlation:

\[
q^* = q_{cF}^* + \frac{A}{A_H} \frac{\Delta h}{h_f} G^*
\]
The second term on the r.h.s of eq. 1 is the contribution of the enthalpy increase to the saturation condition of water and the first term is the heat flux in the CHF condition under flooding in the saturation condition as follow:

\[ q_{i,F}^* = \frac{A}{A_H} \frac{C_w^2 \sqrt{D^*}}{1 + \left(\frac{\rho_g}{\rho_f}\right)^{0.25}} \]  

In eq. 2, \( G^* = G/\sqrt{2 \rho_g g A \rho} \) and \( q^* = q/\sqrt{2 \rho_g g A \rho} \). Mishima reported that the flooding CHF was well reproduced with \( C_w = 1.66, 0.98 \) and \( 0.73 \) for tubes annuli and rectangular channels respectively. Equation 1 presents the CHF in case of low flow rate which is a superposition of the Flooding CHF (eq. 2) and the influence of the flow rate \( (G^*) \).

Several investigators have employed Kutateladze's criterion for the onset of flooding given by the following expression:

\[ K_{g}^{0.5} + mK_{l}^{0.5} = C_k \]

Where \( K_g \) and \( K_l \) are defined by: \( K_i = j_i \rho_i^{0.5} (g \sigma \Delta \rho)^{-0.25} \) (The subscript \( i = l \) or \( i = g \) for liquid or vapor respectively). \( C_k \) and \( m \) are constant which were defined by Pushkina and Sorokin as \( m = 0 \) and \( C_k^2 = 3.2 \) for flooding conditions. Tien and Chung (1979) took \( m = 1 \) in eq. 3. and after substitution \( j_g \) and \( j_l \) they got the following expression:

\[ \frac{q_{C,B}}{h_g (g \sigma \rho_g^{1/4} \Delta \rho)^{1/4}} = \frac{C_k^2}{4} \left( \frac{D_{gy}}{L_g} \right) \left[ 1 + \left( \frac{\rho_g}{\rho_l} \right)^{1/4} \right]^{-2} \]

They defined the constant \( C_k \) as a function of Bond number \( Bo \) as follows:

\[ C_k^2 = 3.2 [\tanh(Bo^{1/4}/2)]^2. \]

As it was presented above most of the experimental works on flooding CHF used annular or rectangular channels. Those researches are related to research reactor which most of them are using plats fuel element and are operating near atmospheric pressure. In nuclear power plants the most common fuel geometry is rods bundle and the operating pressure is near 15 MPa. However in case of LOCA it is possible that the pressure will decrease, a parameter which influences the researched flooding CHF phenomena. The present paper is a part of research which conducted to study that phenomenon in rod bundle near atmospheric pressure. The objective of the present study is to present preliminary results of CHF experiments in stagnant and low flow rate carried out with a triangle rode bundle with small value of \( L/D_{hy} \). The future steps of this research will conducted with more realistic values of \( L/D_{hy} \) related to fuel elements in nuclear power plants.

**EXPERIMENTAL EQUIPMENT AND PROCEDURE**

The test section was made up of three 12- mm OD, stainless steel tube which located inside a Pyrex tube 32 ID. Copper electrodes were silver soldered to each end of the stainless steel tubes. The length of the heated zone was taken for the preliminary experiments as 450 mm and the total channel length was 500 mm as shown in Figure 1(a). In the extension of this research, the heated length will change up to the power limitation of the power supplier. A thermocouple was attached to the outside surface of each heater. The location of two of the thermocouples was 50 mm upflow of its end near the exit zone and one thermocouple was
located at the middle of the third rod high. Those thermocouples were used to measure the wall temperature and to avoid overheating and burnout. The test section was heated electrically by a low voltage D.C power supply of 15 V and 5000 A. The temperatures of the water at the inlet and the exit of the test channel were also measured continually during the experiments by thermocouples.

![Diagram of test section and loop](image)

**Figure 1:** The test section (a) and the test loop (b)

The water in the water tank was heated to the required temperature and the flow-rate through the test section was set to the required value when the flow rate through the bypass was much higher then the flow rate through the test section. At those conditions power applied to the test section and increased at small increments. The occurrence of burnout was observed when the heater wall changed its color to red due to overheating in any location along the heaters. If the overheating of the heater is in the vicinity of the thermocouple a sharp increasing of the measured temperature was observed.

**RESULTS AND DISCUSSIONS**

Figure 4 shows the typical surface temperature variation of the heater rods. Those results are from experiment with zero flow rate. In the figure we can see also the variation of the electrical current which was supplied to the heater. In this experiment the current increased gradually up to the value in which boiling began inside the channel. In those conditions very intensive two phase flow (slug and annular flow) were observed inside the channel. The measured wall temperature was around the saturation temperature. That temperature didn't change when the heaters power continue to increased until the CHF was achieved. In figure 4 it can be seen that when CHF achieved the heaters temperature started to rise and a local overheating of the heater wall was observed by changing its color to red. On those conditions, to avoid damage to the heaters, the heating power turns off.
CONCLUSIONS

The CHF experiments have been carried out to study the behavior of the counter current and flooding boiling limitation in triangle rod bundle inside a round channel at an atmospheric pressure. The experiments were conducted using water at low flow rate of 0 to 100 kg/m²·sec and subcooling of 60 °C. Tracing of the inlet and exit water temperature of the channel was conducted to characterize the boiling zone along the channel. The present data are compared with Mishima et al. (1987) flooding CHF correlation. The following conclusions obtained from the present study.

a. Heater surface temperatures measurements show that when CHF was achieved overheating of the heaters was observed. The location of the overheating was changed in each experiment. It also observed that not all of the heaters were overheat at CHF conditions.

b. Measurements of the water temperature show large fluctuations of the inlet and exit temperatures during the power raising. From visual observation of the channel flow regimes it was observed an intensive slug /churn flow regime. The exit reservoir temperature was almost constant during the experiment.

c. The over heat of the heaters during the experiments after reaching boiling conditions was about 5 °C.

d. Mishima et al. correlation under estimate the experimental results in average value of about 30 %. At zero flowrate the underestimation was about 100 %.
Figure 5: Comparison between the experimental results and Mishima correlation.

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NOMENCLATURE
A  Flow area
AH  Heated area
CW  Constant in Wallis correlation for flooding
D  Tube diameter or characteristic length
D*  Dimensionless tube diameter (D*=D/λ)
G  Mass velocity
hfg  Evaporation heat of the liquid
Δh  Inlet enthalpy subcooling
q*  Dimensionless heat flux
qcf*  Dimensionless heat flux due to flooding
λ  Length scale of the Tailor wave
Rate Theory of Defects Induced by Ion Irradiation of Fe-12.5at%Cr Alloy

A. Gokhman¹, F. Bergner²

¹South Ukrainian National Pedagogical University, Staroportofrankovskaya 26, Odessa, Ukraine
²Helmholtz-Zentrum Dresden-Rossendorf, Bautzner Landstraße 400, Dresden, Germany

Corresponding Author: gokhman@paco.net

INTRODUCTION

Ion irradiation has proven useful and has been applied frequently in order to simulate neutron damage for the purpose of the investigation of basic irradiation effects(1). This application of ion irradiation is based on the insight that the damage caused by self-ions is largely equivalent to the damage produced by neutrons if the exposure measured in units of displacements per atom (dpa) is equal in both cases. This statement has been checked in the present study for the Fe-12.5at%Cr alloy. Ferritic-martensitic chromium alloys are candidate structural materials for future generation nuclear reactors such as fusion or advanced high temperature reactors (Gen IV) or spallation sources because of their remarkable resistance to swelling and of their adequate mechanical properties. The main focus will be put on the elucidation of the formation mechanisms of radiation defects on the nanometer scale. For that purpose the cluster dynamics (CD) is developed, which is calibrated by using transmission electron microscopy (TEM)(2,3) and small angle neutron scattering (SANS)(4-6). Solubility limit of Cr in Fe-Cr system is about 8.8at% at T = 573 K(3). Hence, deviation of the Cr precipitates ensemble from the ideal cluster gas(7) has been taken into account in CD modeling of Fe-12.5at%Cr alloy.

RESULTS

The irradiation conditions and experimental results(2-6) are summarized below for the purpose of comparison with the CD simulations performed in this study. The industrial purity Fe-12.5at%Cr alloy (average grain size 1 µm, pre-existing dislocation density 5.5x10¹³ m⁻² ) at temperature T of 573 K was neutron-irradiated in the Callisto ring (IPS2) in the Belgian reactor (BR2) and ion-irradiated with the 3 MV-Tandetron accelerator at the AIM facility of HZDR, Dresden, Germany. The neutron exposure covered the range from 0.6 to 1.5 dpa. The ion exposure was about 1.0 and 10 dpa. TEM investigations of neutron irradiated Fe-12.5at%Cr alloy(2) reveal the presence of dislocation loops of size, 2Ri, 6 nm and total number density of dislocation loops, Nᵢ, about 1.73x10²¹ m⁻³ for both irradiation conditions 0.6 and 1.5 dpa. TEM investigations of the ion irradiated Fe-12.5at%Cr alloy(3) reveal the dislocation loops with 2Rᵢ and Nᵢ about 3.5 (3.2) nm and 2.7x10²¹ m⁻³ (3.2 x 10²¹ m⁻³) for dose about 1 (10) dpa. Irradiation-induced features with radius of 0.5nm and volume fraction of 1.2 % for dose of 0.06 dpa and of 1 nm and volume fraction of 4.3 % for both doses of 0.6 and 1.5 dpa and A-ratio (ratio of total and nuclear SANS intensity) about 0 have been found by SANS for neutron-irradiated Fe-12.5at%Cr alloy(4-6). These features are related to pure Cr precipitates in □Fe as well as to □Fe atoms dispersed in the □Fe matrix. The CD model used in our study is close to the model(8), where the CD simulations are first performed for the free vacancies, self-interstitial atoms (SIA), vacancy clusters (VC) and dislocation loops and then for the precipitates taking into account the results obtained in the first step. In additional we consider the Cr-effect on the SIA diffusivity according
to the DFT calculations\(^{(9)}\). Correction of the attachment coefficient of Cr atoms to Cr precipitates has been done in our study according to the discussed method\(^{(10)}\) because relatively high chromium concentration. The frustration effect has been taken into account empirically by the use of the thermodynamic free energy expression from CALPHAD\(^{(11)}\) with the correction suggested by Bonny et al.\(^{(12)}\). The public domain library solver LSODA\(^{(13)}\) has been used to integrate directly the master equation of CD\(^{(14)}\) for the study of the matrix defect subsystem. Again the library solver LSODA has been used to integrate the master equations of CD\(^{(15)}\) in order to find the Cr-precipitates (\(\alpha'\) particles) size distribution, \(C_n\) where number of Cr atoms in precipitate, \(n\), up to \(N_{\text{max}} = 9000\).

From the comparison of CD results and both TEM and SANS data (Figures 1-3) it is possible to conclude that suggested rate theory model can be used to describe satisfactorily the kinetics of dislocation loops and \(\alpha'\) particles in ion irradiated Fe-12.5at\%Cr alloy. From the comparison of the data presented on Figures (1-3) and data for neutron irradiated Fe-12.5at\%Cr alloy\(^{(15)}\) the difference between ion and neutron irradiations can be found even for case of equal exposures measured in dpa:

- Size of dislocation loops, \(2R_i\), induced by ion irradiation significantly less than \(2R_i\) of dislocation loops induced by neutron irradiation.
- Saturation behaviour of dose dependence of the volume fraction of \(\alpha'\) particles in Fe-12.5at\%Cr alloy is observed from the small doses in the case of ion irradiation.

![Figure 1. Dose dependence of the total number density of dislocation loops, \(N_i\), for ion irradiated Fe-12.5at\%Cr alloy](image-url)
Figure 2. Dose dependence of the mean size of dislocation loops, $<2R>$, for ion irradiated Fe-12.5at%Cr alloy.

Figure 3. Dose dependence of the volume fraction of $\alpha'$ phase, $C_v$, and mean radius of $\alpha'$ particles in ion irradiated Fe-12.5at%Cr alloy.
CONCLUSIONS
1. Suggested rate theory model can be used to describe satisfactory the kinetics of dislocation loops and $\alpha'$ particles in ion irradiated Fe-12.5at%Cr alloy.
2. The difference between ion and neutron irradiations is observed even for case of equal exposures measured in dpa:
   - Size of dislocation loops, $2R_i$, induced by ion irradiation significantly less than $2R_i$ of dislocation loops induced by neutron irradiation.
   - Saturation behaviour of dose dependence of the volume fraction of $\alpha'$ particles in Fe-12.5at%Cr alloy is observed from the small doses in the case of ion irradiation.

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Study of Neutron Induced Defects in Ceramics with Positron Annihilation Technique using the GiPS Facility

S. May-Tal Beck\textsuperscript{1}, Maik Butterling\textsuperscript{2}, Wolfgang Anwand\textsuperscript{2}, Arie Beck\textsuperscript{1}, Andreas Wagner\textsuperscript{2}, Gerhard Brauer\textsuperscript{2}, Or Hen\textsuperscript{3}

\textsuperscript{1}Physics department, NRCN, P.O.Box 9001, Beer-Sheva, Israel
\textsuperscript{2}Institut für Strahlenphysik, Helmholtz-Zentrum Dresden-Rossendorf, 01314 Dresden, Germany
\textsuperscript{3}School of Physics and Astronomy, Tel-Aviv university, Tel-Aviv 69978, Israel

Corresponding Author: smtbeck@gmail.com

INTRODUCTION
There has been an increased interest in defects within structural materials motivated by future fission and fusion reactor needs. While reactor steels are extensively studied, much more research effort is needed in order to understand radiation damage in ceramic materials and its effect on the macroscopic characteristics of these materials\textsuperscript{(1)}.

Sapphire – the single crystal of Al\textsubscript{2}O\textsubscript{3}, presents a rather simple system where defects are trapped without grain boundaries, which might allow annealing procedures. The family of ceramics that contains Boron is another interesting group of materials for the nuclear industry, mainly due to high cross sections for thermal neutron capture in Boron, which produce helium inside the material. The much higher neutron flux expected in future reactors can cause swelling of materials and macroscopic cracks formation.

Helium is also considered to be one of the most interesting challenges for fusion reactors, due to alpha particles production in the D-T reaction. Materials that will be used as first wall, matrices for Li, or coating materials, will suffer from high radiation damage.

The sensitivity of Positron Annihilation Spectroscopy (PAS) methods to point defects as small as mono-vacancies, in concentrations as low as 1 appm, make them perfect tools to study radiation damage in its first stages of creation. Especially, Positron Annihilation Lifetime Spectroscopy (PALS) is sensitive to size and concentration of the point defects and Coincidence Doppler Broadening (CDB) can probe changes in defect characteristics as well as in electron momenta in the lattice.

We proposed to measure Sapphire and Boron Carbide (B\textsubscript{4}C) samples in the Gamma Induced Positron Spectroscopy (GiPS) facility at the HZDR\textsuperscript{(2)}. The GiPS is a gamma beam produced by bremsstrahlung processes from the electron primary beam of the ELBE accelerator. When hitting the sample under study, it produces positrons (by pair production) which are then detected. This beam is suitable for thick samples ($\geq 1\text{cm}^3$) of solids and liquids. Since the primary electron beam is pulsed, it is possible to measure the positron lifetimes in the material using stop pulses from the detectors at the measuring hall. Four pairs of detectors, each consists of a BaF\textsubscript{2} and a HPGe detectors allow to measure PAL, DB and to use the correlated information to measure also CDB and Age-Momentum Correlation (AMOC).

Preliminary results from a recent measurement of un-irradiated and neutron irradiated samples of Sapphire and B\textsubscript{4}C are presented.
PRELIMINARY RESULTS

We measured two Sapphire samples, each made of two disks of 12 mm in diameter and a total width of 6 mm. One sample was irradiated by neutrons some few years ago to a dose of \(6 \times 10^{18}\) n/cm\(^2\). The other was well annealed before measurements.

Two cubic samples of B\(_4\)C were also measured, with the total size of \(10 \times 10 \times 8\) mm\(^3\). Each of these samples consisted of two smaller samples, each of a size of \(10 \times 10 \times 4\) mm\(^3\). Two of these smaller samples were previously irradiated to a neutron dose of \(~10^{15}\) n/cm\(^2\). The neutrons were expected to create damage only on the surface (3), but since the density of B\(_4\)C is low, 2.52 g/cm\(^3\), and in order to get reasonable counting rate in the experiment, we measured two samples together. However, since all sides of the samples were exposed to the neutron flux, the ratio of surface defects to the bulk should be preserved. In order to get higher sensitivity to the surface, another two B\(_4\)C samples were measured with a total size of \(10 \times 10 \times 6\) (two samples, 3 mm wide).

The sum of annihilation events collected in each pair of detectors for each measurement varied between \(~0.5\) to 1 million events. The total number of events from each measurement summed up to more the 3 million events.

The BaF\(_2\) detectors are fast and therefore are capable of giving the stop signal to be used in positron lifetime measurements. The time resolution was measured using a dedicated Si measurement, to be \(~180\) ps. Annihilation events were selected as those in which the two photons emerged from the annihilation process had left 511 keV in the detectors (photo-peaks). The positron lifetime spectra were then analyzed using the PALSfit(4) program. Table 1 lists preliminary results of PAL analysis from all measurements.

Mean positron lifetimes in Sapphire show distinctive values, of \(~148\) ps and \(~185\) ps in the well annealed and irradiated samples, respectively. The spread of 4-5 ps in the results from the different pairs is much smaller than the \(~35\) ps difference seen between the two samples.

Mean positron lifetimes in B\(_4\)C were measured as \(~167\) ps, and did not vary for the different samples. The spread of 4-5 ps in the results from the different pairs of detectors is shown also here and reflects the systematic uncertainty of the results.
Table 1. Preliminary results of positron mean lifetimes in Sapphire and B\textsubscript{4}C samples. Listed are lifetimes extracted from each pair of BaF2 and HPGe detectors and the averaged mean lifetime from these 4 sets of data.

<table>
<thead>
<tr>
<th></th>
<th>Lifetime Pair 1 [ps]</th>
<th>Lifetime Pair 2 [ps]</th>
<th>Lifetime Pair 3 [ps]</th>
<th>Lifetime Pair 4 [ps]</th>
<th>Average [ps]</th>
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<tbody>
<tr>
<td>Sapphire</td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>Non Irradiated</td>
<td>149 ±0.2</td>
<td>149.5 ±0.2</td>
<td>150.4 ±0.2</td>
<td>146.1 ±0.1</td>
<td>148.8 ±0.2</td>
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<td>Annealed</td>
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<tr>
<td>Sapphire</td>
<td>184.7 ±0.2</td>
<td>185.3 ±0.2</td>
<td>186.5 ±0.3</td>
<td>181.4 ±0.2</td>
<td>184.5 ±0.2</td>
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<tr>
<td>Non Irradiated</td>
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<td>167.1 ±0.3</td>
<td>168.7 ±0.4</td>
<td>163.6 ±0.3</td>
<td>166.6 ±0.3</td>
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<tr>
<td>(width 8 mm)</td>
<td>166.8 ±0.3</td>
<td>167.1 ±0.2</td>
<td>168.1 ±0.3</td>
<td>164.3 ±0.2</td>
<td>166.6 ±0.3</td>
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<td>B\textsubscript{4}C</td>
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<tr>
<td>(width 6 mm)</td>
<td>168.1 ±0.3</td>
<td>167.2 ±0.2</td>
<td>169.6 ±0.3</td>
<td>164.7 ±0.2</td>
<td>167.4 ±0.3</td>
</tr>
</tbody>
</table>

CONCLUSIONS
The two distinctive mean lifetimes measured in the Sapphire samples can be attributed to the undamaged bulk region (~149 ps) and to vacancies in Sapphire (~185 ps), produced by the neutron irradiation. This last statement is based on previous measurements of Sapphire samples that had been irradiated to neutron doses in the region of 1-5×10\textsuperscript{18} n/cm\textsuperscript{2} and show the same mean positron lifetime\textsuperscript{(5)}, meaning a saturated trapping of positrons in the same type of defects. The measurements in the GiPS facility resulted in much more reliable lifetime values compared to the results reported in reference\textsuperscript{(5)}, mainly due to the large systematic uncertainty in the analysis procedure there.

The single mean positron lifetime value measured for the B\textsubscript{4}C samples, of ~167 ps, can be attributed to the undamaged bulk region. This conclusion is supported by the fact that the expected damage on the surface of the detected samples do not cause to any change in the mean lifetime. The insensitivity is caused probably by the integrative nature of the measurement with the gamma beam. More measurements with a slow positron beam are planned to check this hypothesis and to find if there has been damage created on the B\textsubscript{4}C surface and how deep the neutrons could penetrate the material.

REFERENCES
Chemically Deposited Thorium-Alloyed PbSe Thin Films: A Novel Technique for Self-Irradiation Damage Study

M. Shandalov1, M.Biton2, E. Yahel1, Y. Golan2, I. Kelson3

1Physic Department NRCN, POB 9001 Beer-Sheva, 84190 Israel
2Materials Engineering Department Ben-Gurion University, POB 613 Beer-Sheva, 84105 Israel
3School of Physics and Astronomy Tel Aviv University, Tel-Aviv, 69978 Israel

Corresponding Author: michash1234@yahoo.com

INTRODUCTION

In this work we present a new concept for investigating radiation damage in thin semiconductor films by using internal radiation sources rather than the conventionally used external radiation. Alloying the thin films with radioactive elements such as thorium is expected to provide a unique path for studying radiation damage in materials. As first part of this work, we developed a chemical bath deposition (CBD) process1-3 for alloying PbSe thin films (~200 nm) with the stable isotope (t1/2~106 years), 232Th. This was achieved by controlling deposition parameters such as temperature, reagent concentrations and time. Energy dispersive spectroscopy (EDS) confirmed the presence of Th in the PbSe films. EDS mapping in the analytical transmission electron microscope (TEM) indicated that the Th ions were homogeneously distributed throughout the film. In conclusion, a simple, controllable and cost-effective process for alloying of PbSe thin films with Th has been successfully developed. In the next stage of the research, substitution of the stable isotope of Th with radioactive 228Th should provide a novel route to radiation damage studies.

EXPERIMENTAL DESCRIPTION AND RESULTS

We developed a CBD process for alloying PbSe thin films with the stable isotope (t1/2~106 years), 232Th. This was achieved by controlling deposition parameters such as temperature, reagent concentrations and time. Variation of the deposition parameters such as temperature and reagent concentration allows control of the deposition reaction rates, solubility and convection and thus affects the structural and optical properties of the resulting films. PbSe(Th) films were deposited from a solution with a final composition of: 60mM Pb(CH3COO)2.3H2O, 50mM Na2SeSO3 , 0.6M KOH and 4.31mM Th(NO3)4. The solution was purged with N2 in order to minimize concentrations of dissolved O2 and CO2. Film growth was carried out at 20°C for various periods of time in order to obtain the desired thickness (Fig.1). The influence of the Th salt addition on film growth and resulting morphology was studied and compared to deposition of PbSe in the absence of Th. Reference PbSe films were deposited from solutions that were adjusted to the same pH as of the thorium containing solutions.
A. Microscopic Characterization

Scanning electron microscope images (SEM) of the (Th,Pb)Se films deposited for various periods of time indicating uniform, compact layers. Film thickness evaluated from cross sectional SEM images is plotted vs. deposition time, indicating linear dependence with growth rate of 95 nm/hour in the 4-12 hour region (Fig.2). Long incubation times may explain variability in growth rates for deposition time smaller than 4 hour. Below are SEM scans for different deposition times.

![Figure 2: Film thickness as a function of deposition time](image)

TEM micrographs of cross-section samples indicated a change from the nano-crystalline film morphology observed for non-alloyed CBD PbSe films—to columnar domain morphology [2]. Typical length of each domain is 85nm and the width is approximately 25nm. This change was attributed to growth mechanism alteration caused due to pH reduction following addition of the acidic Th salt solution.

![TEM cross section view of the film](image)
HRTEM images of the film and the substrate show a very thin amorphous region having varying thickness of 2-4 nm between the substrate and the film. This intermediate layer was also observed in previous reports\(^2\), where the chemical composition was found to be Ga\(_2\)O\(_3\). TEM did not show any evidence for the presence of an amorphous phase besides for this ultra thin amorphous interfacial layer.

**HRTEM images**

B. Composition Analysis
EDS mapping in the analytical TEM confirmed the presence of Th ions in the films, and indicated that they were homogeneously distributed throughout the film. This homogeneous distribution of Th ions could suggest possible location of these ions as substitutional atoms in the rock salt structure (replacing the lead ions). Among the three known compounds of
thorium selenide, ThSe, ThSe₂ and ThSe₃, the former is the only one to crystallize in the rock salt structure and that has the same stoichiometry as PbSe.

Table 1. Average (±0.1at%) values evaluated from EDS maps:

<table>
<thead>
<tr>
<th>Se-K (at%)</th>
<th>Pb-M (at%)</th>
<th>Th-M (at%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>48.7</td>
<td>51.1</td>
<td>0.2</td>
</tr>
</tbody>
</table>

CONCLUSIONS

In conclusion, a simple, controllable and cost-effective process for uniformly alloying of PbSe thin films with Th was successfully developed, providing a route to radiation damage studies in lead chalcogenide thin films.

REFERENCES

Pyrolysis and Gasification of Solid, Organic, Low-Level Nuclear Waste: HDPE as a Case Study

Y. Peled\textsuperscript{1}, O. Klein BenDavid\textsuperscript{1}, Z. Ohaion\textsuperscript{1}, D. Tavor\textsuperscript{2}, G. Bar-Nes\textsuperscript{1}

\textsuperscript{1}NRCN, 9001 PO Box, Beer-Sheva, Israel
\textsuperscript{2}Sami Shamoon College of Engineering, Basel/Bialik Sts, Beer-Sheva, Israel

Corresponding Author: yaelpe@nrcn.org.il

INTRODUCTION
As environmental concerns have become a high priority matter, treatment of waste and especially radioactive waste are considered as a big global issue. The Israeli national radioactive waste disposal site is situated at the Nuclear Research Center Negev (NRCN). Radioactive waste from hospitals, research institutions and industry is delivered to the site. Pyrolysis is the thermochemical decomposition process of organic materials in the absence of oxygen to produce char, pyrolysis oil and gas. Gasification is defined as the thermal, non combustion, decomposition of organic material with a limited amount of oxygen and/or steam. Both processes are considered today as advanced thermal treatment technologies leading to mass and volume reduction of radioactive/hazardous organic waste, while retaining the radionuclides and inorganic material in solid form to be further disposed.

In the present study, the thermal decomposition of High Density Poly-Ethylene (HDPE) was investigated combining two experimental systems. Thermo-Gravimetric Analysis (TGA) was used to measure the maximal decomposition rates at different pyrolytic experimental conditions and obtain the activation energy of the decomposition reaction. Product distribution and the extent of conversion to gas under pyrolytic and partial oxidating conditions (gasification) have been obtained from experiments performed in the pyrolysis/gasification system. Gas analysis was performed using Gas Chromatography (GC).

RESULTS
The experimental results obtained from TGA measurements (Mettler-Toledo TGA/SDTA851e) at temperatures of 420-480\textdegree{}C with nitrogen flow rates of 20-170 ml/min are presented in Fig.1. HDPE decomposition rates were found to be highly affected by the reaction temperature, whereas the effect of the nitrogen flow rate on the decomposition process was negligible. Statistic analysis using the Analysis of Variance approach show there is no mutual influence between the two parameters. The activation energy was calculated as 274 Kcal/mol, in accordance with data in the literature\textsuperscript{(1)}.

The product distribution obtained from experiments performed in the pyrolysis system is presented in Fig.2. Gas production has increased from 22\% to 63\% upon pyrolysis temperature elevation from 450\textdegree{}C to 525\textdegree{}C. Further increase in the temperature to 600\textdegree{}C has not changed significantly the product distribution.
Figure 1. TGA/DTG results for HDPE subjected to: a. Different isothermal conditions; b. Different nitrogen flow rates.

In order to increase the fraction of gaseous products, a series of gasification experiments was performed using 3%, 7% and 12% of oxygen together with nitrogen. No significant increase in
the gas production was found under gasification conditions compared to the pyrolysis conditions (100% nitrogen).

Gas analysis was performed with a HP-5920 GC, using a 60/80 molecular sieve 5A column and a thermal conductivity detector (TCD). The methane (CH\textsubscript{4}) and hydrogen (H\textsubscript{2}) concentrations during the pyrolysis/gasification reactions at 525\textdegree C are presented in Fig. 3. Gas release begins after approximately 25 minutes when the temperature in the pyrolysis/gasification reactor reaches the value of 500\textdegree C. The addition of oxygen in the gasification experiments has no apparent effect on the methane and hydrogen evolution. However, the oxidation effect was exhibited at the end of the thermal decomposition, leading to a reduction in the residual ash together with increased CO\textsubscript{2} gas evolution. This finding can be attributed to non efficient mass transfer within the experimental system.

![Figure 2. Product distribution vs. temperature of the pyrolysis reaction](image-url)
CONCLUSIONS
The combination of two experimental systems: TGA measurements together with pyrolysis/gasification results, allows gaining better understanding concerning the optimal HDPE decomposition conditions, thermodynamic characteristics and product distribution.

REFERENCES
Immobilization of Low Level Wastes in Geopolymeric Systems

E. Ofer-Rozovsky¹, G. Bar-Nes², M. Arbel-Haddad², A. Katz¹

¹Faculty of Civil and Environmental Engineering, Technion I.I.T., Haifa, 32000, Israel
²NRCN, PO Box 9001, Beer-Sheva, Israel

Corresponding Author: elyaofe@technion.ac.il

INTRODUCTION
In recent years, there is a world-wide interest in finding a solution for the accumulating nuclear wastes. Low activity waste streams are highly alkaline, rich in sodium and nitrate ions, and contain relatively low concentration of radioactive ions, typically cesium and strontium. The selected solidification matrix for such waste streams must therefore bind and retain the radioactive ions, and at the same time must be compatible with the remaining ionic components.

The most commonly used methods for conditioning and immobilizing low-activity radioactive waste are the high-costing vitrification[1] and solidification in Portland Cement (OPC) blended with pozzolanic additives[2, 3], which is more moderately priced. Several authors have found that alkali activation of pozzolanic materials such as fly-ash[4], slag[5] and metakaolin[6] without OPC leads to the formation of strong, low porosity aluminosilicate structures. These structures, named by Davidovitz "geopolymers"[7], frequently have high mechanical strength, flexibility, durability and low liquid permeability. Geopolymers may be amorphous or partially crystalline, depending on their composition and curing conditions. Cations are incorporated into the geopolymer structure as counter ions to the negatively charged aluminate groups. Due to all of these properties, geopolymers are considered as an alternative immobilizing matrix for low-level radioactive wastes. For example, zeolite A, a potential crystalline product of the geopolymeric matrix, was used to adsorb hazardous species in waters around Fukushima site after the accident in 2011.

The aim of the present study is to investigate the effects of the main ions present in the low activity waste, i.e. sodium and nitrate, on the formation and evolution of geopolymeric systems at engineering implemented conditions, i.e. at 40°C. The results presented in this paper relate to geopolymeric matrices prepared from metakaolin, a product obtained through calcination of kaolin, which is often used as a model pozzolanic material.

RESULTS
Metakaolin (MK) was alkali-activated using NaOH solutions of varying concentrations (2M, 4M, 6M, 10M). The amount of solution and MK was adjusted in order to keep the \( \text{Na}^+:(\text{AlO}_4)^- \) ratio at 1:1, the full stoichiometric demand for balancing all the aluminates in the raw material. For the alkali concentrations of 4M and 10M two additional \( \text{Na}^+:(\text{AlO}_4)^- \) ratios were studied: 1.2:1 and 0.8:1. The nitrate was added as NaNO₃ salt in relative concentration to the \([\text{OH}^-]\) ions, i.e. \([\text{NO}_3^-]:[\text{OH}^-]=0.25\). At 6M and 10M NaOH, higher nitrate concentrations of 2.5M and 3.3M, respectively, were applied as well. The samples were cured in sealed containers at 40°C for varying time periods (between one day and three months). The geopolymerization products were examined by X-Ray diffractometry, Fourier Transform Mid-Infrared spectroscopy (FTIR), and scanning electron microscopy (SEM).
**Geopolymerization in absence of nitrate: a reference study**

Reference geopolymeric matrices excluding sodium nitrate were prepared. The geopolymeric product included both an amorphous phase and a zeolitic phase identified as LTA (zeolite A). A second zeolitic phase, identified as faujasite (zeolite X), was detected in addition to the LTA phase after a three day curing period for samples activated by 10M NaOH. The same phase was detected at later stages after one month and three months in samples activated by at 6M and 4M NaOH, respectively. The available data does not allow us to determine whether zeolite X was generated from zeolite A, or formulated together with it. As both zeolites are composed from very similar building units (sodalite unit) it is possible that zeolite X may be the next product in the sequence, leading from the kinetically to the thermodynamically preferred product as shown in figure 1. Since zeolite X has more open cages and cavities than zeolite A, we suggest its stability is due to the expansion of zeolite A to a structure with less steric hindrance. On the other hand, these two zeolites are not considered as competitive crystals in the literature [8]. Further study on the issue should be conducted.

![Diagram](https://via.placeholder.com/150)

**Figure 1.** Products development sequence in geopolymeric matrix in the absence of nitrate.

Zeolite X was not detected in geopolymerization products obtained using Na⁺:AlO₄⁻ =0.8. Under these conditions, for both OH⁻ concentrations (4M and 10M), only LTA was identified along with the dominant amorphous phase. These findings indicate that zeolite X generation requires a number of conditions: long curing time, a Na:Al ratio of 1:1 or higher, and efficient dissolution of the amorphous raw material.

**Geopolymerization in presence of nitrate**

Alkali-activation of MK at an OH⁻ concentration of 10M in presence of NaNO₃ led to immediate formation of a nitrate-containing crystalline phase, nitrate sodalite. Activation at lower alkali concentrations yielded products containing zeolite A, as in the nitrate-free samples. During the curing process, zeolite A transformed into nitrate sodalite which further transformed to nitrate cancrinite. The suggested phase transformation sequence is shown in figure 2. The typical microstructure of nitrate cancrinite was identified in SEM images of MK geopolymers. IR spectra demonstrate the nitrate incorporation in nitrate cancrinite as a function of nitrate concentration.

This reaction sequence from the kinetically favored product (the amorphous gel) to the thermodynamically stable product (cancrinite) is well known for carbonate systems [10], and was suggested as a hypothesis in nitrate systems [12,11]. Due to relatively low temperature in this study (40°C) and long curing periods we succeeded to identify this sequence in MK geopolymeric systems.

The reaction sequence was interrupted when Na⁺:AlO₄⁻ was lower than 1, as was demonstrated in geopolymerization process in the absence of nitrate.

In general, faster crystallization and higher crystalline content were found in the presence of nitrate in comparison to systems without nitrate. A possible explanation to this acceleration may be the higher hydration energy of nitrate relative to silicates and aluminates [9], allowing easier connection of the silicates and aluminates to geopolymers.
Another possible reason might be the higher ionic strength which shifts the equilibrium towards the thermodynamic favored products.

CONCLUSIONS
Sodium and nitrate ions are often present at high concentration in nuclear wastes, and therefore their influence on the immobilizing media must be investigated. The present study demonstrates the influence of composition and curing times on the mineralogy of the geopolymeric product. Various zeolitic phases which have a good immobilizing potential for radionuclides were found among the reaction products (zeolite A, zeolite X, nitrate sodalite and nitrate cancrinite). The reaction sequence in the geopolymeric system goes from the kinetically preferred product, the amorphous gel, to the thermodynamically preferred products. Further investigation of the interaction between radionuclides and the above crystalline phases obtained in geopolymeric matrix will be conducted. In addition, the use of other materials composed of amorphous silica and alumina, e.g. fly-ash, as raw materials for geopolymerization will be examined.

REFERENCES


INTRODUCTION
The complex 3D shapes of IMRT dose distributions and the methods for IMRT dose delivery demand that the dosimetry measurement techniques be reviewed. Understanding the limitations and use of dosimeters to measure these dose distributions is critical to safe IMRT implementation. For point-dose measurements of total delivered dose, small volume cylindrical ionization chambers are generally used because of their excellent stability, linear response to absorbed dose, beam-quality response independence, traceability to a primary calibration standard, and the importance for high-spatial resolution. These chambers have cylindrical symmetry, but in case the dose distributions to be checked include noncoplanar beams, no data is currently available about oblique-incidence response of small cylindrical ionization chambers to megavoltage beams. The two cylindrical chambers most commonly used for IMRT dosimetry at Hadassah were checked for oblique-incidence for low and high energy beams: the angular dependence was first measured for each chamber for the relevant couch angular range, and then assessed by Monte-Carlo calculations.

MATERIAL AND METHODS
A cylindrical water phantom (15cm diameter) was designed and set up on the couch of a Varian, Clinac 2100, linear accelerator. The two cylindrical ionization chambers: PTW PinPoint chamber type 31006 and the IBA Compact Chamber CC01 were inserted in the center of the phantom, and measurements were performed for oblique incidence every 10 degrees, for both the 6MV and 18MV photon beams. These measurements were then compared to Monte-Carlo calculations on EGS5.

RESULTS
The IBA Compact Chamber CC01 demonstrated a significant angular dependence, up to 2.4%, where the PTW PinPoint chamber showed a less significant angular dependence, maximum of only 0.6%. In both cases the angular dependence is more pronounced for the 6MV beam than for the 18MV beam see figure 1.
Figure 1: Measured angular dependence for PTW PinPoint chamber type 31006 and IBA Compact Chamber CC01 using high energy (18MV) and low energy (6MV) megavoltage beams.

CONCLUSION
Care should be taken, in case a dose distribution verification measurement includes noncoplanar beams, to take into account the sensitivity of the chamber for such geometries and eventually correct the measurement by an angular factor experimentally determined.
TrueBeam Commissioning: The Beilinson Experience

A. Khatib Hamad1, J. Menhel1, Y. Krutman1, D. Bragilovski1

1Beilinson Hospital, Rabin Medical Center, Petach Tikvah, Israel

Corresponding Author: aminehha@clalit.org.il

INTRODUCTION

Two linear accelerators of the type True Beam (Varian Medical Systems, Palo Alto, Ca, USA, VMS) were purchased at the Rabin Medical Center within the last two years. One is already in clinical use for a year and the second linac is about to be released for clinical use. The energies available on both linacs are 5 photon energies and five electron energies. Among the photon energies, two are with an unflattened beam quality 6X FFF and 10X FFF.

The work presented will describe the commissioning procedures, the equipment used and the results obtained investigating this relatively new linear accelerator.

RESULTS

The IBA(Belgium) Blue Water Phantom was used with the CC01, CC04, and CC13 chambers for the acquisition of the beam scanning data. Choice of the chamber used depends on the quality tested. For scanning parameters, two CC13 were used, one for field measurements and the other as a reference chamber in order to account for dose rate variations. CC04 was used for very small fields scanning data.

The field sizes investigated were chosen according to the Beam Data Acquisition manual supplied by the company. For each field listed measurements of PDD's, Profiles and Output Factors were obtained. Each scan was performed three times and after averaging and smoothing the final curve was saved as the final data to be used in beam configuration. Non scanned data was obtained as an average of three measurements for statistical variations.

A sample of the profiles measured for 6X-FFF at all depths required is shown in Fig.1.

![Figure 1 Cross plane Profiles for 6XFFF Field Size 10x10](image-url)

For the configuration of dynamic techniques of beam delivery, Dynamic leaf Gap is required. This data is crucial for safe delivery in IMRT and Rapid Arc modes. Dynamic Leaf gap and MLC transmission were measured for each beam energy. Portal Dosimetry was commissioned in order
to be used for patient specific IMRT QA. All chambers used clinically were commissioned according to the TRS398, for routine photon and electron beam measurements as relevant. Verification plans for the current working protocols were prepared and measured both with ion chamber, Film and Portal Dosimetry.

CONCLUSIONS

The results of measurements obtained were compared with results from other institutions abroad and with our own existing linear accelerators. The verification plans deviations were consistent and within the expected range of 1-2%. Plans measurements at the CAX were within 1%. Plans verified with film and portal dosimetry were within 3% using a gamma of 3% 3mm.

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Biological Responses to Reactive Oxygen Species Forecast the Need for Additional Microdosimetric Considerations in Radiotherapy: To Be or Not to Be

Brenda Laster¹, Liat Barad², Rachel Bar-Deroma³

¹Jerry J. Cohen Radiobiology Laboratory, Department of Nuclear Engineering
²Department of Biochemistry, Ben Gurion University Beer Sheva 84105 Israel
³Department of Oncology, Rambam Medical Center, Haifa 31096 Israel

Corresponding Author: blaster@bgu.ac.il

ABSTRACT
Cells respond to radiation by engaging the mechanism of autophagy in which damaged organelles or molecules are sequestered in a separate compartment, the autophagosome. This activity enables time for the activation of salvaging repair mechanisms, which, if successful, represent reparative autophagocytosis and restoration of homeostasis. However, sequestration of damaged mitochondria, replete with H₂O₂ and iron can also cause a Fenton reaction whose effect is lysosomal rupture, upon fusion of the autophagosome with the lysosome, due to hydroxyl radical (•OH) formation. In some cases, •OH radicals cause the loss of cell membrane integrity and the spillage of iron and H₂O₂ into the extracellular environment. Additional H₂O₂, secreted by inflammatory cells, that may also be regionally present, further augment •OH formation. Here we show autophagy, induced by graded H₂O₂ concentrations, its effect on the metabolic activity of the cell and its generation of apoptosis and necrosis over time. Our results suggest that the bystander effect, observed in non-targeted cells, may be a consequence of the Fenton reaction in the latter stages of macroautophagy.

INTRODUCTION
The formation of highly energetic and damaging reactive oxygen species (ROS) during the interaction of ionizing photons with the 70-80% body water content (radiolysis) contributes to cellular damage to an even greater extent than that produced by the direct action of photons impinging on cellular DNA [1]. This report will limit its scope to two members of ROS, hydroxyl radicals (•OH) and superoxide anions (•O₂⁻) and their influence on biological mechanisms that affect the life or death destiny of cells and body tissues as a consequence of radiation- or H₂O₂-induced macroautophagy. The highly reactive •OH has a half-life of 10⁻¹⁹ seconds and, upon interaction with macromolecules, can, among other effects, cause mutations in DNA, lipid peroxidation, and alter protein structures. Superoxide anions (•O₂⁻) are efficiently neutralized by the body’s ubiquitous superoxide dismutase enzymes (SODs) that convert •O₂⁻ to hydrogen peroxide (H₂O₂) and molecular (O₂), thereby conferring to H₂O₂ the status of an ROS derivative in the body. Unlike •O₂⁻ and H₂O₂, there are no physiological agents, such as catalases, peroxidases among others, that detoxify •OH. The production of both •OH and •O₂⁻ in the body after exposure to ionizing radiation (IR) is largely a function of the radiation dose delivered [2-5].

H₂O₂ induces macroautophagy [6, 7], an intracellular salvaging process utilized by the damaged or dying cell to restore itself. The failure of reparative autophagy leads to the death of the cell. But autophagic cell death, in and of itself, is a double-edged sword. The same process that gives
the challenged cell an opportunity to exploit salvaging mechanisms, may also contribute to the death of neighboring cells within the vicinity of the cell undergoing autophagy. The sequestration of large quantities of iron into the autophagosome, which merges with the redox-active lysosome, sustains the iron in a reduced state due to its acidic pH and high concentration of thiols [8-22]. In its reduced state, the iron is available for interacting with H$_2$O$_2$ in a Fenton reaction that results in the production of OH$^\cdot$, rupturing the lysosome as well as the cell membrane. *In vitro*, the leakage of iron and H$_2$O$_2$ into the extracellular compartment and the formation of OH$^\cdot$ radicals can damage previously unaffected cells, such as those that may have recently divided or those that had repaired the damage. We suggest that this phenomenon may explain the bystander effect, where non-targeted cells, unexposed to ionizing radiation (IR) but that neighbor irradiated cells, undergo apoptosis [23-42].

In the *in vivo* situation, this situation can be even more severe. The local concentration of H$_2$O$_2$ produced within a radiation treatment volume is not only dependent upon the radiation dose or number of O$_2$ ion radicals produced and converted to H$_2$O$_2$ by SODs, but also depends upon the number of inflammatory cell recruited to the targeted area to clean up cellular damage and debris. Through their own respiratory burst of O$_2$ and the ubiquitous presence of SOD enzymes, inflammatory cell responders increase the local concentration of H$_2$O$_2$ and its availability for interacting with iron. The resultant OH$^\cdot$ radicals can then damage surrounding normal cells outside the treatment volume.

**MATERIALS AND METHODS**

We conducted this three-part experiment on L428 Hodgkins Lymphoma cells, removed from the same treatment vessels (50 mm Petri dishes), which had been seeded with exponentially-growing cells at a concentration of 3·10$^5$ cells/ ml then incubated with H$_2$O$_2$ for 24h. We assayed one aliquot of cells immediately after the 24h incubation, replaced the H$_2$O$_2$- containing medium with H$_2$O$_2$-free medium, continued to incubate the cultures, and assayed additional aliquots from the cultures at 48 and 72h after the start of treatment (i.e., 24 and 48 hrs after the end of treatment). To determine metabolic activity (MA), the XTT assay was performed on 100 μl of the cell suspension. For the determination of apoptosis and necrosis, acridine orange/ethidium bromide (AO/EB) stains were applied to a 10 μl aliquot of the cell suspension that had been placed on glass slides. The numbers of cells in early and late apoptosis and necrosis were counted based upon the staining and cellular morphology using a fluorescent microscope [43]. For autophagy, the monodansylcadaverine (MDC) stain to identify autophagosomes [44] was mixed with 10 μl of the cell suspension and placed on slides as described above. Results are shown in Figs. 1 and 2. The percentage of apoptotic and necrotic cells are shown in Table 1.

**RESULTS AND DISCUSSION**

The panels in Fig.1 (left) show autophagy in cells immediately after the 24h incubation with the various H$_2$O$_2$ concentrations. At concentrations below 50 μM, fewer autophagic cells were observed. Of major interest are the amorphous bodies of unknown origin,
incorporated within an opaque broader mass of material that appears to encompass the entire cell and seen predominantly in autophagic cells at the 50 μM concentration. We interpret this opaque area as indicative of lysosomal rupture and the release of its contents throughout the cell [13]. (We intentionally increased the contrast in the 50 μM panel with the hope that, by increasing the contrast, the reader could see this phenomenon in the reproduction of this image.)

The amorphous bodies, depicted by single arrows, apparently consist of material that was excited at the same wavelength used to excite the MDC stain that reveals the autophagosome (350 nm). Although it is possible that these may be artifacts, it is also possible that they represent the iron-containing protein, ferredoxin, which is present in the mitochondrial matrix and shows 75% absorption at 350 nm [45]. Damaged or defective mitochondria are replete with ferredoxin and ROS [46] are incorporated into the autophagosome [13]. The jagged arrow in the panels points to leakage from the cell into the external environment. As shown in Table 1, cells incubated with higher concentrations of H2O2 (80 and 100 μM) displayed higher levels of apoptosis at the 48 and 72 h time intervals. In light of our demonstration of autophagy and lysosomal leakage, it would be reasonable to suggest that the Fenton reaction may have caused some of this additional damage. At concentrations of 80 and 100 μM, an increase in the number of apoptotic and necrotic cells is observed at 24h compared to the lower H2O2 concentrations, supporting Loos’ hypothesis that increasing the severity of a cellular insult is likely to decrease the time between the induction of cell death and its execution so that the Point Of No Return (PONR) is reached earlier.

Although we see an immediate increase in apoptosis and necrosis at these concentrations at 24h, we do not see a decrease in the cell’s metabolic activity (Fig. 2), which suggests that not all cells within the population are similarly affected by the insult. It seems probable that those cells with an insufficient energy capacity to react to the insult and engage autophagic mechanisms underwent apoptosis and necrosis during the 24h incubation with H2O2. Since the insult was removed after 24h giving surviving cells an opportunity to engage salvaging mechanisms, it would be the 48 and 72h time points that would be most useful in evaluating the dynamics of autophagy on dying cells.

Table 1 (below) shows the impact of the varying concentrations of H2O2 on apoptotic and necrotic cell death as a function of time. At concentrations of 50 μM and greater, cell death increases with concentration over time. The higher levels of necrosis observed with 0 μM and 10 μM at the 72h point is presumed to be caused by medium depletion as a result of their rapid growth rate. That is, regulatory mechanisms were activated during autophagy to salvage the ‘dying’ cells and were successful [9]. The reduced number of 80 and 100 μM H2O2-treated cells available for counting on the slides is presumed to be a function of cell death and is consistent with the measured reduction of metabolic activity (indicative of dead or dying cells) shown in Fig. 3 below. Apoptosis and necrosis resulting from H2O2 concentrations of 80 and 100 μM are significantly greater than that observed at any other concentration. However, the apoptotic and necrotic condition is only reflective of those cells whose engagement of autophagy was unsuccessful. At all concentrations, reductions in MA are maximal at the 48 hr time point, but by 72h an increase in MA is observed (Fig. 2). This suggests that the mechanism of autophagy might take even longer than 72h before the ultimate destiny of the ‘dying’ cell is determined and that the increased MA is a function of the cell’s activation of regulatory mechanisms.
Table 1. Apoptotic and necrotic changes in cells incubated for 24 h with graded concentrations of H$_2$O$_2$.

<table>
<thead>
<tr>
<th>H$_2$O$_2$ conc. (μM)</th>
<th>Time, (h)</th>
<th>Cells counted</th>
<th>% apoptosis</th>
<th>% necrosis</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>24</td>
<td>99</td>
<td>0</td>
<td>1</td>
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<td>10</td>
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<td>125</td>
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<td>1.6</td>
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<td>11.2</td>
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<td>106</td>
<td>13.2</td>
<td>5.6</td>
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<td>65</td>
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</tr>
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<td>100</td>
<td>72</td>
<td>31</td>
<td>67.7</td>
<td>25.8</td>
</tr>
</tbody>
</table>

Measurements of metabolic activity in cells, over time, are shown in Figure 3 (below). Immediately after the 24h incubation with H$_2$O$_2$, the MA was stable at all concentrations. An increase in MA at the 10 and 25 μM concentrations is consistent with the cell’s response to detected damage probably through its activation of the pro-survival DNA transcription factor, NF-κB [47]. Little change in MA is seen at 25 and 50 μM; there is only a ~20% variation in metabolic activity at all time intervals, possibly reflecting reparative autophagocytosis within the measured time intervals.

Our findings suggest that the process of autophagy is a dynamic response that enables the restoration of homeostasis to damaged or dying cells until they reach a PONR and die. Although our experiments evaluated the impact of H$_2$O$_2$ on autophagy only up to 72h, it appears that the kinetics of salvaging mechanisms continue to be operable up to this point and most likely beyond. Lee et al showed that astrocytic cell death induced by 200 μM H$_2$O$_2$ was accompanied by lysosomal membrane permeability (LMP) [48]. LMP, after an IR dose of 40 Gy was attributed by Persson et al to an increase of intralysosomal iron that was associated either with 'reparative autophagocytosis' or that sensitized cells to lysosomal rupture and consequent apoptotic/necrotic death following a second fraction of 20 Gy [13]. We suggest that this leakage not only represents the PONR as the cell membrane loses its integrity, but that the spilled contents would also impact on regional cells, particularly if inflammatory cells were present in the region of damage. The released lysosomal iron and H$_2$O$_2$ from damaged mitochondria will interact in a Fenton reaction producing 'OH and damaging neighboring cells within the tissue, a possible explanation for the bystander effect [23, 25, 27-30, 34, 38, 42]. Our findings suggest the importance of developing microdosimetric models, such as that detailed in [41], while including 'OH radical formation after lysosomal rupture, as a means of evaluating the risk to normal tissues in patients undergoing radiotherapy as detailed in [49]. However, we also point out that leakage from the cytoplasmic membrane causes the leakage of soluble factors into the extracellular...
environment. These factors signal adaptive immune T cell responses [50]. Whereas even low concentrations of H$_2$O$_2$ are associated with cell membrane rupture, it is reasonable to conclude that immune signals from the released factors could predominate over any damage from the Fenton reaction. This may be an explanation for the reported health benefits from low dose radiation [51-55] because it produces lower endogenous levels of H$_2$O$_2$, as well as the relationship between bystander effects and adaptive responses [24, 28, 31].

Acknowledgements: These studies were supported by the Jerry J. Cohen Research Foundation.

REFERENCES


Comparative Prospective Study - Exposure to Radiation during URS vs. ESWL

S. Haruz-Waschitz\textsuperscript{1}, Dr. A Ben-Shlomo\textsuperscript{2}, Dr. Y Siegel\textsuperscript{3}, Dr. A Cooper\textsuperscript{3}, Dr. Y Shilo\textsuperscript{3}

\textsuperscript{1} Assaf Harfek Medical Center Safety and Health Section, Zerifin 70300, Israel
\textsuperscript{2} Soreq NRC; Yavne
\textsuperscript{3} Assaf Harofeh Medical Center, Urology Dep. Zerifin 70300, Israel

INTRODUCTION

The most common stone therapies; Extracorporeal Shock Wave Lithotripsy (ESWL), Ureteroscopy (URS) and Percutaneous Nephrolithotomy (PCNL), are performed under fluoroscopy guidance. Most of the clinical outcome is assessed with x-ray imaging.

ESWL X-ray image is used to target the stones. Once the stone is targeted, a special unit emits shock waves focused precisely on the stone. These waves produce enough force to cause stone fragmentation, while avoiding injury to adjacent organs and tissues. The stone and resulting pieces are fragmented until they can pass through the ureter, into the bladder, and subsequently exit the body.

URS is an alternative to ESWL to remove stones endoscopically from the urinary tract. URS is performed under general anesthesia and involves the placement of a miniaturized fiber optic scope under direct vision are introduced through the nature orifices or through percutaneous access to the stone. A small laser fiber or special ultrasound electrodes are passed through the ureteroscope to break up stones. Other instruments may be passed to capture and retrieve the fragments from the urinary tract. URS may be used to remove stones in all portions of the urinary tract and may be used to retrieve multiple stones during the same procedure.

X-rays (up to 100 kV) have been used to diagnose diseases in the kidney and urinary tract, for about a century, to visualize urinary tract and to highlight a kidney stone or tumor that could block the flow of urine.

Due to the increased use of radiation during urological procedures, protection of patients from ionizing radiation is becoming increasingly important. With such usage there is a need for adopting dose management techniques in every radiological examination without compromising the image quality and clinical purpose.

There is a possibility of staff getting high exposure, in the fluoroscopy room, if protection principles and tools are not employed.

In recent years there is an increase in awareness to radiation exposure in Endo-Urology. Patients with urinary tract stones are exposed to radiation as part of diagnostic and therapeutic procedures.

PURPOSE
The purpose of this study was to perform a comparison of the radiation dose exposure of patients who underwent URS procedure vs. ESWL.

METHODS

During 2010-2011 a prospective study was carried out in 106 adult patients who underwent URS or ESWL: 47 (44.4%) and 59 (55.6%) respectively. Demographic and medical data, including: age, sex, body mass index (BMI), size and location of stones were collected. Measurements of dose–area product (DAP) were performed. For ESWL an exterior DAP meter was assembled to the C-Arm. The C-Arm used for URS includes build-in DAP meter. Coefficient factor between the devices was calculated.

RESULTS

Patient's urological data and the dose area product (DAP) are given in table 1.

<table>
<thead>
<tr>
<th>Properties</th>
<th>URS</th>
<th>ESWL</th>
</tr>
</thead>
<tbody>
<tr>
<td>Avg. Age (years)</td>
<td>52.8 (20 – 81)</td>
<td>54.3 (24 – 81)</td>
</tr>
<tr>
<td>BMI (Avg±std.)</td>
<td>28.71±5.37</td>
<td>26.98±4.36</td>
</tr>
<tr>
<td>Avg. stone size (mm)</td>
<td>11.3</td>
<td>12.5</td>
</tr>
<tr>
<td>DAP [cGy*cm²] (Avg±std.)</td>
<td>240.25±180.19</td>
<td>90.7±48.60</td>
</tr>
</tbody>
</table>

Table 1: The patient's urological data and the dose area product (DAP) of their procedure.

CONCLUSION

No major statistical valid differences of the demographic data were observed. In this work we found that patients undergoing URS are exposed to higher radiation doses compared with those undergoing ESWL. This information should be taken into consideration while choosing the medical treatment.
Hospital Preparedness for a Radiological Terrorist Event

R. Bar-Deroma, L. Utitz, K. Moskovitz-Shir

Rambam Health Care Campus

Corresponding author: r_bar-deroma@rambam.health.gov.il

PURPOSE

Radiological incidents present uniquely challenging scenarios for hospital emergency planning and response. Radiological terrorism has been recognized as a probable scenario with high impact and hospitals are being educated to deal with this threat. The purpose of this paper is to present the hospital's experience preparing staff and infrastructure for response to a Radiological Dispersal Device (RDD) explosion.

METHODS

Rambam Health Care Campus has been designated to treat casualties from RDD explosions. Even for an experienced healthcare provider that knows how to respond to mass casualties from terrorist events, the fact that radioactive materials are involved impose changes in the management of the patients. In order to deal with this challenge the national authorities organize drills to simulate the response to RDD events. Such an exercise has been organized for January 18, 2012. During the last 9 months, several sessions were coordinated to train the staff, including nurses, physicians, medical physicists, radiation technologists, radiation protections officers, and clerical staff. The trainers' task was to educate the hospital caregivers on the different aspects of radiation protection, decontamination procedures, operation of radiation monitors and on the behaviour in the suspected contaminated areas. A major task was to explain that radioactive exposure to the staff in such scenario is minimal and if the correct protection measures are observed, no health threat is expected. Staff was instructed on how to wear the protective suit and also how to undress it in the decontamination station. Special attention was given during the different training session to the fact that injuries caused by the explosion are the ones to be treated first since they will determine the survival of the patient and not the exposure to the radioactive materials in the RDD.

RESULTS

More than 100 persons attended the training sessions. The hospital staff was divided into groups according to their specialty and task when responding to such an event. Different areas of the hospital were signalized, contaminated or clean including areas in the emergency room and operating theatre. Different scenarios were discussed and changes to the checklists were adopted.

CONCLUSION

Responding to a terrorist event is a stressful task and even more when radioactive materials are involved. Up to now no such situation has actually occurred in any hospital. The hospital staff has been prepared for this type of event and this will be tested during the exercise on January 18.
Consideration of Radon Leakage From Marinelli Beakers for Accurate Determination of $^{226}$Ra Activity Concentration in Building Materials

K. Kovler$^1$, S. Levinson$^2$, Z. Priluzky$^1$, N. Lavi$^3$, Z. Alfassi$^3$, H. Naser$^4$, U. German$^2$

$^1$Technion - Israel Institute of Technology, Haifa 32000, Israel
$^2$Nuclear Research Centre Negev, P.O.B. 9001, Beer Sheva 84190, Israel
$^3$Ben Gurion University of the Negev, Beer Sheva 84105, Israel
$^4$Ministry of Environmental Protection P.O.B. 34033, Jerusalem 95464, Israel

Corresponding Author: cvrkost@technion.ac.il

INTRODUCTION

Radon is responsible for most of the public exposure to ionizing radiation, being a gas which can easily enter the human body by inhalation. It is created as part of the radioactive decay chains of natural radioisotopes. All building materials contain various amounts of natural radioisotopes (NORM), the most important being $^{40}$K and members of two natural radioactive series, represented by the isotopes $^{226}$Ra and $^{232}$Th. $^{220}$Rn is produced in the $^{232}$Th decay chain, but to a much lower extent than $^{222}$Rn, which is produced in the $^{238}$U chain as daughter of $^{226}$Ra. In practice, when referring to radon, the $^{222}$Rn isotope is meant.

Countries apply restrictions to the content of $^{40}$K, $^{226}$Ra and $^{232}$Th in the building materials, by issuing national standards. The Austrian Standard (1) and the Israeli Standard (2), address also radon emanation from building materials as part of the control criteria. The most widespread method for determination of the NORM content in building materials is gamma spectroscopy, by analyzing the pulse height spectra for characteristic gamma ray energies. As $^{226}$Ra emits only a low energy and low yield $\gamma$-ray (about 186 keV), which is overlapped by a $\gamma$-ray of similar energy from $^{235}$U, and $^{222}$Rn emits practically no $\gamma$-ray, the $^{226}$Ra activity can be determined by the $\gamma$-ray emission of the daughters, mainly 609 keV from $^{214}$Bi and 352 keV from $^{214}$Pb. The common method is to grind the building material sample, and to close it in a container of defined geometry, for which the detector system was calibrated. For optimal efficiency, Marinelli beakers are used, filled with the fine grinded material (to ensure minimal geometry variations), which envelop the detector.

During grinding and filling the Marinelli beaker most of the Radon escapes. Secular equilibrium must be attained again before counting the sample, for correct correlation between the measured $^{214}$Pb and $^{214}$Bi activity and the $^{226}$Ra activity. To achieve the equilibrium, the container is tightly closed and a wait time of about 3 weeks (several times the half-life of $^{222}$Rn: 3.824 d) is needed to achieve the radon buildup up its maximum concentration. This equilibrium concentration, which represents the activity of all radioisotopes in the radioactive chain, depends on the radon emission rate in the sample (which is determined by the $^{226}$Ra content) and the known decay rate of $^{222}$Rn. However, if there is a leak of radon from the Marinelli beaker, the equilibrium concentration will be lower, inducing an error in the determination of $^{226}$Ra activity.

Standard Marinelli beakers are made of plastic materials (mostly polyethylene or polypropylene), which are known to be permeable to radon to some extent (3,4), therefore inaccuracies in determination of the equilibrium activity are expected. The purpose of the present work is to investigate this effect for Marinelli beakers in use.
MATERIALS AND METHODS

Two series of experiments were undertaken. The first one was performed at the Technion, Haifa. A special calibrated radon emanation source produced by NIST – National Institute of Standards and Technology, USA, was placed in different type of containers, including a standard Marinellibeaker. The containers were sealed and were placed in a closed radon chamber made of metal with a volume of 11.5 L. A continuous radon gas monitor RAD-7, Durridge Company Inc., USA, was closed together with the tested container in the chamber, while the radon concentrations due escape from the container to the chamber volumewere measured continuously.

The second series of experiments were performed at the Ben-GurionUniversity. Samples of two types of Marinelli beakers of almost similar shape were supplied by GA-MA & Associates Inc. USA. One beaker was a regular one made of polypropylene, type 133N-E. The other one was a high impact polystyrene beaker type G-130G, defined as gas tight by the producer. The two Marinelli beakers were filled completely with fine-grinded, thoroughly mixed1371.7 g phosphogypsum, which contains a high concentration of radium(≈800 Bq/kg). After hermetically closure of the beakers (using alsoplastinc, which was proven as impermeable to radon), the two beakers were counted by a high resolution Ge spectrometry system every 2 days to record the radon buildup until reaching the equilibrium concentration.

RESULTS AND DISCUSSION

The results of the first experiments which measured leaked radon concentration from different containers are presented in Figure 1.

Figure 1. $^{222}$Rn concentrations measured in the radon chamber outside the closed containers enclosing the calibrated radon source.
It can be seen that Petri polystyrene dish is almost transparent to radon, the standard Marinelli container is also permeable, but containers made of glass, high-density polyethylene (with a special double lid) and PET (polyethylene terephthalate) are much less permeable.

The second series of experiments is preliminary and was performed to evaluate if there are detectable differences in the radon equilibrium concentration due to the permeability properties of the beaker materials. Theoretically, the time dependent radon concentration in a closed volume containing the radon source is given by:

\[
C(t) = C_0 e^{-\lambda_{\text{eff}} t} + C_{\text{max}} \left(1 - e^{-\lambda_{\text{eff}} t}\right) \tag{1}
\]

\(C_0\) is the initial radioisotope concentration, which was trapped between the solid particles at the moment of closing the beaker. \(C_{\text{max}}\) is the asymptotic equilibrium concentration due to the radon emanation. \(\lambda_{\text{eff}}\) is the effective decay constant of radon in the beaker, which is due to natural radioactive decay and leakage from the container. The difference between the actual \(\lambda_{\text{eff}}\) and the decay constant of radon (0.181 d\(^{-1}\)) is the measure of the beaker tightness. Figure 2 presents data of the radon activity as a function of time calculated by the 609 keV peak results for the two types of Marinelli beakers. A similar behavior was obtained for the 352 keV peak.

![Figure 2. The time-dependent activity relative to the initial value, measured by the 609 keV line for the two beaker types.](image)

The first data point indicates the initial radioisotope concentration trapped in the phospogypsum at the moment of closing the beakers. The adapted function is according to equation (1).

The results presented above are preliminary, and include a limited number of data. Therefore, no reliable function fitting could be performed, especially for the rising part of the spectrum.
from which the \( \lambda_{\text{eff}} \) can be determined. More detailed experiments will be made to evaluate this parameter. However, the equilibrium activity can be evaluated by the asymptotic values. It can be seen that this value is higher by about 10% for the high impact polystyrene than for the regular polypropylene one, pointing to a possible beaker material influence on the activity determination, although part of the difference may be attributed to small differences in geometrical shape of the beakers, which may influence the calibration factor. In order to check the significance of the radon leakage, the two Marinelli beakers were placed in closed radon chamber with active coal cassettes for 6 days. When counting the two cassettes, 63 Bq was obtained for the cassette enclosed with the 133N-E standard beaker, while a non-measurable radon activity was obtained for the other, indicating a significant difference between the radon tightness of the two beakers.

**CONCLUSIONS**

The results presented here show that the beaker material may play an important role in the accuracy of determined radon (and hence \( ^{226}\text{Ra} \)) activity. This issue was already mentioned in some works. An attempt to use a special aluminum beaker instead of a plastic Marinelli was undertaken (5), as well as using a container made of brass foil (6). However, special containers made of metal are not commercially available yet.

The present results point to a possible not negligible difference in radon leaking from counting beakers. The permeability of the Marinelli beaker to radon has a direct influence on the evaluated radon equilibrium concentration and hence on the accuracy of \( ^{226}\text{Ra} \) activity determination.

**ACKNOWLEDGEMENT**

We would like to thank GA-MA & ASSOCIATES Inc. USA, for sending us samples of the different Marinelli beakers to perform the present work.

**REFERENCES**

Radon Exhalation from Concrete Containing Fly Ash: *In-Situ* and Laboratory Measurements

G. Haquin¹, K. Kovler² and R. Becker²

¹Radiation Safety Division, Soreq Nuclear Research Center, Yavne, 81800, Israel
²Faculty of Civil and Environmental Engineering, Technion - Israel Institute of Technology, Haifa 32000, Israel

Corresponding author: gustavo@soreq.gov.il

INTRODUCTION

Most building materials of terrestrial origin contain small amounts of Naturally Occurring Radioactive Materials (NORM), mainly radionuclides from the $^{238}$U and $^{232}$Th decay chains and the radioactive isotope of potassium, $^{40}$K. The external radiation exposure is caused by gamma emitting radionuclides, which in the uranium series mainly belong to the decay chain segment starting with $^{226}$Ra. The internal (by inhalation) radiation exposure is due to $^{222}$Rn, and its short lived decay products, exhaled from building materials into the room air.

Due to economical and environmental reasons there is an increased tendency to use recycled industrial by-products containing Technologically Enhanced Natural Occurring Radioactive Materials (TENORM) in the building material industry. Fly ash (FA), produced as by-product in the combustion of coal, is extensively used in Israel since the early nineties of the last century in concrete and as an additive to cement (¹).

The increase of $^{226}$Ra activity concentration, the mineralogical characteristics of the FA and of the concrete may influence on the radon exhalation rate and consequently on the radon exposure of the public.

The recently published Israeli Standard 5098 (IS 5098) "Content of natural radioactive elements in building products" (²) limits the content of natural radionuclides as well as the radon emanation from concrete.

This paper present a study conducted to investigate and quantify that influence and to correlate between *in-situ* and laboratory measurements.

MATERIALS AND METHODS

Concrete samples containing different concentrations of FA (0, 120 and 140 kg/m$^3$ of concrete) were prepared at the concrete plant, reflecting common production processes, and used in Dwelling Shielded Spaces (DSS) the rooms made of massive concrete, which are used as living rooms normally.

The natural radionuclides content in the concrete was determined by gamma spectrometry method as indicated in the IS 5098 standard.

The free exhalation rate and the concrete emanation coefficient were measured according to IS 5098 standard procedure. Uncoated concrete samples of 10 cm x 10 cm x 20 cm were placed in a close chamber and the radon concentration was monitored using continuous radon monitor (CRM) as seen in Figure 1.

The free wall exhalation rate was determined under sealed conditions of the DSS as seen in Figure 2. The ventilation rate at closed condition in two DSSs was determined by two independent methods: decay of SF$_6$ concentration with time and calculation of the ventilation rate from the ingrowth curve of the radon concentration in the room.
RESULTS
The average free radon exhalation rate was determined using two approaches: Non-linear regression of the radon concentration ingrowths curve (Equation 1) and by linear regression of the slope for the first hours of the measurement (Equations 2 and 3).

\[ E_{0,av} = \frac{V}{\sum_{n} S_n} \cdot (C_{\text{max}} \cdot \lambda_{\text{eff}} - N \cdot C_o) \]  

\[ E_0 = \frac{V}{\sum_{n} S_n} \cdot [a + \lambda_{\text{eff}} \cdot C(0) - N \cdot C_o] \]  

when \( C(0) \approx 0 \) or \( C(0) \ll C_{\text{max}} \) and \( C_o \rightarrow 0 \) then

\[ E_{0,av} = \frac{V \cdot C_{\text{max}} \cdot \lambda_{\text{eff}}}{\sum_{n} S_n} \quad \text{or} \quad \frac{a \cdot V}{\sum_{n} S_n} \]  

where \( E_{0,av} \) is the average free exhalation rate, \( V \) is the volume of the measured space (or volume of DSS), \( C_{\text{max}} \) is the radon concentration at saturation, \( S_n \) are the exhaling areas, \( C(0) \)
is the radon concentration at measurement start, $C_0$ is the radon concentration in the laboratory (or outside of the DSS), and $\lambda_{\text{eff}}$ is the effective radon decay in the tested space. The slope of the radon concentration dependence for the first hours is obtained by $C(t) = a \cdot t + b$

The average free radon exhalation rate $E_{0,\text{av}}$ (and the emanation coefficient $e$) measured in laboratory samples tends to decrease in concrete containing FA as seen in Figures 3 and 4.

Figure 3: Average free exhalation rate and emanation coefficient from uncoated concrete sample.

Table 1 does not show a clear influence of the FA in the concrete on the average wall radon exhalation rate as measured in-situ using different approaches for the $E_{0,\text{wall,av}}$ calculation. The calculated average radon exhalation was not influenced by the ventilation rate during the in-situ measurement. Measurements under sealed ($\lambda_{\text{eff}} = 10^{-3}–10^{-4} \, \text{h}^{-1}$) and closed conditions ($\lambda_{\text{eff}} \sim 0.15 \, \text{h}^{-1}$) gave similar results.

Table 1: Average free radon exhalation from the DSS wall as measured in-situ

<table>
<thead>
<tr>
<th>FA content [kg/m$^3$]</th>
<th>$E_{0,\text{wall,av}}$ [Bq/m$^2$·h]</th>
<th>Standard Deviation [%]</th>
<th>Range [Bq/m$^2$·h]</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>7.16</td>
<td>27.4%</td>
<td>3.6-9.4</td>
</tr>
<tr>
<td>120</td>
<td>8.30</td>
<td>10.6%</td>
<td>6.3-10.3</td>
</tr>
<tr>
<td>140</td>
<td>7.54</td>
<td>23.6%</td>
<td>4.9-11.7</td>
</tr>
</tbody>
</table>

An evaluation of the influence of the measurement period on the maximum radon concentration ($C_{\text{max}}$) and on the effective decay of radon ($\lambda_{\text{eff}}$) was performed. A minimum period of 4 days was needed to achieve reliable results as shown in Figure 4.
CONCLUSIONS
The free radon exhalation rate was measured *in-situ* at DSS rooms made of massive concrete, with and without FA, and from the representative laboratory samples, as part of the study of the influence of FA addition into concrete.
It was found that in the laboratory measurements the radon exhalation rate decreases with addition of FA, while in the *in-situ* measurements no clear influence was found.
The ventilation rate in the DSS rooms was measured using SF$_6$ decay method and radon ingrowths. The ventilation rate in the sealed room is significantly lower than the radon decay rate, while in the most conservative living case (sealed window and closed door) the ventilation rate is $\sim 0.15$ h$^{-1}$.
The uncertainty of the measurement (*in-situ* and laboratory) was assessed to be $\sim 30\%$. The measurement conditions and the calculation method are the major factors influencing on the uncertainty.

ACKNOWLEDGMENT
The work is part of a study of the influence of FA addition to the radon exhalation from concrete jointly supported by the Ministry of Construction and Housing and the National Coal Ash Board. The authors are thankful to Mr. Zakhar Prilutsky and Zohar Yungrais for the assistance in conducting the *in-situ* and laboratory tests.

REFERENCES
Assessment of the Dose to the Population of Israel due to Inhalation of Radon

L. Epstein¹,², J. Koch¹, T. Riemer¹, G. Haquin¹ and I. Orion²

¹Radiation Safety Division, Soreq Nuclear Research Center, Yavne, Israel
²Department of Nuclear Engineering, Ben-Gurion University of the Negev, Beer Sheva, Israel

Corresponding Author: cahana@soreq.gov.il

INTRODUCTION

Radon (²²²Rn) is a radioactive noble gas that is the decay product of ²³⁸U. Radon has a half life of 3.8 d that allows it to travel easily from its place of origin (the ground or the building materials) to the open air. Radon accumulates in closed spaces like underground mines or inside buildings. It can reach relatively high concentrations in areas where high concentrations of ²³⁸U and especially of its decay product, ²²⁶Ra, exist in the ground.

Radon is known as the main contributor to the exposure of the population to natural sources of ionizing radiation. The exposure occurs as a result of the inhalation of the radon decay products which are attached to the aerosols in the air. Most of the dose comes from the alpha and beta particles emitted from the 4 short-lived decay products of radon: ²¹⁸Po, ²¹⁴Pb, ²¹⁴Bi and ²¹⁴Po.

The results of a radon survey in single-family houses in Israel were published in 2006 (¹). The survey included almost 2,000 dwellings and its goal was to determine the average radon concentration in single-family houses and to identify radon prone areas around Israel based on long-term measurements using Solid State Nuclear Track Detectors (SSNTD) commercially known as CR-39. It was found that the average radon concentration inside single-family houses is equal to 47.3 Bq/m³ and that a correlation between indoor radon concentration and the geological group on which the dwelling is built exists.

In the present study a radon survey of apartments in multistory buildings was conducted. The results of the two surveys were combined in order to assess the average radon concentration in Israeli dwellings and the annual dose to the Israeli population due to radon inhalation.

METHODS

The survey in multistory buildings was intended to assess the influence of the characteristics of the local building industry on the radon concentrations. These characteristics include the addition of fly ash (in which the concentrations of the natural radionuclides are higher than in other building materials) to concrete, beginning in 1985 and increasing gradually, and the improvement in sealing of doors and windows achieved using new building techniques. Another characteristic of buildings built since the early 1990's is a shielded area built in every apartment that is known as a DSS (Dwelling Shielded Space). The DSS is a room built from massive concrete walls, floor and ceiling that can be hermetically sealed and is intended to protect its residents from a missile attack. In many apartments the DSS is used as a bedroom.

The combined influence of the above mentioned features on the radon concentration was estimated by dividing the participating dwellings into two groups: apartments in buildings older...
than 20 years, built before fly ash was added to concrete and without a DSS and apartments in buildings newer than 10 years, built with fly ash, improved sealing and including a DSS.

The measurements were conducted using CR-39 SSNTDs that were placed in one of the bedrooms or the living room for a minimal period of 3 months. A second detector was placed in the DSS of dwellings in new buildings.

300 dwellings participated in the survey, one third of which were in new buildings and the rest were in old buildings.

RESULTS

In order to analyze the results the CR-39 detectors were divided into 3 groups:
1. Detectors that were placed inside dwellings in old buildings.
2. Detectors that were placed inside the living/bedroom of dwellings in new buildings.
3. Detectors that were placed inside the DSS of dwellings in new buildings.

Most of the measured radon concentrations were in the range of 10-100 Bq/m³. The arithmetic and geometric mean and standard deviation (SD) of the radon concentrations of the 3 groups are presented in Table 1.

<table>
<thead>
<tr>
<th></th>
<th>Arithmetic Mean and SD (Bq/m³)</th>
<th>Geometric Mean and SD (Bq/m³)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Living/bedrooms in old buildings</td>
<td>25.5 ± 12.6</td>
<td>22.9 ± 1.6</td>
</tr>
<tr>
<td>Living/bedrooms in new buildings</td>
<td>44.6 ± 18.9</td>
<td>41.3 ± 1.5</td>
</tr>
<tr>
<td>DSS in new buildings</td>
<td>49.5 ± 30.4</td>
<td>43.6 ± 1.6</td>
</tr>
</tbody>
</table>

The radon concentrations in each of the above mentioned groups present a log-normal distribution, as seen in the Israeli 2006 survey of single-family houses (1) as well as in other national radon surveys (3). T-tests were performed in order to test the statistical significance of the differences between the groups. It was found that a significant difference exists between the average radon concentration inside new and old buildings (P<0.001). This validates the assumption that the combined effect of the characteristics of new buildings (higher concentrations of radionuclides in the building materials, better sealing and the presence of a DSS) causes higher indoor radon concentrations.

A paired t-test was conducted to assess the significance of the differences between the radon concentrations in the DSSs and in the living/bedrooms of the same dwelling. The test showed that there is a significant difference between the two groups with higher concentrations in the DSS (P=0.04). The assumption that the combination of the higher density of the walls of the DSS, the higher concentrations of radionuclides and the better sealing causes higher radon concentrations in DSSs than in the living/bedrooms was therefore validated.

The difference between radon concentrations in single-family houses and dwellings in multistory buildings was also tested using a t-test and it was found to be significant (P<0.001).
It was found that the average radon concentration in Israeli dwellings, according to the distribution of the population in all dwelling types (single-family houses, apartments in old and new multistory buildings) is equal to 31.1 Bq/m³. The average radon concentration outdoors is equal to 10 Bq/m³\textsuperscript{(2,3)}.

The annual average dose to the population of Israel was estimated based on the average radon concentrations indoors and outdoors, the fraction of time spent indoors (0.8) and an equilibrium factor between the radon and its decay products (0.4 indoors \textsuperscript{(3)} and 0.6 outdoors \textsuperscript{(2,3)}). The conversion factor from radon concentration to dose that was used is based on the updated information included in the 2009 ICRP Statement on Radon \textsuperscript{(4)}. The annual average dose to the population of Israel due to radon inhalation is therefore 1.2 mSv.

CONCLUSIONS

The results of the 2006 radon survey in single-family houses in Israel were combined with the results of a recent survey of radon concentrations in multistory buildings in order to assess the average dose to the population of Israel due to radon inhalation. It was found that the average radon concentration in apartments in new buildings was significantly higher than in old buildings and the average radon concentration in single-family houses was significantly higher than in apartments in multistory buildings. The average dose due to radon inhalation was estimated on the basis of the updated information included in the 2009 ICRP Statement on Radon \textsuperscript{(4)} and was found to be equal to 1.2 mSv.

REFERENCES

The Exposure of the Israeli Population to Natural Sources of Ionizing Radiation

L. Epstein\textsuperscript{1,2}, J. Koch\textsuperscript{1}, G. Haquin\textsuperscript{1} and I. Orion\textsuperscript{2}

\textsuperscript{1}Radiation Safety Division, Soreq Nuclear Research Center, Yavne, Israel
\textsuperscript{2}Department of Nuclear Engineering, Ben-Gurion University of the Negev, Beer Sheva, Israel

Corresponding Author: cahana@soreq.gov.il

INTRODUCTION

Natural sources account for the greatest part of the exposure to ionizing radiation for most individuals \cite{1}. Natural sources have two origins: terrestrial radionuclides in the earth's crust and cosmic radiation.

The main contributors to the dose from natural sources are the terrestrial radionuclides, mainly \ensuremath{^{40}K} and the radionuclides in the decay chains of \ensuremath{^{238}U} and \ensuremath{^{232}Th}. These radionuclides can be found in the ground, the building materials, food and water and thus cause internal and external exposure. Approximately 50\% of the annual average worldwide dose from natural sources is contributed by the inhalation of radon (\ensuremath{^{222}Rn}) \cite{1} which is a decay product of \ensuremath{^{238}U}.

16\% of the annual worldwide dose comes from cosmic radiation \cite{1}. The cosmic radiation is composed of energetic particles that are produced as a result of the interaction of cosmic rays with the atoms in the atmosphere. The cosmic rays also produce several cosmogenic radionuclides. The dose from cosmic radiation depends on the altitude, the geographic location and the solar activity.

The goal of this study was to assess the average radiation dose from natural sources to the population of Israel, taking into account local characteristics (geology, geography) as well as specific features of the local lifestyle.

The building industry in Israel has a few distinctive characteristics that have an effect on the average annual dose: almost all buildings are made of concrete or concrete blocks and since 1985 fly ash (in which concentrations of terrestrial radionuclides are higher than in other building materials) has been added to cement and concrete in increasing amounts. Another characteristic of buildings built since the early 1990's is a shielded area built in every apartment that is known as a DSS (Dwelling Shielded Space). The DSS is a room built from thick concrete walls and ceiling that can be hermetically sealed and is intended to protect its residents from a missile attack. In many apartments the DSS is used as a bedroom.

RESULTS

The annual average dose to the population of Israel from natural sources of ionizing radiation was calculated as the sum of the doses from the different sources: radon inhalation, external and internal exposure to the terrestrial radionuclides and cosmic radiation.

The annual dose due to radon inhalation was calculated based on the average radon concentration in Israeli dwellings \cite{2}. The dose was found to be equal to 1.2 mSv \cite{2}, based on the updated information included in the ICRP Statement on Radon \cite{3}.
The dose as a result of external exposure to the terrestrial radionuclides was calculated inside buildings, in which the exposure is due to the gamma rays emitted from the radionuclides in the building materials, and outdoors, where the exposure arises from the radionuclides in the ground. The dose rate inside buildings was assessed using a model developed by de Jong and van Dijk (4,5) that takes into account the concentrations of terrestrial radionuclides in the building materials, the dimensions of the room and the density and thickness of the walls. The dose rates above different geological groups in Israel, as measured by the aerial survey conducted in 1998 (6), were used to calculate the average dose rate outdoors. The average dose rate inside buildings was found to be 25 nSv/h, approximately 1.3 times the average dose rate outdoors (20 nSv/h).

Internal exposure due to food and water consumption was estimated on the basis of the average consumption of different food categories in Israel (7) and the average worldwide concentrations of the radionuclides in the decay chains of $^{238}$U and $^{232}$Th in each food category (1). In addition, the annual dose as a result of internal exposure to $^{40}$K and $^{220}$Rn (thoron) was assessed based on average worldwide concentrations (1). It was found that the average annual dose as a result of internal exposure to the terrestrial radionuclides other than radon is equal to 0.4 mSv.

The annual dose due to cosmic radiation was calculated for the 120 most populated cities and towns in Israel using the PARMA analytical model, as implemented in the EXPACS code (Excel-based Program for calculating Atmospheric Cosmic-ray Spectrum) (8,9). The distribution of the annual dose from cosmic radiation was assessed, taking into account all of the factors affecting the dose. The average annual dose was found to be 0.2 mSv.

CONCLUSIONS

In the present study the average annual dose to the population of Israel from natural sources of ionizing radiation was estimated. It was found that the annual dose from all sources is 2.0 mSv, while the range of values extends from 1.7 mSv to 2.7 mSv. The main contributor to the dose is radon (60% of the total dose). Table 1 describes the average dose from each source and its range and Figure 1 shows the relative contribution of each source to the annual dose.
Table 1. The average annual dose to the population of Israel from natural sources of ionizing radiation.

<table>
<thead>
<tr>
<th>Source</th>
<th>Average dose (mSv/y)</th>
<th>Dose range (mSv/y)</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Cosmic radiation</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cosmic particles</td>
<td>0.20</td>
<td></td>
</tr>
<tr>
<td>Cosmogenic radionuclides</td>
<td>0.012</td>
<td></td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td><strong>0.21</strong></td>
<td><strong>0.17-0.28</strong></td>
</tr>
<tr>
<td><strong>External terrestrial exposure</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Indoors</td>
<td>0.18</td>
<td></td>
</tr>
<tr>
<td>Outdoors</td>
<td>0.03</td>
<td></td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td><strong>0.21</strong></td>
<td><strong>0.15-0.26</strong></td>
</tr>
<tr>
<td><strong>Internal exposure other than radon</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{40}$K</td>
<td>0.17</td>
<td></td>
</tr>
<tr>
<td>$^{238}$U and $^{232}$Th decay chains in food</td>
<td>0.11</td>
<td></td>
</tr>
<tr>
<td>$^{238}$U and $^{232}$Th decay chains in air</td>
<td>0.006</td>
<td></td>
</tr>
<tr>
<td>Thoron ($^{220}$Rn)</td>
<td>0.10</td>
<td></td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td><strong>0.39</strong></td>
<td><strong>-</strong></td>
</tr>
<tr>
<td><strong>Radon ($^{222}$Rn)</strong></td>
<td>1.2</td>
<td><strong>1.0-1.7</strong></td>
</tr>
<tr>
<td><strong>Grand total</strong></td>
<td>2.0</td>
<td><strong>1.7-2.7</strong></td>
</tr>
</tbody>
</table>

* The internal exposure other than radon has no range because the dose calculations are based on worldwide averages of radionuclides concentrations rather than on local measurements.
Figure 1. The relative contribution of the different natural sources to the average annual dose to the population of Israel.

REFERENCES
Differentiation between the Effect of Temperature and Pressure on Radon Transport within the Subsurface Geological Media

H. Zafrir, G. Haquin, U. Malik, S. M. Barbosa

1 Geological Survey of Israel, Jerusalem, Israel
2 Soreq Nuclear Research Center, Yavne, Israel
3 University of Lisbon, IDL, Portugal

Corresponding Author: zafrir@gsi.gov.il

INTRODUCTION

In radon long-term monitoring the primary requirement is to work in a passive mode under natural and undisturbed conditions. Such a passive mode is currently achieved by diffusion-type alpha particle detectors (silicon diodes and ionization chambers) and by gamma-ray detection systems.

There is an inherent difference between gamma and alpha detectors related to the sensing method. Gamma detectors count the photons emitted by the radon decay products (RDPs), Pb-214 and Bi-214. When set, for example, inside boreholes, within subsurface soil or rocks, they respond to the radon content within a cylindrical block of approximately 60cm x 60cm (around 30 cm is the average attenuation length for gammas from RDPs) and a volume of about 0.15m³ (300-400kg). Alpha detectors, set inside boreholes respond to radon atoms that enter by diffusion from the borehole air space into the detection volume of the detector (tens of cm³) and measure the alpha particles that are emitted by Radon-222 and by the RDPs, Po-218 and Po-214. The gamma detection method, aside of the difference in the sensing volume (2-3 orders of magnitude), has the advantage of monitoring temporal radon variations directly within the porous media before radon moves to the air space where the alpha detector is located. The behavior of alpha silicon diodes, gamma crystal scintillators and ionization chamber detectors employed for long-term radon monitoring in geological media was studied recently (1). It was found in this study that gamma detectors are more sensitive and suitable for borehole measurements than alpha detectors (silicon and ionization chambers).

Long-term monitoring studies of radon conducted in recent years within shallow soil, country rock and alluvial deposits (2) as well as with simulation facilities (3, 4, 5), revealed a typical cyclical daily signature in radon temporal variability. Longer radon signals, lasting for several days (6, 7, 8) and even long term seasonal patterns (9) have also been observed. Van der Spoel et al. (4) conducted an experiment with time-dependent advection radon transport, forced by a cyclic square wave pulse of upward air-flow for 2 and 4 day time intervals, in the bottom of a tank of a dry sand column. This experiment exhibited at the upper part of the sand column, a smoothly oscillating concentration wave of radon with the same periodicity (2 and 4 days).

Therefore it is reasonable to assume that periodic environmental parameters such as atmospheric pressure and temperature may cause a cyclic variation of radon concentration with the same period, within geological media. A long-term radon monitoring based on simultaneous alpha and gamma measurement, was carried out in an arid region in the southern part of Israel. Radon gamma and alpha detection systems were installed at the Amram mountain research tunnel.
situated 25 km NW of Elat, and radon gamma and alpha detection systems in shallow boreholes at the Gevanim valley in Makhtesh Ramon (Figure 1).

The results shows that the method to differentiate between the impact of ambient temperature and pressure on radon transportation within porous media, by long-term radon monitoring based on simultaneous alpha and gamma measurement, is functional and the above general assumption is acceptable.

Figure 1: Radon monitoring sites in southern Negev, Israel: (a) The geological setting of the Amram tunnel (b) The radon setup of 3 boreholes at the Gevanim valley in Makhtesh Ramon.

Figure 2: Annual radon time series, as measured by both gamma and alpha sensors in the internal closed space of the Amram tunnel, exhibit the same temporal variations.

RESULTS

In the Amram tunnel the monitoring site is a closed measuring space with undisturbed environmental conditions. The radon in the air space reaches equilibrium with the radon in the rock. Then the radon time series as measured by both gamma and alpha sensors exhibit the same temporal variations (Figure 2) as was also analyzed by the wavelet-based decomposition method.
to obtain a scale-by-scale description, from large to short time-scale, of the variability of radon and environmental time series (9). The results in the latter case indicated that the diurnal, intra-seasonal (multi-day) and seasonal variation in the radon concentration is clearly associated with the ambient temperature gradient outside the rock air interface (9) as seen in Figure 3, to a depth of 100 m. Radon variations are independent of atmospheric pressure also according to the evidence that intra-seasonal pressure waves lasting for several days frequently are independent of the intra-seasonal temperature waves (Figures 4), although the diurnal pressure is anti-correlated with the ambient temperature as expected from the world general climatic temperature-versus-pressure profiles.

**Figure 3:** The radon within the Amram tunnel consistently follows (a) the summer seasonal, (b) intra-seasonal and (c) diurnal ambient temperature variations with a daily delay of about 10 hours.
Figure 4: Atmospheric pressure and temperature at (a) Amram and (b) Ramon. Inside the Amram tunnel the year-round internal temperature is 28°C±0.3°C, while the temperature in the 80m bottom of the open borehole in Ramon has a daily cycle as on the surface. Intra-seasonal, atmospheric pressure and temperature are not correlated, but the diurnal pressure is anti-correlated with the ambient temperature. Yet, in shallow open boreholes as in the Gevanim valley, no equilibrium between the radon within the porous media and the radon in the open borehole air is necessarily established. The results of radon monitoring are different. Gamma sensors that measure the changes in radon concentration in the porous rock indicated a clear correlation between radon down to 80 m and the daily variation of the external surface temperature (Figures 5). The alpha detectors that measure the changes in radon concentration in very shallow borehole air (Figures 5a) revealed a clear anti-correlation with atmospheric pressure waves on semi-daily, daily, and intra-seasonal time scales. At depths of several tens of meters outer pressure waves induce anti-correlated radon variations lasting the same time but destroy the ordered radon daily periodicity in the measuring air space (Figure 5b). Noticeable is that, similar to the measurements in the Amram tunnel, the intra-seasonal pressure waves at the Ramon site are independent of the intra-seasonal temperature waves, (Figures 4b).

Figure 5: Radon by gamma and alpha detectors within 1.2 m (a) and 80-85m (b).

CONCLUSIONS
This work proposes a new method to differentiate between the impact of ambient temperature and pressure on radon transportation within porous media, by long-term radon monitoring based on simultaneous alpha and gamma measurement.
For a monitoring site as a closed measuring space with undisturbed environmental conditions, as in the Amram tunnel, the radon in the air space will reach equilibrium with the radon in the rock. Then the radon time series as measured by both gamma and alpha detectors exhibit the same temporal variations. The results in this case indicated that the diurnal, intra-seasonal and seasonal variation in the radon concentration is clearly associated with the ambient temperature gradient outside the rock air interface, to 100m above the monitoring cell.

If the monitoring site is a shallow borehole, no equilibrium between the radon within the porous media and the radon in the open borehole air is necessarily established and the results of radon monitoring are different. Gamma detectors measuring the changes in radon concentration in the porous rock indicated a clear correlation between radon concentrations and the daily variations of external surface temperature, from about 1m up to 85m.

On the contrary, the alpha detectors measuring the changes in radon concentration in very shallow borehole air space (about few meters) revealed a clear anti-correlation with atmospheric pressure waves at semi-daily, daily, and intra-seasonal time scales. At depths of several tens of meters, outer pressure waves induce anti-correlated radon variations lasting the same time. The ordered radon daily periodicity in the measuring air space is destroyed but the daily radon variation within the surrounding porous media, as measured by the gamma ray detectors is not disturbed.

REFERENCES
A New Laboratory System for the Study of the Effect of Temperature on Radon Transport Processes

G. Haquin1, H. Zafrir2, D. Katz1, Y. Yaffe1, D. Ilzycer1, Y. Ganot3, Z. Alfassi4, S. M. Barbosa5, D. Rendlich6, E. Sayag6, and N. Weisbrod3

1Radiation Safety Division, Soreq Nuclear Research Center, Yavne, 81800, Israel
2Geological Survey of Israel, Jerusalem, Israel
3Department of Environmental Hydrology and Microbiology, Blaustein Institutes for Desert Research, Ben-Gurion University of the Negev, Israel.
4Department of Nuclear Engineering, Ben-Gurion University of the Negev, Beer Sheva, Israel.
5University of Lisbon, IDL, Portugal
6Workshop Unit, Soreq Nuclear Research Center, Yavne, 81800, Israel

Corresponding author: gustavo@soreq.gov.il

INTRODUCTION

Radon (222Rn) is a radioactive inert gas with half-life of \( T_{1/2} = 3.825 \) days produced by radioactive decay of 226Ra (\( T_{1/2} = 1600 \) years) which is part of the 238U decay chain. Radon in nature appears in 3 different isotopes: 222Rn, 220Rn (Thoron – from the 232Th decay chain) and 219Rn (Actinon – from the 235U decay chain). The last two have short half-life (< 1min) which limits their transport range within the ground and therefore they are of less geophysical and health concern.

Several studies have been done on 222Rn in different areas from public health, geophysics, atmospheric science and more. Since 1992 in Israel research groups including scientists from the Geological Survey of Israel (GSI) and the Soreq Nuclear Research Center (SNRC) monitored the radon emanation from various lithological formations to ascertain the relationship between the radon concentration within the shallow and deep subsurface materials (soil and rocks) and tectonic events along the Dead Sea Rift and Makhtesh Ramon (1, 2, 3) region.

In-situ monitoring studies of radon as well as with simulation facilities (4, 5, 6) revealed several cyclical signatures: daily, longer radon signals lasting for several days (7, 8, 9) and even long term seasonal patterns (10). These studies related the temporal variations of radon to various atmospheric and geological parameters affecting radon transport in the subsurface.

Following Van der Spoel experiment (5) where the forced advection radon transport in a sand column was studied, it is reasonable to assume that periodic environmental parameters such as atmospheric pressure and temperature may cause a cyclic variation of radon concentration with the same period within geological media.

The goal of the laboratory system presented in this work is to investigate the mechanisms that control radon temporal changes produced within the geological subsurface media, specifically to explore the impact of temperature on this mechanism.

RADON MONITORING METHODS

Geophysical radon long-term monitoring is mainly performed using passive detectors under natural and undisturbed conditions. The monitoring is based on the detection of either the alpha radiation from the 222Rn, 218Po and 214Po decay or by gamma radiation from the radon decay products (RDPs) 214Pb and 214Bi.
The gamma detection method, besides the difference in the sensing volume (2-3 orders of magnitude), has the advantage of monitoring temporal radon variations directly within the porous media (up to 30 cm) before radon moves to the air space. The gamma detectors have been found to be more efficient than the alpha detection method having 50 counts per hour per Bq/m³ in the sensing volume. The advantages of the gamma detection have been published by Zafrir et al. (11).

An example of the geophysical radon monitoring using gamma detection systems installed in shallow boreholes at the Gevanim valley in Makhtesh Ramon, can be seen in Figure 1. It is clearly seen that the gamma sensors which measure the changes in radon concentration in the porous rock are strongly correlated to the daily variation of the external surface temperature. At depths of several tens of meters outer pressure waves induce anti-correlated radon variations lasting the same time but destroy the ordered radon daily periodicity in the measuring air space (alpha sensors).

Figure 1: Radon fluctuation recorded by gamma and alpha sensors in shallow boreholes at the Gevanim valley in Makhtesh Ramon at 80m depth.

CLIMATE CONTROL ROOM
The controlled experiments will be conducted in a Climate Control Room (CCR). The CCR at the Blaustein Institutes for Desert Research of Ben-Gurion University has the ability to sustain long-term steady conditions. At the CCR the temperature can be controlled by thermal radiation, thus air movement occurs by air density gradients and not by forced wind.

The CCR was constructed inside an insulated shipping container, 6 m long, 2.5 m wide, and 2.5 m high as seen in Figure 2. Cooling is obtained by flowing cool air (from an air conditioner) through aluminum pipes in a double-layered ceiling, which can be sealed but is noninsulated, thus cooling the room by radiation to impose no-wind conditions (12). Heating can be controlled by six spiral heaters that are located around the CCR, 1.5 m above the floor. The temperature in the CCR is monitored with six thermocouples (TC), and can be controlled by two of the TCs located near the radon container. Barometric pressure is monitored by a pressure transducer located inside the CCR.
Figure 2: CCR 2-D cross-sectional illustration, blue color indicating the cool air flow in an independent system. 1. Air conditioner; 2. Air router box; 3. Four aluminum air pipes; 4. Double layered ceiling; 5. Four aluminum air pipes for return air along the walls; 6. Hanging thermocouples for monitoring and controlling temperature.

RADON CYLINDRICAL CONTAINER
A large cylindrical container (60 cm diameter x 117.5 cm high) was constructed at Soreq workshop and it is equipped with horizontally collimated gamma detectors, and vertically inserted temperature and pressure sensors at different internal levels. In the bottom of the container a curvature shape base with a pressure hose allows the development of a 2D pressure gradient along the container. In the center of the container a PVC rod with 10 TCs will record the temperature gradient along the container, in parallel to the monitor of the pressure gradient performed at various heights of the container. The container will be packed with homogeneous crushed rock material (soil, granite or phosphate rocks) and will be encircled with thermal insulation material except for the upper surface as seen in Figure 3. Heat will be applied to the top of the material column to enable continuous stable heat flow producing a steady state of down-migrating radon.
**Figure 3:** Cylindrical container equipped with horizontal collimated gamma detectors and vertically dispersed temperature and pressure sensors at different internal levels.

**EXPERIMENTS**

The working hypothesis of this work is the assumption that time-dependent heating of rock or soil surface media creates an oscillating radon flow with a similar period along a porous media column.

The simulation with the natural daily and seasonal variations (summer and winter) will be achieved by heating or cooling the top of the material column to different temperatures: 20 to 45°C for the summer and 7 to 25°C for the winter. The high sensitivity of the gamma detectors in the collimated housing and the implementation of fragmented phosphate rocks with radon concentration 100 times higher than widespread rocks at the column are the key expedients to achieve radon transport parameters including thermal conductivity at this "micro scale".

An upward air flow will be induced under mass-flow control through the internal material column in order to balance the radon thermal diffusion flow within it. The conditions at equilibrium could enable us to extract the physical parameters that are responsible for the "thermal advection" of the radon within the porous media.

The set of experiments will be in the future supported by a theoretical study in order to develop an expansion of the existing 3D model for radon transport by diffusion and advection in shallow subsurface rocks (13) based on thermal gradient and thermal advection as an additional driving force for radon gas flow.

**REFERENCES**


Radon Diffusion in Open Air

Y. Shitrit\textsuperscript{1,2}, A. Dody\textsuperscript{1}, Z.B, Alfassi\textsuperscript{2}

\textsuperscript{1}Nuclear Research Center - Negev, P. O. B 9001, Beer-Sheva 84190, Israel
\textsuperscript{2}Department of Nuclear Engineering Ben Gurion University, Beer-Sheva, Israel

Corresponding Author: yakovsh@bgu.ac.il

INTRODUCTION

Radon (\(^{222}\text{Rn}\)), a radioactive noble gas with a half-life of 3.8 days is formed in the soil from radium (\(^{226}\text{Ra}\)), which is a decay product of Uranium (\(^{238}\text{U}\)). Radon is able to move from sub-surface to the free atmosphere by mechanisms of convection and diffusion while the exhalation is controlled by meteorological parameters. Most articles explained the fluctuations in the radon concentration in air by positive or negative correlations to air temperature\textsuperscript{(9,14,18)}, barometric pressure\textsuperscript{(8,17)}, wind speed\textsuperscript{(2,3,4,18)} and relative humidity\textsuperscript{(1-4,16)}. Others showed no correlation to those parameters.

Because there are no agreements between the articles about the influence of these meteorological variables on radon concentration, we decide to explore this issue deeply.

The aim of our study is to get a better understanding of the reasons of diurnal radon fluctuations. We will measure radon counts under lab conditions with known air temperature, relative humidity and barometric pressure indicator. We will use a closed column to prevent the influence of wind speed.

These measurements will be carried out with an alpha detector based on photodiodes, which mounted on the top of a system containing P.V.C column and a \(^{226}\text{Ra}\) source as a radon emitter\textsuperscript{(14)}.

Effect of Meteorological Variations on Radon Concentration-Literature Survey

Effect of Temperature

Results obtained in some researches do not indicate on inconclusive and clear correlation between atmospheric radon concentration and air temperature. Magalhaes et al.\textsuperscript{(9)} showed an inverse correlation between radon at equilibrium-equivalent concentration and temperature. The increase in temperature caused to reduction of radon concentration and vice versa (Fig. 1).

Shitrit et al.\textsuperscript{(14)} found a positive correlation between the air temperature and \(\alpha\) counts. High \(\alpha\) counts were obtained during the daytime, when the temperature was higher, and low counts measured during night and early morning hours, when the air temperature was low (Fig. 2). The same behavior was observed by Zikovsky and Chah\textsuperscript{(18)} in their measurements in Montreal, Canada.
Richon et al.\(^{(11)}\) measured the radon concentration in atmosphere along horizontal tunnel located in the French alps, where the air temperature was 6.7°C with a seasonal differences of 0.1°C. Despite the constant temperature variations of radon concentrations during the daytime were found.

**Effect of Barometric Pressure**

Klusman\(^{(8)}\) in his study found an inverse correlation between this barometric pressure and radon concentration. When the pressure decreases, gas is sucked out of the ground and radon concentration is increased. Whittlestone et al.\(^{(17)}\) reported the same correlation.

In contrast to these results, no correlation was found by other investigators. According to Kesikikuru\(^{(6)}\), changes in barometric pressure were negligibly affecting the rate of radon concentration in air. The same behavior was shown by Rovenska and Thinova\(^{(12)}\) in Bozkov cave, Czech Republic, during summer season 2008. They found that there is no correlation between barometric pressure and radon levels. Richon et al.\(^{(11)}\) found in their study diurnal fluctuations in radon concentration with constant pressure (pressure differences smaller than 1 Pa).
Effect of Wind Speed
The relationship between wind speed and atmospheric radon concentration is not straightforward and clear: Duenas et al.\(^2\) found a high positive correlation between the hourly mean variation of radon and wind speed. This behavior has been corroborated by several investigators: Soto\(^{15}\), Gonzalez\(^4\) and others. Zikovsky and Chah\(^{18}\) found that the wind velocity had a little effect on radon concentration. Galmarini\(^3\) found a negative correlation with wind speed. Marley\(^{10}\) found that wind speed appears to have the potential for a dual influence on radon variability: directly, through wind pressure differences (up chart of Fig. 5) and indirectly, through changes to the air components (bottom chart of Fig. 5).

Fig. 5: 24h moving average values (MAV) of radon related to wind speed\(^{10}\)

Effect of Relative Humidity (RH)
Tanner\(^{16}\) found high positive correlation between atmospheric radon concentration and relative humidity. An increase in humidity produces an increase in radon concentrations above the ground. These results are analogous to those obtained by Duenas et al.\(^2\), Gonzalez and Garzon\(^4\), Cautenet\(^1\) and Galmarini\(^3\). The same results were observed in Seftelis\(^{13}\) research. Outdoor radon concentrations, together with meteorological parameters of gold mining areas in NW. Province, South Africa were investigated by Kgabi et al.\(^7\). In two sites they found that the relative humidity have very little or no correlation (r=0-0.3) on radon concentration.

SUMMARY
From the Literature survey we can see that there is a great confusion and many arguments about the influences of meteorological parameters on atmospheric radon concentration. There is no one and clear opinion about these influences and a comprehensive study is need.
In our study we are going to check the real parameter which had the main influence on radon concentration in air in diurnal cycle, by isolating the other meteorological parameters. It can be that another parameter which is not related to meteorological parameters had the most influence.
REFERENCES
Water Chemisorption on a Sputter Deposited Uranium Dioxide Film – Effect of Defects

S. Cohen\textsuperscript{1,2}, N. Shamir\textsuperscript{1}, S. Zalkind\textsuperscript{1}, A. Seibert\textsuperscript{3}, T. Gouder\textsuperscript{3} and M. H. Mintz\textsuperscript{1,2}

\textsuperscript{1}Nuclear Research Center-Negev, POB 9001, Beer-Sheva 84190, Israel
\textsuperscript{2}Dept. of Nuclear Eng, Ben-Gurion Univ. of the Negev, POB 653, Beer-Sheva 84104, Israel
\textsuperscript{3}European Commission – Joint Research Centre, Institute for Transuranium Elements, POB 2340, 76125 Karlsruhe, Germany

Corresponding Author: shaico@bgu.ac.il

1. INTRODUCTION

Uranium Dioxide (UO$_2$) is mainly used as fuel for nuclear power plants for electricity generation and in the manufacturing of radioisotopes. Today more than ~440 nuclear power plants are operating, generating a total capacity of ~0.4 TW. Therefore, UO$_2$ surface structure and reactions have been studied extensively\textsuperscript{(1)}. The two main corrosion threats to UO$_2$ are moisture and hydrogen. Moisture is mostly effective in atmospheric conditions (endless supply), and its reactivity can be further increased by radiation damage and radiolysis during long term storage. The enhanced oxidation and hydrogen formation can lead to a buildup of pressure and eventually a threat of breaching the storage container and a release of radioactive materials. Therefore, the safe disposal of UO$_2$ spent fuel is crucial to environment safety. Previous studies\textsuperscript{(2,3)} had measured the effects of water adsorption on single crystal and thin film UO$_2$ surfaces with high resolution X-ray photoelectron spectroscopy (XPS) and temperature programmed desorption (TPD). In the present study, the characteristics of water vapor chemisorption on a UO$_2$ film, obtained by reactive sputter deposition, were studied over a temperature range of 300-500 K by combined measurements of direct recoil spectrometry (DRS) and XPS.

2. EXPERIMENTAL

2.1. The Experimental System

An ultra-high-vacuum (UHV) system pumped by turbo-molecular and titanium sublimation pumps (to a base pressure of ~2x10\textsuperscript{-10} Torr) incorporated with Auger electron spectroscopy (AES), XPS and DRS was employed\textsuperscript{(4)}. The DRS\textsuperscript{(5)} is based on grazing irradiation of the surface with a pulsed beam of 3 keV Ar\textsuperscript{+} ions and the time of flight measurements of the surface atoms and ions, which are recoiled in a forward direction, following the direct collision inflicted by the impinging primary ions. The main characteristics of this technique are topmost surface sensitivity, detection of light atomic masses, including hydrogen, and being non-destructive. The DRS can also probe different geometrical arrangements of the adsorbing species due to the so-called “shadowing” effect. Shadowing effects of the H atoms by neighboring O atoms, as manifested by hydrogen/oxygen ratio (H(DR)/O(DR)) can differentiate between full and partial dissociation routes of water molecules surface as well as point to the geometrical arrangements of hydroxyl groups on the surface\textsuperscript{(6)}.

2.2. The Experimental Procedures

Stoichiometric polycrystalline thin films (100 Å) of UO$_2$ were deposited onto a Mo foil substrate (1 mm thickness). The sample was prepared at the Institute for Transuranium Elements (ITU), by direct-current sputter deposition from native U metal with an Ar/O$_2$ as sputtering gas. Prior to each experiment, the sample was sputtered-cleaned, and then annealed for 5 min under oxygen atmosphere (2x10\textsuperscript{-5} Torr) at 600 K. Surface sample defects were...
obtained by 30 min 5KeVAr⁺ sputtering at RT. Both sputtered and annealed samples were monitored by the DRS technique, during water vapor exposure to pressures of $2 \times 10^{-8} \text{ - } 2 \times 10^{-7}$ Torr at the temperature range of 300-500 K. XPS measurements were taken before each experiment (clean reduced or oxidized sample), and after 100 L (1 L = $10^{-6}$ Torr×sec) water vapor exposure.

3. RESULTS AND DISCUSSIONS
The DRS hydrogen to oxygen intensity ratio vs. water exposure at the temperature range of 300-500 K, for both sputtered and annealed surface, are presented in Fig. 1 and Fig. 2 respectively.

The H(DR)/O(DR) ratio decreases with rising temperatures on both types of surfaces without any pressure dependence. For the annealed sample the H(DR)/O(DR) ratio (Fig. 2) is a regular behavior with no hydrogen shadowing and saturation is almost achieved at ~5 L. The H(DR)/O(DR) ratio of the sputtered sample (Fig 1) show somewhat different behavior. The hydrogen accumulation curves show rapid adsorption up to 10L with small shadowing observed above 5 L with twice intensity compared to the annealed sample. These results suggest that on sputtered surface, dissociative chemisorption of water vapor is enhanced on surface defects. With increasing temperature, the surface defected sites heal and the sticking coefficient decreases, so the dissociation of the water molecules and adsorption decrease. Fig. 3 shows XPS results of the oxygen peak O1s at the temperature range of 300-500 K before (black curve) and after 100 L water adsorption on the surface of sputtered and annealed sample, respectively. The results of the sputtered sample show with rising temperature from RT to 500 K that oxygen intensity increases with a shift of 0.5 eV to lower binding energy even before water exposure. After 100 L water exposure at RT, the oxygen concentration increases by 20%. In addition, a chemical shift is observed from 530.6 eV to a lower binding energy of 529.9 eV at RT and 400 K. These results combined with previous DRS results support the indication that annealing of the UO₂ surface heals defected sites, probably by migration of oxygen atoms from the oxide bulk as suggested by Idriss et al.². After water exposure at RT, water molecules dissociate on oxygen deficient sites (due to the sputtering process), heals surface defects and oxygen intensity increases. The movement to a lower
binding energy after annealing or water adsorption is associated with oxidation of UO$_{2-x}$ to a Stoichiometric UO$_2$ surface. At 500 K there is no change in the binding energy and oxygen intensity before and after 100 L water exposure since all surface defects are completely healed. The results of the annealed sample at all temperatures do not show any chemical shift after water adsorption, but the oxygen intensity increases with rising temperatures. These result combined with the small adsorption as seen by DRS results suggest that water molecules do not dissociate at all on annealed surface but only on defects that have not healed yet. None of our XPS results indicate the formation of a shoulder at O1s peak that can be attributed the presence of OH as seen by previous studies$^{(3)}$ with high resolution XPS, probably due to a poorer resolution of our instrument.

![Fig. 3: O1s peak with and without water exposure, at various temperatures, of the sputtered and annealed samples.](image)

Therefore, based on these combined measurements of DRS and XPS results we propose here an adsorption model of H$_2$O on thin film UO$_2$ Surface illustrated at Fig. 4. Up to 1 L, many sites are available on the defected surface and there is higher concentration of hydroxyls compared to the sparse concentration on the annealed surface. But since the surface coverage is still low and there is a vast distance between neighboring fragments (OH and H), no hydrogen shadowing occurs on both surfaces. At higher coverage of ~5 L, the annealed surface is already close to saturation and OH coverage remains low. No hydrogen shadowing is observed due to low availability of adsorption Sites. The higher availability of adsorption sites on the sputtered surface at higher coverage, increases hydroxyls adsorption on the surface. Since OH are now more packed on the defected surface, tilting due to hydrogen bonds causes some of the hydrogen atoms to enter DRS shadowing cone. Increasing the dose to 10 L does not change significantly the OH concentration on annealed surface and it remains low. On the sputtered sample, the surface is now saturated with a mixing of tilted and standing OH on the defected surface. The proposed model is currently further investigated, using high resolution electron energy loss spectroscopy (HREELS) in order to confirm the suggested adsorption model.
4. CONCLUSIONS

The effect of defects on the chemisorption behavior on stoichiometric UO₂ surface at the temperature range of 300-500 K was evaluated by comparing sputtered and annealed surfaces. It seems that dissociative chemisorption takes place on both types of surfaces. Dense clusters of tilted hydroxyls are assumed to be formed on the sputtered surface, whereas the more isolated hydroxyls present on the annealed surface stay upright. Hopefully, HREELS measurements will confirm the suggested adsorption model.

5. REFERENCES

Crystallographic Structure and Magnetic Properties of HAVAR and HAVAR-H under High – Pressure using Diamond Anvil Cell (DAC)

Itzhak Halevy\textsuperscript{1,2}, Shlomo Haroush\textsuperscript{3,4}, Yosef Eisen\textsuperscript{4}, Ido Silberman\textsuperscript{4}, Dany Moreno\textsuperscript{4}, Amir Hen\textsuperscript{5}, Amir Broide, Mike L. Winterrose\textsuperscript{2} and Zhiqiang Chen\textsuperscript{6}

\textsuperscript{1}Nuclear Research Center - Negev, P.O. Box 9001, Beer-Sheva, Israel
\textsuperscript{2}Department of Materials Science California Institute of Technology, Pasadena, CA 91125, USA
\textsuperscript{3}On sabbatical leave of absence from NRC Negev, Israel, Soreq NRC, Yavne, Israel
\textsuperscript{4}Soreq NRC, Yavne, Israel
\textsuperscript{5}Departments of Nuclear Engineering, Ben Gurion Univ., Beer-Sheva, Israel
\textsuperscript{6}NSLS, Brookhaven National Laboratory, Upton, NY 11973, USA

Corresponding author: halevyi@caltech.edu

ABSTRACT
Annealed (H1), cold-rolled (H2) HAVAR and hydrogenated cold-rolled HAVAR have been studied using high-pressure synchrotron x-ray diffraction. A structural phase transformation was discovered at \( \sim 13 \) GPa at ambient temperature, transforming from m\(-3\)m (S.G. 225) to P\(-63/m\)m\(_c\) (S.G. 194) symmetry. The transition was not reversible on pressure release. The low-pressure cubic phase was found to be more compressible than the high-pressure hexagonal phase. Conventional Mössbauer and NFS show that the HAVAR is not magnetic at room temperature and no splitting is observed.

The SQUID indicates a huge difference in the temperature dependence of the magnetic susceptibility between the cold Rolled HAVAR compared to the annealed HAVAR. The HAVAR-H was found to be stiffer than the regular HAVAR by the high pressure results and BPT (Ball Punch Test).

The phase transition is at the same pressure in the region of the pressure uncertainty. There are differences in the temperature dependence of the magnetic susceptibility between the cold Rolled HAVAR and HAVAR-H by the SQUID.

INTRODUCTION
HAVAR foils are of technological importance in applications such as windows for gas and liquid targets. In medical cyclotron physics, for example, HAVAR foils are used as windows for H\(_2\)O\(_{18}\) targets which produce \(^{18}\)F isotope as the main component of the FDG (fluorodeoxyglucose). HAVAR foils are used because of their ability to withstand high-temperatures and pressures intrinsic to FDG production. As such, a deeper understanding of the behavior of HAVAR foils under extreme conditions is desirable.

Medical cyclotron is a particle accelerator used in producing short lived radiotracers such as \(^{18}\)F, \(^{11}\)C, \(^{15}\)O, \(^{13}\)N etc. These radiotracers are labeled with suitable pharmaceuticals for use to gather information related to metabolic activity of the cell using Positron Emission Tomography (PET) scan. The main goal of FDG-PET scan (FDG – fluorodeoxyglucose), is to detect metabolically active malignant lesions including lung cancer, colorectal cancer, lymphoma, melanoma, breast cancer, ovarian cancer, brain cancer and multiple myeloma. FDG-PET scan may also be used to stage and monitor the response to therapy of malignant disease.
HAVAR consists primarily of Co together with Cr, Ni, Fe [1]. This paramagnetic alloy has a cubic crystallographic structure (FCC) with a 3.582 Å lattice parameter. The aim of the present study is to investigate the high-pressure structural properties of annealed (Ann.) and cold rolled (CR).

SAMPLE PREPARATION AND METHODS

The composition and structure of the annealed, at 1200°C (H1) and cold Rolled (H2) HAVAR samples were characterized by TEM (Jeol, 2010 Fastem (200 KV)). The samples were cleaned ultrasonically and in acetone and kept under Ar atmosphere until loading in the diamond anvil cell (DAC). A special sample, HAVAR-H, was prepared by hydrogenating the HAVAR at 500°C and 50 Atm. of hydrogen.

Sample characterization

Pressure (up to ~36.5 GPa) was generated using a “Tel-Aviv”- type diamond anvil cell (DAC) (a type of Merrill-Bassett DAC) [2]). The experiments were conducted using a monochromatic X-ray beam (0.4066Å and 0.37677 Å for Havar and Havar-H, respectively). The high pressure measurements were carried out on a sample ~30 μm in thickness and 100 μm in diameter. The pressure was measured using the Ruby fluorescence technique [3]. Silicon oil in the sample cavity was used as a pressure medium. The pressure distribution inside the sampling space was checked at different regions, and was determined to vary by less than 5%. Data was acquired for approximately 15 minutes at each pressure step.

X-ray

High-pressure angle dispersive X-ray diffraction studies were performed at beamline X17-C of the National Synchrotron Light Source (NSLS). The angle dispersive data were collected using a 2D detector with opening Bragg angle (2θ=30°). The high-pressure X-ray powder-diffraction measurements were taken at discrete pressure steps in the range of 0 to 36.4 GPa. The data was collected by the angle- dispersive- system, ADS, technique. A monochromatic X-ray beam (0.4066Å and 0.37677 Å for Havar and Havar-H, respectively) from high order Si crystal monochromator was used. The angle dispersive measurements were carried out in transmission configuration using the 2D image detector technique. The data was analyzed using a commercial Rietveld analysis software package [4]. The 2D data was converted to the 2θ dimension by the Fit2D [5] software.

RESULTS AND DISCUSSION

Electron microscope studies showed that the micro-structure of the annealed specimen (H1) contains equiaxed grains about 2 μm grain sizes, few dislocations in the bulk and small quantity of carbides. Electron diffraction measurements showed that the carbides are of M23C6 type (M= 3d metal, more likely Cr), contain almost all the matrix elements, and have FCC crystallographic structure with a 10.65Å lattice parameter.

The cold rolled specimen (H2) displays preferred orientation texture (deformation texture) due to the rolling process. The specimen has an FCC crystallographic structure with a lattice parameter of 3.580(5)Å for the HAVAR and of 3.590(5)Å for the HAVAR-H. Further, the H2 specimen shows a very high dislocations density as well as malty twinning compared to the annealed sample. In addition, the H2 sample contains the same carbides as the annealed sample, corresponding to composition and structure.

The X-ray diffraction patterns from H1 and H2 and HAVAR-H samples at high pressure (Fig. 1) indicate that HAVAR does not keep the same symmetry as the volume is reduced. We fitted the data to the Fm -3 m (S.G. 225) symmetry up to ~13 GPa. The P 63/m m c (S.G. 194) symmetry was used for the fitting above 13 GPa. A clear decrease in unit cell parameters
occurs with increasing pressure. On pressure release the original unit cell parameters and symmetry were not recovered and the symmetry of the high pressure phase remained, indicating the existence of a hysteresis.

**Fig. 1:** X-ray Diffraction pattern of HAVAR (H1) annealed and (H2) cold Rolled as function of pressure. [6]

**Fig. 2:** X-ray Diffraction pattern of HAVAR-H cold Rolled as function of pressure.

The Lattice parameters of cold rolled HAVAR and HAVAR-H are given in Fig 3, as function of pressure. The unit cell parameter ratio (c/a) in the HCP symmetry is almost constant (0.605(5)) from the transition until the highest pressure.
**Fig. 3:** The unit cell parameters $a$ and $c$ for the hexagonal symmetry and $a$ for the cubic symmetry as function of pressure obtained by Rietveld analysis. HAVAR-H, on the left and HAVAR right.

The line width of the annealed (H1) HAVAR specimen differs significantly from the cold rolled (H2) sample.

The volume-pressure curves, calculated from the data, are depicted in Fig.4. The relationship between pressure and volume change for both materials was determined by the Vinet equation of state [6].

$$P(V) = 3B_0 \left(\frac{V}{V_0}\right)^{-\frac{2}{3}} \left(1 - \left(\frac{V}{V_0}\right)^{\frac{1}{3}}\right) \exp \left(\frac{3}{2} (B_0' - 1) \left(1 - \left(\frac{V}{V_0}\right)^{\frac{1}{3}}\right)\right)$$

$B_0$ is the isothermal bulk modulus at room temperature and ambient pressure, and $B_0'$ is the partial derivative of the isothermal bulk modulus against pressure under the same conditions. This constitutive relationship was shown to be universally valid for all solids under the same conditions.

**Fig. 4:** The reduction of the $V/V_0$ ratio of the cold rolled HAVAR (right) and HAVAR-H (left) as function of pressure. Bulk moduli for the cubic and hexagonal phases are shown. The line is the fit to the Vinet [7] equation of state.

The bulk modulus ($B_0$) results for the cubic HAVAR annealed and cold rolled specimen are $138.2 \pm 5.7$ and $146.7 \pm 6.5$ GPa, respectively. The $B_0'$ was keep at 4.0 for both samples. The bulk modulus ($B_0$) results for the hexagonal HAVAR cold rolled specimen is $216.5 \pm 3.8$ GPa. The bulk modulus ($B_0$) results for the, cubic and hexagonal, cold rolled HAVAR-H specimen are $305 \pm 10$ and $316 \pm 11$ GPa, respectively. The $B_0'$ was keep at 4.0 for both samples.

**MAGNETIC PROPERTIES**

HAVAR foils were used for a conventional Mössbauer and NFS for identifying the magnetic states at room temperature. The HAVAR and HAVAR-H are not magnetic at room temperature and no magnetic splitting is observed.
The SQUID measurements indicate a huge difference in the temperature dependence of the magnetic susceptibility between the cold Rolled HAVAR compared to the annealed HAVAR, and small differences between HAVAR and HAVAR-H, see Fig 5.

![SQUID measurement for annealed and cold Rolled HAVAR and HAVAR-H.](image)

The SQUID measurement for annealed and cold Rolled HAVAR (left) indicates a big difference in the ordering temperature hinting for different environment at the $^{57}$Fe Mössbauer probe. Further work under high pressure and low temperatures should be done in the different samples.

BPT (Ball Punch Test) results are 64.8±3, 45±1.5 and 40.6±3.0 [N] for failure for the HAVAR-CR, HAVAR-CR-HT and HAVAR-CR-HT-H, respectively.

**CONCLUSIONS**

The cold Rolled HAVAR-H is much stiffer than the HAVAR as it is shown in the high pressure results and Ball Punch Test. There is a similar nonreversible phase transition as function of pressure.

**REFERENCES**

A previously unknown neptunium-transition-metal binary compound, Np$_2$Co$_{17}$, has been synthesized and characterized by means of powder x-ray diffraction, $^{237}$NpMössbauer spectroscopy, SQUID magnetometry, and x-ray Magnetic Circular Dichroism (XMCD). The compound crystallizes in the Th$_2$Ni$_{17}$-type hexagonal structure, with room temperature lattice parameters $a = 8.3107(1)$ Å and $c = 8.1058(1)$ Å. Magnetization curves indicate the occurrence of ferromagnetic order below a $T_C > 350$ K. Mössbauer spectra suggest a Np$^{3+}$ oxidation state and give an ordered moment of $\mu_{\text{Np}} = 1.57(4)$ $\mu_B$ and $\mu_{\text{Np}} = 1.63(4)$ $\mu_B$ for the Np atoms located, respectively, at the 2b and 2d crystallographic positions of the P$6_3$/mmc space group. Combining these values with a sum rule analysis of the XMCD spectra measured at the neptunium M$_{4,5}$ absorption edges, one obtains the spin and orbital contributions to the site-averaged Np moment ($\mu_S = -1.88(9)$ $\mu_B$, $\mu_L = 3.48(9)$ $\mu_B$). The ratio between the expectation value of the magnetic dipole operator and the spin magnetic moment ($\mu_{\text{md}}/\mu_S = +1.36$) is positive as predicted for localized 5f electrons, and lies between the values calculated in intermediate coupling (IC) and in jj approximations. The expectation value of the angular part of the spin-orbit interaction operator is in excellent agreement with the IC estimate. The ordered moment averaged over the four inequivalent Co sites, as obtained from the saturation value of the magnetization, is $\mu_{\text{Co}} = 1.6$ $\mu_B$. The experimental results are discussed against the predictions of first-principle electronic structure calculations based on the spin-polarized local spin density approximation plus Hubbard interaction.
RESULTS

The X-Ray Diffraction of the Np2Co17 is given in figure 1.

![Figure 1: Observed (dots) and calculated (red line) x-ray diffraction patterns recorded at room temperature for Np2Co17. The lower trace (blue line) is the difference profile. The intensity distribution is plotted as a function of the full diffraction angle 2θ (Cu Kα radiation). Vertical ticks indicate calculated angular positions of the Bragg peaks for the Np2Co17 phase (upper row, black) and the NpO2 impurity phase (lower row, red).](image)

The Rietveld analysis results yields the atoms configuration and parameters as given in table 1 and figure 2.

Magnetic field dependence of the Np2Co17 magnetization measured on a polycrystalline sample at different temperatures, from 20 to 350 K is given in figure 3.

$^{237}$Np Mössbauer spectrum of Np2Co17 taken at 4.2 K in the ferromagnetic state is given in figure 4 and support the Th2Ni17-type structure.

X-ray absorption spectra $\mu^+,-$ measured at 50 K at the M4,5 Np absorption edges in Np2Co17; data have been measured with photon helicity parallel and antiparallel to a 7 T magnetic field applied along the beam direction. The results are given in figure 5.
TABLE I: Refined structural parameters for Np$_2$Co$_{17}$ at room temperature. The parameters are referred to the hexagonal axes (space group P6$_3$/mmc, $a = 8.3107(1)$ Å and $c = 8.1058(1)$ Å). An isotropic Debye-Waller factor BDW = 0.5 Å$^2$ and full occupation were assumed for all sites.

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Figure. 2: Crystal structure of Np$_2$Co$_{17}$, corresponding to the ABAC stacking sequence characteristic of the ordered Th$_2$Ni$_{17}$-type structure. The coordination environment of the Np atoms located at sites 2b and 2d is shown in panels b) and c) respectively.

Figure. 3: Magnetic field dependence of the Np$_2$Co$_{17}$ magnetization measured on a polycrystalline sample at different temperatures, from 20 to 350 K. The inset shows the temperature dependence of the magnetization measured up to 300 K in a field $\mu_0 H = 0.1$ T under field cooled (FC) and zero-field cooled (ZFC) conditions.
Figure. 4: $^{237}$Np Mössbauer spectrum of Np$_2$Co$_{17}$ taken at 4.2 K in the ferromagnetic state. The solid red line represents the best fit to the data (dots), as given by the superposition of component spectra associated to the two inequivalent Np positions. The individual spectra, vertically shifted with respect to the base line, are shown as dotted blue and green lines on the top panel. The lower trace (black line) is the difference profile. The fit slightly improves by adding the contribution from the NpO$_2$ impurity, giving a single absorption line at 7.7 mm/s, where the difference profile shows a small dip.

Figure. 5: X-ray absorption spectra $\mu^+,−$ measured at 50 K at the M$_{4,5}$ Np absorption edges in Np$_2$Co$_{17}$; data have been measured with photon helicity parallel (red line) and antiparallel (black line) to a 7 T magnetic field applied along the beam direction. The XMCD signal (blue line) is given by $\mu^+−\mu^−$. The inset shows the field dependence of the XMCD signal at the M$_4$ edge.

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What is the Structure of Liquid Bismuth?

El'ad N. Caspi¹, Yaron Greenberg¹, Eyal Yahel¹, Brigitte Beuneu², Guy Makov³

¹Physics Department, Nuclear Research Centre-Negev, 84190 Beer-Sheva, Israel
²Laboratoire Léon Brillouin (CEA-CNRS), CEA/Saclay – 91191 Gif-sur-Yvette cedex, France, EU
³Materials Engineering Department, Ben-Gurion University of the Negev, 84105 Beer-Sheva, Israel

Corresponding Author: caspie@nrcn.org.il

INTRODUCTION

RDF (radial distribution factor) cannot be measured directly in real space. Instead, diffraction patterns from the liquid are collected and the liquid structure factor is then obtained by a series of approximations to account for scattering (multiple and inelastic) and absorption in the sample and the experimental setup. All these issues have been known for a long time and a standard set of procedures has been developed to analyze the experimental data and extract the structure factor. Following the application of these correction procedures, the accumulating systematical error in the case of neutron scattering from heavy elements is of the order of 4-5 %⁷⁻⁴. Yet, an additional error is introduced by the use of finite range in the momentum transfer, q, space introducing errors into the RDF, most notably short wavelength oscillations (i.e. numerical noise). These oscillations may affect the interpretation of the RDF, and in particular prevent identification of small changes therein.

In the present work, we report on two separate studies, 5 years apart, of the structure of liquid Bi at ambient pressure that we have performed at the 7C2 diffractometer at Saclay. Using different experimental setups, we show that the structure factors obtained agree to better than 1%, thus establishing the consistency of our analysis procedures. Using this structure factor as a reference, we compare a large number of experimental studies of liquid Bi at ambient pressure in order to investigate the experimental scatter in the structure factor and find out its significance. We then explore how the errors in the experimental determination of the structure factor affect the calculated RDF, focusing on two main issues: sensitivity to convergence in the high q range and choice of q_max.

RESULTS

Structure factor – S(q). In this work, we have the opportunity to compare our two measurements separated over 5 years using different setups, on the same instrument. This comparison can be undertaken to a high accuracy, while removing extraneous, non-physical sources of experimental scatter. The validity of the procedure described above can be determined by requiring that different experimental measurements of I(q) produce the same structure factor. This comparison is difficult to achieve, since the corrections to the measured signal applied by different researchers are not identical, nor are the details of the analysis procedures.

Trivially, the experimental signal to noise ratio decreases as the collection time is increased. The ratio between two measurements (5 years apart and at the same temperature) of liquid bismuth structure factor, S(q), shows that the deviation is typically less than 1% i.e. of the order
of the statistical noise. This is an indication that the obtained structure factor is indeed the “physical one”. Nevertheless, since the analysis methods were the same in both cases, a systematic error due to uncertainties in dimensions of the diffractometer’s components within the beam path used for the absorption corrections\(^{1}\), and the approximation made in the inelastic scattering corrections\(^{3}\), cannot be ruled out. We compared our results with previous studies over the last 6 decades (Figure 1a). Clearly, the positions of the maxima and minima are essentially the same, as expected from the relatively exact determination of the momentum transfer in diffraction measurements. In contrast, the amplitude of \(S(q)\) varies considerably between the measurements across the years. The magnitude of the variation is approximately 30\% in (\(S(q)-1\)) even for the first peak in \(S(q)\). Moreover, relative scatter between measurements increases with increasing \(q\). Finally, all measurements reduce to noise with the typical \(q_{\text{max}}\) of 10 Å\(^{-1}\), with the exception of our results at 7C2 which extend to approximately 14 Å\(^{-1}\) keeping a reasonable signal to noise ratio (Figure 1b).

**Figure 1**: (a) Various measurements of structure factor of liquid bismuth near the melting point over the last 60 years\(^{5-12}\). (b) The high \(q\) range of the structure factor.

Radial distribution function – \(G(r)\). The RDF, \(G(r)\), is the manifestation in direct space of the structure factor and as such is easier to interpret as representing a one dimensional average liquid structure. However, due to the experimental constraints, \(S(q)\) can be determined up to a certain \(q_{\text{max}}\) value. This may prompt the question of how to obtain the “best” representation of \(G(r)\) from the limited data. A naïve approach might be to use the maximum amount of \(q\) range available in the data, and, trivially, to aim for the longest measurement time.

The effect of measurement statistical noise in determining the structure factor on the RDF was found to be negligible when comparing the RDFs obtained from structure factors measured by us with twice and six times the number of neutron counts.

In Figure 2, we find that the oscillations in the RDF associated with the finite \(q\)-space measurements may increase with increasing \(q_{\text{max}}\) (Figure 2). We note that the main impact of the oscillations is in the region between the first and second hard sphere peaks in the RDF. This is to be expected as the high \(q\) range in the structure factor affects mainly the finer details of the liquid structure, such as the existence of shoulder, or hump between the 1\(^{\text{st}}\) and 2\(^{\text{nd}}\) hard sphere peaks in \(G(r)\) in anomalous liquid metals\(^{5,12}\).
Further progress may be obtained if we consider the effect on the calculation of $G(r)$ of using different cut-off values [$q_{\text{max}}$ at the $S(q)$ measured data] at selected positions from a positive to negative node in $S(q)$ (cf. Figure 1). We see that choosing the cut-off at a node, i.e. at $S(q_{\text{max}})=1$, leads to a reduction in the oscillations (Figure 2) with better results at a positive node. However, choosing the cut-off at the highest available positive node in the data requires some caution if the chosen $q_{\text{max}}$ is in the region where the statistical noise is too high. The sensitivity of $G(r)$ to $q_{\text{max}}$ is examined by considering alternative choices determined by the positive nodes of $S(q)$ (Figure 3). We see that as expected from the analysis above, as $q_{\text{max}}$ is increased the results for $G(r)$ converge and the oscillations are reduced. From these results we see that measurements limited to 10 Å$^{-1}$ still introduce a spurious contribution to the RDF, which is significantly reduced, but not eliminated at 11.8 Å$^{-1}$ (Figure 3).

The RDF is well-known to be sensitive to the $S(q)$ data cut-off limit, i.e. $q_{\text{max}}$ up to which the $S(q)$ data is chosen for the calculation of $G(r)$. We see that extending the measurement range in $q$ space, even significantly, does not ensure convergence of the RDF. We also note that statistical convergence does not in itself ensures the convergence of the RDF, which is also affected by the choice of $q_{\text{max}}$. 

**Figure 2:** Liquid bismuth RDF as determined from the $S(q)$ measured at 2006 on the 7C2 diffractometer at Saclay. Various $q_{\text{max}}$ cut-offs in the $S(q)$ data are presented. A positive node is when $S(q_{\text{max}})=1$, and $dS(q)/dq>0$ (at $q=q_{\text{max}}$). 

The RDF is well-known to be sensitive to the $S(q)$ data cut-off limit, i.e. $q_{\text{max}}$ up to which the $S(q)$ data is chosen for the calculation of $G(r)$. We see that extending the measurement range in $q$ space, even significantly, does not ensure convergence of the RDF. We also note that statistical convergence does not in itself ensures the convergence of the RDF, which is also affected by the choice of $q_{\text{max}}$. 

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The RDF is well-known to be sensitive to the $S(q)$ data cut-off limit, i.e. $q_{\text{max}}$ up to which the $S(q)$ data is chosen for the calculation of $G(r)$. We see that extending the measurement range in $q$ space, even significantly, does not ensure convergence of the RDF. We also note that statistical convergence does not in itself ensures the convergence of the RDF, which is also affected by the choice of $q_{\text{max}}$.
CONCLUSIONS
We suggest the requirements on the accuracy needed in measuring $S(q)$ for adequate $G(r)$ calculations: choosing the highest value of "positive-node" $q_{\text{max}}$ in the $S(q)$ data, while keeping significant (factor of 3) signal to noise ratio in the dip (peak) just before (after) this $q_{\text{max}}$ value. Moreover, we demonstrate here how the measurement of $S(q)$ up to a $q_{\text{max}}$ value, which is too low, hinders the ability to study the fine details in the calculated RDF, even when the statistics of the measured $S(q)$ is adequate. We would like to stress, that it is only through careful study of liquid structure factors up to large $q_{\text{max}}$ that it will be possible to identify the small structural changes occurring in liquids.

REFERENCES
Numerical Analysis of the Microwave Auditory Effect

N. M. Yitzhak, R. Hareuveny, R. Ruppin
Radiation Safety Division, Soreq NRC, Yavne, Israel

Corresponding Author: nirmor@soreq.gov.il

INTRODUCTION
The ability of appropriately modulated radiofrequency and microwave electromagnetic (EM) pulses to produce auditory sensation in human beings and mammals is known for decades. The common explanation of this phenomenon is the thermoelastic expansion mechanism, proposed by Foster and Finch (1): The absorption of the electromagnetic (EM) energy produces a fast thermal expansion, resulting from a small but rapid temperature rise, of the order of $10^{-5}$ °K. This launches a thermoelastic wave that travels via bone conduction to the inner ear, which causes an auditory sensation. Detailed review of the results of human, animal and modeling studies, which support this explanation of the hearing effect, has been published by Elder and Chou (2).

Several analytical calculations of this phenomenon were presented (3),(4). However those calculations assumed a simplified homogeneous spherical head model. Numerical analyses were also presented. These analyses were performed only for an EM plane wave of a single frequency (915 MHz) (5) or for high-pass birdcage MRI coil for several lower frequencies (64, 300 and 400 MHz) (6).

In the present work a numerical calculation of the thermoelastic wave propagation in three human models (male: body and head & shoulders and female: body) is performed. The EM absorption and the elastic wave propagation are calculated by the finite-difference time-domain (FDTD) technique. An analysis of the results over a wide EM spectrum is presented.

METHODS
The calculations were executed in two steps. The first one was a FDTD calculation of Maxwell's equations and the result was the specific absorption rate (SAR) caused by the EM plane wave. The simulation of this step was done by a commercial program, XFDTD Bio-pro ver. 6.6.10.4 of Remcom Inc., State College, PA. It also supplied the three human models and their relevant tissues' properties for the EM step as well as the specific heat capacities. Table 1 summarizes the physical parameters of these models.

The second step was an elastic FDTD simulation that was written for this research and used the SAR results as the source for the calculation. The finite difference equations were based on a previous work (5). The linear thermal expansion coefficient was taken to be $2.7 \times 10^{-5}$/°K for hard tissues and $13.0 \times 10^{-5}$/°K (the value for blood) for soft ones (7). Since the experimental data on the elastic properties of tissues vary over several orders of magnitude, approximately median values have been taken, and those values and a literature review will be published elsewhere.

Table 1. Physical parameters of the three models

<table>
<thead>
<tr>
<th></th>
<th>Male</th>
<th>Female</th>
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<tbody>
<tr>
<td></td>
<td>Body</td>
<td>Head &amp; shoulders</td>
</tr>
<tr>
<td>Cell size (mm)</td>
<td>5</td>
<td>2</td>
</tr>
<tr>
<td>Height (mm)</td>
<td>1,875</td>
<td>348</td>
</tr>
</tbody>
</table>
RESULTS
A typical result of the acoustic wave that evolves in the cochlea is shown in figure 1 (for pulse of 70 µs, 1 mW/cm², horizontally polarized plane wave that is incident from the back of the model), and its fast Fourier transform (FFT) in figure 2. The main acoustic frequency is approximately 8 kHz like in a previous analysis (5). The calculated acoustic pressure in the cochlea for different frequencies and for the same type of pulse can be seen in figure 3.

<table>
<thead>
<tr>
<th>Width (mm)</th>
<th>570</th>
<th>532</th>
<th>530</th>
</tr>
</thead>
<tbody>
<tr>
<td>Depth (mm)</td>
<td>325</td>
<td>284</td>
<td>285</td>
</tr>
<tr>
<td>No. of tissue's types</td>
<td>39</td>
<td>26</td>
<td>32</td>
</tr>
<tr>
<td>Total mass (gr)</td>
<td>108,902.0</td>
<td>14,248.9</td>
<td>84,777.6</td>
</tr>
</tbody>
</table>

Figure 1. The acoustic pressure in the cochlea (whole body male model) due to 2450 MHz EM pulse.
Figure 2. Acoustic pressure vs. acoustic frequency in the cochlea (whole body male model) due to 2450 MHz EM pulse.

Figure 3. The acoustic pressure in the cochlea as a function of the EM frequency for all the three models.

CONCLUSIONS
The acoustic spectrum has been calculated, and the dependence of the phenomena in the EM spectrum has been presented. The main acoustic frequency of the microwave auditory effect is found to be in the range of 5 - 8 kHz as has been found in humans experiments\(^{(2)}\). Furthermore, the dependence on the EM frequency has been calculated and the acoustic pressure is found to be highest in the range of 100 – 2,000 MHz (past experiments have given only the upper limit of 10,000 MHz\(^{(2)}\)). The maximal acoustic pressure (for the same power density) is at the frequency of 750 MHz for the male body model, 200 MHz for the female model and 240 MHz for the head & shoulders model.

REFERENCES
Occupational Exposure to Electromagnetic Fields from MRI Systems

N.M. Yitzhak, R. Hareuvany, R. Ruppin
Soreq NRC, Yavne, Israel

Corresponding Author: ronen@soreq.gov.il

INTRODUCTION
The interest in the possible health hazards of occupational exposure of MRI workers to electromagnetic fields, and the compliance with the relevant safety standards has recently increased. One of the reasons for this is the Physical Agents (Electromagnetic Fields) Directive adopted by the European Union in 2004\(^{1}\), which was due to be incorporated into the domestic legislation of member states by April 2008. Restrictions imposed by this directive would adversely affect current clinical MRI activities. In view of the concern raised by the MRI community, the deadline for the implementation of the directive was postponed to April 2012, so as to allow sufficient time to take into account new recommendations from relevant international bodies. Here we present the results of comprehensive measurements of the static magnetic fields in the vicinity of the MRI systems in medical centers in Israel.

MEASUREMENTS
We have mapped the static magnetic fields of the MRI systems by performing measurements along the lines shown in Figure 1, using an isotropic Metrolab THM 7025 Gauss meter. All measurements were performed at a height of 1 m above the ground. Typical results of the static magnetic field measurement are shown in Figure 2.

STATIC FIELD EXPOSURE GUIDELINES
The exposure of workers in Israel to non-ionizing radiation in general, and to static magnetic fields in particular, is subject to the limitations defined by the latest Threshold Limit Values (TLV) issued by the American Conference of Governmental Industrial Hygienists (ACGIH). At the time of survey, the ACGIH TLV specified that routine occupational exposures should not exceed 60 mT whole body or 600 mT to the limbs on a daily, time-weighted average basis. The ceiling values were 2 T for whole body exposure and 5 T for the limbs. Furthermore, medical electronic device wearers should not be exposed to field levels exceeding 0.5 mT. Based on these TLV and on the results of our measurements, we formulated safety instructions for MRI workers. However, shortly afterwards, new and less restrictive TLVs were published\(^{2}\), which did not involve any time weighted average. The new TLVs specified that routine occupational exposure should not exceed 2 T in the general workplace environment, but can have ceiling values of 8 T for workers with special training and operating in a so called "controlled workplace environment". Special training involves making workers aware of
transient sensory effects that can result from rapid motion in static magnetic fields with flux densities greater than 2 T. A controlled workplace environment is one in which forces exerted by static magnetic fields on metallic objects do not create potentially hazardous projectiles.
Exposure of the limbs of workers in the general workplace environment should not exceed 20 T. The 0.5 mT limitation on medical device wearers remained unchanged. Thus, in MRI systems with static fields lower than 2 T, which comprise almost all of the systems in Israel, there is no need to limit the occupational exposure to these fields. In the few systems with static fields higher than 2 T, the conditions of "special worker training" and "controlled workplace environment" have to be implemented. The entrance of individuals wearing implanted electronic medical devices into regions in which the static magnetic field is higher than 0.5 mT should be forbidden.

TIME VARYING ELECTROMAGNETIC FIELDS
Two types of time dependent electromagnetic fields exist in MRI systems:
(a) Switched gradient – This is a magnetic field varying linearly in space, which is switched on and off rapidly during the imaging process. The frequency content of the pulsed field is generally in the range of a few hundreds Hz. The TLV for magnetic fields specified by the ACGIH in the frequency range between 300 Hz and 30 kHz is 0.2 mT. In order to check whether the exposure of MRI workers in Israel complies with this TLV, it will be necessary to measure these fields.
(b) Radiofrequency (RF) field – This is a field in the 10s – 100s MHz range (the frequency is proportional to the strength of the static magnetic field). The RF field is applied to the subject being imaged using coils, either within the magnet housing or around the subject's body. The RF field is significant in safety terms only in the immediate vicinity of the coils, and the whole body exposure of MRI workers is expected to be lower than the relevant TLVs.

SUMMARY
The results of a survey of MRI systems in Israel, initiated by the Ministry of Industry Trade and Labor, have been presented. The static magnetic fields in the vicinity of the MRI systems have been measured. Most of the medical MRI systems in Israel employ 1.5 T magnets, so that the static magnetic fields around them are lower than the current occupational threshold limit values. Even in the few cases where stronger magnets are used, compliance can be achieved by implementing the conditions of "special worker training" and "controlled workplace environment". However, a mapping of the region which is forbidden to individuals wearing electronic medical devices (the 0.5 mT line) is always required.

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REFERENCES
Temporal Variations of Power Frequency Magnetic Fields in Residential Buildings with Indoor Transformer Stations

R. Hareuveny 1, N. Yitzhak 1, S. Kandel2, R. Ruppin1

1Soreq NRC, Yavne, Israel
2The Hebrew University, Jerusalem, Israel

Corresponding Author: ronen@soreq.gov.il

INTRODUCTION
Beginning with the 1979 work by Wertheimer and Leeper (1), many epidemiological studies on the association between extremely low frequency (ELF) magnetic fields (MF) exposure and childhood leukemia have been performed. Two pooled analyses (2,3) pointed to a consistent increase in the risk of developing childhood leukemia in children exposed to residential MF levels above 0.3-0.4 µT. However, this might be attributed to selection bias (4). The international epidemiologic study TransExpo, which utilizes the MFs in apartments located above internal transformer stations (ITS), is intended to minimize the potential for selection bias. The MFs in such apartments were reported to be significantly higher than in other apartments in the same building (5-8). In the TransExpo study the exposure assessment will be based on the relative location of the apartments to the transformer rooms in the same building. Due to rarity of both the disease and the exposure of interest, TransExpo is planned as an international study involving several countries, so as to achieve a sufficient sample size for a meaningful study. Pilot studies, aiming to clarify the relation between MF levels and apartment location relative to the ITS, have been performed in Finland (5), Hungary (6), Israel (7) and Switzerland (8). These studies have shown that a simple classification of exposure into "high"/"low" MFs, based on whether an apartment is located above an ITS is feasible with remarkable sensitivity and specificity. Here, the results of 24h continuous MF monitoring in some apartments are reported and analyzed (9).

METHODS
The 24-h measurements described here have been performed in conjunction with the spot measurements reported previously (7). The following four types of apartments were defined: First floor apartment right above the transformer room (AAT); Other first floor apartments (FF); Second floor apartment above transformer room (SAT); Other apartments on second and higher floors (HF). In the analysis of the results, the FF and the SAT apartments will be combined together as an Intermediate group.

Continuous 24-h measurements were performed in 16 apartments (8 AAT, 2 FF, 2 SAT, 4 HF) in 8 buildings. For these continuous measurements a MF meter (EMDEX II, Enertech Consultants, Campbell, CA, USA) was employed, using the broadband mode (40-800Hz), with a dynamic range of 0.01-300 µT and measurement accuracy of ±3%. The meter was placed as close as practically possible to the location of the highest MF of the apartment, as determined from the spot measurements, but avoiding the close proximity of working household appliances. Recordings were made at time intervals of about 1.5 sec (or 5 sec in some cases).

RESULTS
Typical 24-h recordings in two apartments in the same building are shown in Fig. 1.
One minute averages were calculated for all the 24-h recordings. These data were normalized to the average value of the magnetic field of the apartment and then averaged over the various apartment types. It was found that in all apartment types the MFs peak in the evening hours, around about 6 PM to 12 PM. In order to investigate the relation between the time dependence of the MFs in the various apartment types, temporal correlation coefficients were calculated between various pairs of apartments. Using the running one minute averages of the measured MFs, the correlation coefficients were obtained from

$$r(x, y) = \frac{\sum (X_i - \bar{X})(Y_i - \bar{Y})}{\sqrt{\sum (X_i - \bar{X})^2} \sqrt{\sum (Y_i - \bar{Y})^2}}$$

Here $X_i$ and $Y_i$ are the MF levels measured at the $i$-th minute (one minute average), in apartments $x$ and $y$, respectively. $\bar{X}$ and $\bar{Y}$ are the corresponding 24-h time averaged MFs. The calculated correlation coefficients were then averaged over pairs of apartment types. For example, the average HF-HF correlation was obtained by calculating the average of all correlation coefficients for which both $x$ and $y$ represent HF apartments ($x \neq y$). It was found that the correlations between AAT apartments and Intermediate ones in the same building were high, while the corresponding AAT-HF correlations were very low. Regarding correlations between apartments in different buildings, the AAT-AAT average correlations are high. As for the others, there exist medium AAT-Intermediate correlations, and only small AAT-HF and HF-HF correlations.

Calculated hourly load curves for the day of peak power consumption were supplied by the Israel Electric Corporation (IEC) for all the ITSs above which the MF measurements have been performed. The measured peak load values for the different ITSs varied between 210 and 811.
kW. In order to evaluate the temporal correlations between the calculated hourly load curves and the measured MFs, one hour averages of the latter were first calculated. The temporal correlation coefficients are 0.89 and 0.02 for the AAT and HF cases, respectively. The normalized calculated load curves were averaged over all ITSs, and in Figure 2 the result is compared with the normalized measured MF averaged over the AAT apartments. The temporal correlation coefficient between the two curves is 0.90. The calculated load curve refers to the winter workday of peak consumption, so that the peak at about 8 PM is higher than for non peak days. Thus, it can be assumed that if the load curve referred to an average day, the temporal correlation coefficient would be even larger. The corresponding correlation coefficients between the load curve and the average normalized hourly MFs in Intermediate and HF apartments are 0.64 and 0.29, respectively.

![Figure 2](image)

**Figure 2.** Average normalized calculated hourly ITS load (blue squares) and average normalized hourly MF measured in AAT apartments.

**DISCUSSION**

In the spot measurements it was found that the average MF in AAT apartments (0.33 µT at a height of 0.5 m) was significantly higher than in other apartment types (0.07 µT in FF, 0.11 µT in SAT and 0.06 µT in HF). The association between the MF in AAT apartments and the ITS electric currents also manifests itself in the temporal correlations. Thus, there exists a high correlation (0.90) between the AAT apartments average MF and the calculated average load of the ITSs. The high temporal correlation, which was found to exist between the MFs in AAT apartments in different buildings, can be utilized for estimating the 24h average MF in AAT apartments from spot measurements. This can be done by placing the measurement result on its appropriate time point on the average AAT curve of Figure 2, and scaling accordingly.

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REFERENCES


Cognitive Effects of Radiation Emitted by Cellular Phones: A Summary of Three Experiments

I. Eliyahu¹, R. Hareuveny¹, R. Luria². M. Margaliot¹, N. Meiran³

¹Soreq NRC, Yavne, Israel
²Tel Aviv University, Tel Aviv, Israel
³Ben-Gurion University of the Negev, Beer-Sheva, Israel

Corresponding Author: ilan@soreq.gov.il

INTRODUCTION
The widespread use of cellular phones has initiated research regarding the possible biological effects of exposure to Radiofrequency Radiation (RFR). This issue is of particular relevance because a considerable number of people are exposed to cellular RFR emitted in close proximity to their heads. Some studies, recently reviewed by Barth et al.¹, examined whether cognitive functions of human beings are altered while exposed to RFR emitted by cellular phones. This paper summarizes the results of three experiments we conducted in order to examine the effects of exposure to RFR emitted by a standard GSM phone at 890 MHz on human cognitive functions²,³,⁴.

MATERIALS AND METHODS
In all three studies, healthy right-handed male subjects were asked to perform cognitive tasks while two standard Nokia™ 5110 GSM cellular phones were attached to both sides of their head by a specially designed non-conductive frame. The phones were controlled by a base station simulator and operated at their full power. Specifically, the power was controlled by an HP GSM test system model E6392B. This system maintained the phones at either no transmission or full power transmission (890.2 MHz, 2 W peak power). The maximum SAR value reported for the Nokia 5110 model ranges from 0.54 W/kg to 1.09 W/kg, depending on the phone position. The phones were located in a position as similar as possible to that of typical use, and the antenna was located approximately 1.5 cm away from the subject’s head. The phones were battery operated during the experiments. The RF exposure regime was single-blinded. The phones were silent during the whole test. At the end of each experiment, subjects were asked whether they heard any noises, but none of them reported such. It is interesting to note that subjects also failed to judge which phone was operating during the experiment.

In the first experiment, four distinct cognitive tasks were used – a spatial item recognition task named "Face", a verbal item recognition task, and two spatial compatibility tasks. The tasks chosen were those that are known to have high hemisphere specificity. However, only the "Face" task were used in the other two experiments. This task was as follows: three target "faces" are presented for 650 ms each, one after the other, in three random locations out of eight possible. These eight possible locations are positioned as a 3 by 3 square, excluding the middle position. After an additional 3000 ms, another face appears in a random position. The subject has to decide whether the last face location matched any of the preceding three locations, and to respond by pressing a key with either the right-hand (to mark a match) or the left-hand (a mismatch), using the “/” key and the “z” key respectively. The examined parameters were the response time (RT) and the percentage of correct responses made by the
subjects. Only trials in which the response was correct were included in the response time analyses. A repeated measures analysis of variance (ANOVA) was performed.

**FIRST EXPERIMENT**

Our first study\(^2\) attempted to establish a association between the exposure of a specific area of the brain and the cognitive functions associated with that area. Thirty six subjects switched between the four distinct cognitive tasks described above. All subjects performed all four tasks under three exposure modes: right-side of the head, left-side, and sham exposure (counterbalanced within subject design). This resulted in 12 sub-sessions per subject. The experiment was divided into two sessions of one hour each. Each subject served as his own control, namely, his performance without exposure was compared to his own performance under exposure (a repeated measures design).

Statistically significant slowing effect was found in left hand responses under left side RFR exposure in the spatial item recognition task - "Face" (Fig. 1). This effect became evident only in the second part of the experiment, namely: after an hour of test, of which 40 minutes were under full power exposure. Same trend was also observed in the spatial compatibility task.

![Figure 1. First experiment: response times in ms for right-hand responses (right panel) and left-hand responses (left panel)](image)

**SECOND EXPERIMENT**

The objective of this study\(^3\) was to replicate our previous findings and to examine the time dependence of the effect. Specifically, this time we divided the experiment into 12 blocks of 50 trials each. The whole experiment lasted typically 1 hour per subject. Forty eight subjects were randomly divided into three equal groups. The subjects in each group were exposed to only one of the three exposure conditions: left-side of the head, right-side, or sham exposure. The "Face" task was chosen because it showed the largest effect in the previous study.

It was found that the average RT of the right-hand responses under left-side exposure was significantly longer than under right-side exposure and sham exposure averaged together during the first two blocks (Fig. 2). Left-hand responses showed the same pattern, but it was weaker and non-significant. These results confirmed the existence of an effect of exposure on RT and replicated some of the results obtained in the first work, namely: in both experiments RT's of left-side exposed subjects were different from those of the sham and right-side exposed ones. In addition, both studies demonstrate that RFR effects may be time dependent, namely they can be observed only at specific phases of the experiment.
Some differences between the two studies should be mentioned. One of them is that this time the effect was significant for the right hand, and not significant (although, numerically evident) for the left hand, while in the first experiment slowing was observed for the left-hand responses only. In addition, the time dependence of the effect was different: in this experiment the RFR effects were apparent at the first two blocks of the experiment and decreased over time. Hence, RT's of the various groups became similar towards the end. In contrast, in the previous work the average RT's of all exposure groups were similar in the first half of the experiment. The RT (of the affected hand of the left-side exposed subjects) increased over time, and in the second (and last) part became longer (relative to sham and right-side exposed condition).

![Figure 2. Second experiment: response times in ms for right-hand responses (right panel) and left-hand responses (left panel)](image)

**THIRD EXPERIMENT**

This experiment was designed to determine whether our previous findings should be attributed to RFR or to other agents. The main innovation of this study was the following: external remote antennas were connected to the cellular phones preventing any emission of RFR from the cellular internal antenna. The external antennas were placed approximately 2m away from the subjects, thus drastically reducing RFR exposure, but maintaining exposure to non-RFR factors such as non-RFR heating and low frequency magnetic fields unchanged.

A total of twenty nine subjects were randomly divided into 2 groups. The subjects in each group were exposed to only one of the two exposure conditions: left-side of the head (15 subjects) or right-side (14 subjects). Sham exposure was deemed unnecessary for this study. Since the second experiment revealed differences in RTs only during the first time segments, we concluded that three time segments, 50 trials each, are adequate.

As can be seen in Fig. 3, average RT of the right-hand responses under left-side exposure showed a trend for longer RT relative to the right-side exposure (by 104 ms) during the first time segment. These results are similar to those obtained in the second experiment, in which, apart from the RFR exposure, we employed an identical procedure (the difference between the exposure conditions, i.e. right side combined with sham exposure vs. left exposure was 146 ms, see Fig. 2). An analysis confirmed that the results of the present experiment were statistically indistinguishable from those of the second one.
SUMMARY
In summary, it was demonstrated that experiment duration, exposure side and responding hand may have major influence on the detection of cognitive functions related to the use of cellular phones. These parameters might explain the failure of certain studies to observe or replicate such effects. The inclusion of the above parameters in the design of any future experiment seems to be crucial. We have also found that some of the effects previously attributed to RFR can be the result of some confounders: non-RFR heating caused by the phone and low frequency magnetic fields originating from electric currents are possible explanations. Only by ruling out non-RFR agents as a source of an effect, can one argue that RFR can indeed influence the central nervous system or cause any other effect.

REFERENCES
Relative Response of Electric Field Probes to Pulsed Radiofrequency Radiation – Preliminary Results

I. Ben David, R. Hareuveny, I. Eliyahu, N. M. Yitzhak

Soreq NRC, Yavne, Israel

Corresponding Author: ilan@soreq.gov.il

INTRODUCTION
Pulsed Radiofrequency Radiation (RFR) is emitted by many apparatuses such as cellular GSM base stations, radars and avionics systems. According to major safety standards, such as ICNIRP 1998(1), assessment of human exposure to pulsed RFR requires the evaluation of both peak and average electric field amplitudes. Since there is no practical way of measuring the peak field, one can rely on average field measurements, and calculate the peak field by dividing the average field by the RFR duty cycle.

RFR safety measurements are usually performed with wideband electric field meters. Two major kinds of electric field probes (EFP) are commercially available for these meters, based on two different physical principles.

Thermocouple based devices are designated to measure the field true RMS, regardless of the pulse shape. However, thermocouple probes have many disadvantages (i.e. high susceptibility to ambient temperature, vulnerability to high fields, limited sensitivity and narrow dynamic range); therefore they are not commonly used.

As opposed to thermocouples, diode probes are much less vulnerable to these disadvantages. Thus, these probes are used for most commercial electric field meters and safety measurements. Consequently, diode probes which are known to be not suitable for pulsed RFR measurements are practically used in most cases.

Hence, we find it essential to assess the response of commercial EFP to pulsed RFR. In a series of experiments, we exposed EFPs to pulsed RFR and examined their response under different pulse parameters.

MATERIALS AND METHODS
The exposure system consists of an RF Signal Generator (HP 83640A, 10 MHz - 40 GHz), an RF Power amplifier (Amplifier Research 500W1000A, 80 MHz - 1 GHz, 500 W maximum power output) and a log periodic antenna (Amplifier Research AT1080, 80 – 1000 MHz). Six commercial field meters with isotropic diode based EFP were exposed to RFR (960 MHz, peak power density of ca. 5 mW/cm²) in an anechoic chamber. Their power density (S) readings were recorded. The maximal power output of the amplifier remained constant during the whole experiment.

Each probe was first exposed to continuous wave (CW) RFR and its reading was labeled as \( S_{cw} \). The probe was then exposed to five series of RFR pulses, each with a different pulse width (PW): 1, 10, 100, 1000, 10000 µs. During each series, the pulse repetition frequency (PRF) was gradually increased. The duty cycle (DC) and the \( S_{real} \) average power density were calculated using the following equations:

\[
\text{DC} = \frac{\text{PW[s] \cdot PRF[Hz]}}{
\frac{\text{DC} = \frac{\text{PW[s] \cdot PRF[Hz]}}{S_{real} [mW/cm^2]} = S_{cw} [mW/cm^2] \cdot DC}}
\]
The relative response (RR) of the probe for each set of pulse parameters (DC and PRF) was defined as the ratio between the measured field during exposure to these pulses ($S_{\text{meas}}$), and the real field ($S_{\text{real}}$):

$$RR = \frac{S_{\text{meas}}}{S_{\text{real}}}$$

**RESULTS**
The six EFP showed similar patterns; the relative response of two EFP of different manufactures are shown in Fig 1.

*Figure 1. Relative Response of two typical electric field probes as a function of the duty cycle for 5 different pulse widths*
DISCUSSION
As can be seen in fig. 1, the relative response of diode based probes to pulsed RFR is higher for shorter pulse widths. This phenomenon was found for all EFPs and duty cycles examined. For low duty cycles (up to ca. 1%) the relative response is rather flat. Afterward, it increases and reaches a maximum at duty cycle of 5- 50%. Finally the relative response gets to 100 % as the RFR pulses converge to CW. It seems that for shorter pulses the maximal relative response tends to appear at lower duty cycles.

For most EFP and pulse parameters tested, exposure to short pulses (1 µs) results in over response of up to few hundred percent. On the contrary, longer pulses (≥ 10 µs) with low duty cycles (up to 1÷10%) result in under response readings.

To summarize, any use of diode base probes for the measurements of pulsed RFR must be carried carefully. A specific correction factor must be applied for each set of pulse parameters. As a rule of thumb, our preliminary results suggest that due to their over response diode based probes can be used for hazard assessment of short pulses (1 µs). For long pulses, in the commonly used range of 10-1000 µs pulse width, correction factor of 2-4 might give a rough estimate of the actual average power densities. The above motioned conclusions are limited to the exposure parameters tested, i.e. frequencies around 1 GHz and peak densities of ca. 5 mW/cm².

REFERENCE
Ariel Mapping of Soil Radiation Contamination

Eli Eltsufin\(^1\), Shlomo Mark\(^1\), Ilan Yaar\(^2\)

\(^1\) Software Engineering Department, Shamoon College of Engineering (SCE), Ashdod, Israel  
\(^2\) Nuclear Research Center Negev (NRCN), P.O.Box 9001, Beer-Sheva 84190, Israel

Corresponding Author: elijaelt@gmail.com

One of the main uses of radiation aerial monitoring is in the mapping of an area contaminated by radioactive isotopes. In this work, an interactive computing tool that applied numeric methods to estimate the radiation activity level of a contaminated area, using air monitoring information will be presented.

1. INTRODUCTION

A survey of a radiation contaminated area can be conducted with hand held devices. The radiation monitoring is done in this survey by a person, who monitors the target zone and measures the radiation manually or by a land vehicle. The radiation levels are recorded along with their GPS location, and at the end of the process the data is sent for analysis. This process is quite accurate, but takes a long time to achieve and exposes the inspector to dangerous levels of radiation. To improve this process, aerial methods were introduced. The biggest advantage of the aerial method is the speed in which this process can be done (and the resulting coverage area) and the relative safety benefit of minimal radiation exposure.

![Figure 1 - Example for aerial information gathering](image1)

The main primary disadvantage of aerial measurements is its relative inaccuracy, as the sensor that measures the radiation receives a sum of readings emitted from the soil around the aircraft, so that specific measurements don't represent the real radiation underneath the aircraft.

![Figure 2 - Illustration of radiation readings](image2)

Altitude also affects on the measurement, resulting in decrease of aerial detector reading. Taking thus effect into consideration, the radiation level recorded by the detector can be defined by the general formula\(^3\),
\[ D = \frac{C(1 + kR)e^{-\mu R}}{R^2} \]

*\( C \) – Detector proportion coefficient [cps m²/gamma]

*\( K \) – radiation built up factor in air [m²]

*\( \mu \) – radiation absorption coefficient in air [m⁻¹]

*\( R \) – source to detector distance [m]

An examination of the problem in a matrix, for a measurement plane (M) and where we want to find the contamination plane (C), as demonstrated in Figure 4,

![Figure 3 - Matrix view of the problem](image)

The output of the current work is an algorithm that implements two different numeric methods of fixing the nuclear radiation contamination monitored information:

1. The algebraic Method, divides the target area into even cells, as demonstrated in Figure 5,

![Figure 4 - Target zone division](image)

and calculates the matrix \( D \cdot C = M \) in various numeric methods, So that if two identical solutions exist, there is a solution to the problem.

The following method is used in order to solve the problem \(^{[1]}\):

a. SOR (Successive Over-Relaxation) method

\[ x^{(k+1)} = (D + \omega \cdot L)^{-1} \cdot \left[ \omega \cdot b - (\omega \cdot U + [\omega - 1] \cdot D) \cdot x^{(k)} \right] \]

b. Jacobi method

\[ x^{(k+1)} = D^{-1} \cdot (b - R \cdot x^{(k)}) \]

c. Gauss-Seidel method

\[ x^{(k+1)} = L^{-1} \cdot (b - U \cdot x^{(k)}) \]

2. The Stein equation method, is based on a unique method of queue convergence \(^{[4,5,6]}\).

This method is implemented using a matrix as a parallel computation.

The structure of the equation is:

\[ X = A^T \cdot X \cdot A + Q \]
The application uses Shuttle Radar Topography Mission (SRTM\textsuperscript{[7,8]}; a mission that was launched by NASA and the German space agency in the year 2000). Mission data was freely published on the internet\textsuperscript{[12]}, but in poor resolution and with voids\textsuperscript{[9,10]}. To improve the continuity of the data, we used the Kriging method\textsuperscript{[11]} to improve the data resolution as seen in Figure 6.

\begin{figure}[h]
\centering
\includegraphics[width=0.5\textwidth]{srtm_kriging.png}
\caption{Direct SRTM reading and Kriging method implementation}
\end{figure}

2. RESULTS

The 3D engine algorithm was developed using C#.NET and DirectX\textsuperscript{[13]}. The application uses online internet satellite maps that are textured on the fixed Kriging solution, created in "real time" from the SRTM files (.hgt format), as shown in Figure 7.

\begin{figure}[h]
\centering
\includegraphics[width=0.5\textwidth]{real_time_texture.png}
\caption{Real time texture on Kriging}
\end{figure}

The application generates automatic instructions to the pilot, aimed for improving of the radiation information collection process, and processes the retrieved information via the algorithms detailed above. The research output is shown on the map, as shown in Figure 8.
Figure 7 - example for research results

Table 1 – Some research results for sea level terrain

<table>
<thead>
<tr>
<th>Detector parameters</th>
<th>Measurements</th>
<th>Algebraic method</th>
<th>Stein equation method</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.5</td>
<td>C</td>
<td>$M = \begin{pmatrix} 5 &amp; 2 \ 2 &amp; 1 \end{pmatrix}$</td>
<td>$C = \begin{pmatrix} 14386 &amp; -2988 \ -3278 &amp; -236 \end{pmatrix}$</td>
</tr>
<tr>
<td>0.75</td>
<td>k</td>
<td>$M = \begin{pmatrix} 1 \end{pmatrix}$</td>
<td>$C = \begin{pmatrix} 90 &amp; 24 \ 23 &amp; 14 \end{pmatrix}$</td>
</tr>
<tr>
<td>2</td>
<td>C</td>
<td>$M = \begin{pmatrix} 1 \end{pmatrix}$</td>
<td>$C = \begin{pmatrix} 89 &amp; 24 \ 42 &amp; 13 \end{pmatrix}$</td>
</tr>
<tr>
<td>0.01</td>
<td>μ</td>
<td>$M = \begin{pmatrix} 1 \end{pmatrix}$</td>
<td>$C = \begin{pmatrix} 60 &amp; -164 &amp; 80 \ -213 &amp; 793 &amp; -196 \ 92 &amp; -236 &amp; 120 \end{pmatrix}$</td>
</tr>
<tr>
<td>10</td>
<td>C</td>
<td>$M = \begin{pmatrix} 1 \end{pmatrix}$</td>
<td>$C = \begin{pmatrix} 10 &amp; -2 &amp; 0 &amp; -4 \ 0 &amp; 7 &amp; 2 &amp; -6 \ -10 &amp; 44 &amp; 7 &amp; 12 \ 20 &amp; -10 &amp; 12 &amp; -10 \end{pmatrix}$</td>
</tr>
<tr>
<td>100</td>
<td>k</td>
<td>$M = \begin{pmatrix} 1 \end{pmatrix}$</td>
<td>$C = \begin{pmatrix} 1325 &amp; 1156 \ 1113 &amp; 1320 \end{pmatrix}$</td>
</tr>
<tr>
<td>0.01</td>
<td>μ</td>
<td>$M = \begin{pmatrix} 1 \end{pmatrix}$</td>
<td>$C = \begin{pmatrix} 1507 &amp; -178 \ -200 &amp; 1 \end{pmatrix}$</td>
</tr>
<tr>
<td>10</td>
<td>C</td>
<td>$M = \begin{pmatrix} 1 \end{pmatrix}$</td>
<td>$C = \begin{pmatrix} 1462 &amp; 112 \ 79 &amp; 702 \end{pmatrix}$</td>
</tr>
</tbody>
</table>

3. DISCUSSION AND CONCLUSIONS

1. Radiation monitoring fixing methods

Use of Stein equation method in measurements where there is no equality $Q = Q^T$ is impossible. In matrices greater than 3x3 we didn't reach convergence in the construction of matrix $A$ (according to the method applied from the research). Uses of Algebraic methods rectify negative values. Matrices of size 5x5 and above cannot construct a combination of values of detector that produce only positive values.

2. Application

Kriging interpolation significantly improved the accuracy and continuity of the database in areas where ground resolutions above 90 meters are required. In areas where surfaces change radically, there were voids that produced inaccuracies. The database size of the SRTM mission is large (about 40GB). This type of database (files) allows splitting and reducing the amount of information by regional interest. In addition, the application enables performing topography-dependent studies, in a flexible manner across most of the planet (except the poles).

Also, using DirectX for the mapping engine allows for installing the application on any PC and doesn't require dedicated hardware manipulating it.
Finally, a big disadvantage (that reduces the flexibility of this method) is its dependence on the GMap.NET component, which requires an Internet connection to receive satellite maps in "real time". This problem can be solved by downloading Earth maps to a local computer, so that this solution requires a large data storage capacity.

REFERENCES
Environmental Radiation Monitoring Around the Soreq Nuclear Research Center


Radiation Safety Division, Soreq Nuclear Research Center, Yavne, Israel

Corresponding Author: datz@soreq.gov.il

INTRODUCTION

In accordance with the Radiation Protection Standard of the Israel Atomic Energy Commission (IAEC), the Soreq Nuclear Research Center (SNRC) must demonstrate that the annual dose to the most exposed member of the public as a result of its activities does not exceed a dose constraint of 0.3 mSv per year. This study describes the monitoring activities undertaken since 2006 and their results.

METHODS

Several sources of ionizing radiation are routinely working at SNRC, among them the 5 MW Research Reactor (IRR-1), the 10 MeV Cyclotron for the production of $^{18}$F (to be incorporated in radiopharmaceuticals), the production facility of radionuclides for nuclear medicine, the SARAF linear accelerator and a few radiochemical laboratories at the Radiation Safety Division. The environmental survey performed included characterization and monitoring of sources, monitoring of the effluents in air and wastewater, monitoring of groundwater, biota and food products and monitoring of the external dose on the fence of SNRC.

The environmental radiation monitoring includes the following activities:

1. Monitoring of emissions
   - Characterization of emissions from the stacks of the above-mentioned facilities and continuous monitoring of gamma radiation in the stacks.
   - Radioactivity monitoring of the wastewater released from SNRC into the sewage system after decay of the short-lived radionuclides.

2. Environmental monitoring
   - Radioactivity surveillance of agricultural products, including milk and crops produced in farms in the vicinity of SNRC (Kfar HaNagid and Bnaya).
   - Radioactivity surveillance of natural vegetation (foliage of White Acacia trees, which are typical of the SNRC environment).
   - Radioactivity surveillance of the soil close to the SNRC perimeter fence.
   - Radioactivity surveillance of the groundwater in the vicinity of SNRC, including potable water and treated wastewater used for irrigation.
- Radioactivity surveillance of particulate matter in the air.

3. Monitoring of external radiation levels using high sensitivity Thermoluminescence dosimeters (TLD) (LiF:Mg,Cu,P), at 11 points on the SNRC perimeter fence and at four control points (Yavne, Bnaya, Lehavim and Ra'anana).

4. Assessment of the dose to the most exposed member of the public in the vicinity of SNRC using well established dispersion and exposure models.

**RESULTS**

SNRC activities release into the atmosphere trace quantities of short-lived radioisotopes, such as $^{18}$F ($T_{1/2}=110$ min), $^{99m}$Tc ($T_{1/2}=6.02$ h) and $^{41}$Ar ($T_{1/2}=109$ min). The release of wastewater from SNRC into the sewage system is in compliance with the Regulations for Disposal of Radioactive Waste [2].

As expected, natural radionuclides were detected in agricultural products, in the soil and in the groundwater in the vicinity of SNRC. Their concentrations in the groundwater comply with the Regulations for Drinking Water Quality [3].

Air monitoring for radionuclides showed that natural radionuclides were detected, as well as $^{137}$Cs, originated from the Chernobyl accident in 1986 and the nuclear tests conducted in the atmosphere until October 1980. The average activity concentration of $^{137}$Cs (~1 $\mu$Bq/m$^3$) was found to be stable and similar to the average global concentration [4]. Its variation is mainly caused by changes in the local atmospheric conditions, as shown in Figure 1.

![Figure 1. $^{137}$Cs activity concentration variation with time as measured at SNRC](image)

Figure 1. $^{137}$Cs activity concentration variation with time as measured at SNRC
At the end of March 2011, several artificial radionuclides (\(^{131}\)I, \(^{134}\)Cs, and \(^{137}\)Cs) were detected by the air monitoring system following the Fukushima Nuclear Plant accident. The radioactive plume that dispersed over the whole northern hemisphere reached Israel on March 27, bringing detectable concentrations of fission products till the beginning of May, as shown in Figure 2.

**Figure 2** - Artificial radionuclides measured at SNRC following the Fukushima accident.

**Figure 3** - The dose rate in each of the nine measurement periods during 2008 to 2010, as recorded on SNRC perimeter fence and at 4 control points (Yavne, Bnaya, Lehavim and Ra'anana). The data represent weighted averages using the Least Squares method \(^{[5]}\) and the error bars are calculated for one standard deviation.
The radiation doses recorded on the SNRC perimeter fence were slightly higher than the radiation doses at the control points. However, due to a high level of uncertainty in the results, there is some overlap of the error bars of the fence points and the control points, as can be seen in Figure 3, which shows the average doses per day for each of the nine measurement periods during 2008-2010. The difference between the average dose on the SNRC perimeter fence and the average dose at the control points is $62 \pm 29 \, \mu\text{Sv per year (68% CI)}$. This finding is being further studied, but it could mean that the potential member of the public most exposed externally (the one living on the fence) may be exposed to an average additional dose of around $60 \, \mu\text{Sv per year}$. This additional dose is still lower by a factor of 5 than the dose constraint for members of the public according to the Radiation Protection Standard of the IAEC.

On the basis of the stack sampling measurements, annual activities released to the atmosphere were estimated and a conservative assessment of the exposure of members of the public was conducted. It was found that the dose to the theoretical most exposed member of the public in the vicinity of SNRC (a few hundred meters beyond the fence) is $6 \, \mu\text{Sv per year}$.

**CONCLUSIONS**

Environmental surveillance is routinely performed around SNRC according to the requirements of the IAEC Radiation Protection Standard.

Releases of trace amounts of short-lived radionuclides into the atmosphere were measured. The release of wastewater from SNRC into the sewage system was found to be in compliance with the Regulations for Disposal of Radioactive Waste.

Any artificial radionuclides linked to SNRC activities were not found in agricultural products, soil, natural vegetation and groundwater around SNRC.

The air is continuously monitored for radionuclides in particulate matter. Natural radionuclides are routinely detected, as well as $^{137}\text{Cs}$ at global activity concentrations. From March 27, 2011 till the beginning of May 2011 artificial radionuclides from the Fukushima accident were also detected at SNRC.

Measurements of external dose showed that the dose on the SNRC perimeter fence was slightly higher than at the control points. It is important to note that these findings refer to the dose on the perimeter fence and they do not represent the dose in the town and villages closest to SNRC, where no additional exposure was observed.

The radiation dose to the most exposed person in the vicinity of SNRC, due to stack releases from its facilities and taking into account conservative assumptions, was found to be lower by more than one order of magnitude than the dose constraint for members of the public according to the Radiation Protection Standard of the IAEC.

**REFERENCES**

Combined Fast-Neutron and Gamma Imaging with a Novel Liquid-Xenon Detector Concept

A. Breskin¹, I. Israelashvili¹, ², M. Cortesi², L. Arazi¹, S. Shchemelinin¹, R. Chechik¹, V. Dangendorf², B. Bromberger³ and D. Vartsky⁴

¹The Weizmann Institute of Science, Rehovot 76100, Israel
²Paul ScherrerInstitut (PSI), Villigen PSI 5232, Switzerland
³Physikalisch-Technische Bundesanstalt (PTB), Braunschweig 38116, Germany
⁴Soreq Nuclear Research Center (SOREQ NRC), Yavne 81800, Israel
⁵Nuclear Research Center of Negev (NRCN), Beer-Sheva 9001, Israel

Corresponding Author: israelashvili@gmail.com

INTRODUCTION

An important application of fast-neutron and gamma imaging technologies is in aviation and border security; their high penetration and elemental-discrimination capability permit screening of large objects. The general objective is to improve screening techniques by more efficient detection of contraband, including explosives, illicit drugs, illegal imports, weapons and nuclear materials (¹, ²).

High energy X-ray or Gamma inspection methods (radiography and tomography) provide high resolution images of shape and density, but they lack the capability to distinguish between organic materials of same density but of different chemical composition. Selectivity of high-Z elements can be achieved by using well defined dual-energy gamma radiation fields. In particular the $^{11}\text{B}(d,n;\gamma)^{12}\text{C}$ reaction provides well separated 15.1 MeV and 4.4 MeV gamma lines. This technique is called Dual-Discrete-Gamma Radiography (DDEG) (³). In addition to the discrete gamma rays also fast neutrons in a broad spectral range are produced (up to ~18 MeV).

Fast-neutron methods are a sensitive probe for low-Z elements like H, C, N and O, which are the main constituents of explosives and narcotics. In fast-neutron resonance radiography (FNRR) (⁴) two-dimensional (2-D) elementally-resolved images are obtained from fast-neutron radiographic images, taken at different neutron energies (1-10 MeV) chosen to cover the resonance cross-section features of those low-Z elements. A fast (ns) pulsed deuteron beam and the above reaction would permit neutron-energy selection by time-of light (TOF) techniques.

An inspection system featuring both fast-neutron resonance and dual-energy gamma radiography techniques will combine the capability of low-Z objects detection and substance-identification of FNRR with the high-Z selectivity of DDEG. This requires, to the intense fast source, an efficient imaging detector of fast-neutrons and gamma radiation. In this work we focus on a novel detector concept sensitive to both radiations, and evaluate its suitability.

The detection system is based on a liquid xenon (LXe) scintillator, contained in “fiber-like” capillaries, made in a suitable low-reflective-index bulk material (e.g. Teflon), to ensure scintillation-light propagation along the capillaries by total reflection. The cloud of primary scintillation photons emerging out of the capillaries is detected by a cascade Thick Gas Electron Multipliers (THGEM) (⁵, ⁶, ⁷), with the first element coated with an efficient CsI UV-photocathode. Localization of the neutron/gamma interaction point in the LXe converter medium is obtained by the center of the gravity (CG) of the detected photoelectrons. All the described setup, LXe converter and THGEM, should operate in cryogenic conditions, at a temperature of around 160 Kelvin.
LXe has a number of interesting features as radiation converter and scintillator. Its high density ($\rho \approx 3 \text{ g/cm}^3$), high atomic number ($Z = 54$) and elastic scattering cross section of few barns for fast-neutrons, ensure high neutron/gamma conversion efficiency with a few-cm thick device. LXe has high scintillation yield, of 46 photons per deposited KeV for gammas and around 7-9 photons per deposited KeV for fast-neutrons induced Xe recoils (8). It has a fast primary-scintillation decay-time, of 2.2 nsec; it is faster than other (crystalline) high-Z scintillators, with the exception of the much-lower yield fast component of BaF$_2$. Its UV-photon emission spectrum peaks at 178 nm, which perfectly matches the quantum efficiency (QE) of CsI photocathode; the latter value close to 30% at this wavelength (9).

A series of computer simulations were performed to prove the new detector concept and assess its expected performances, using the GEANT4 Monte Carlo (MC) simulation toolkit (version 9.3.2) (10). In particular, this work has been focused on the characterization and optimization of the operational principle and optimal geometry of the neutron/gamma LXe radiation converter coupled to the gaseous photomultiplier (GPM) - a CsI-coated double-THGEM. The assessed parameters were the spectra of deposited energy in the LXe-filled capillaries by radiation-induced charged particles (electrons and Xe recoils), generation of UV-scintillation, computation of UV-photon transport and collection at the photocathode, spatial distribution of the photoelectrons emitted from the photocathode, calculation of the expected spatial resolution and that of the overall detection efficiency. All the above parameters have been computed for neutrons and gammas in the relevant energy range of 2 to 14 MeV.

The chosen converter/detector geometry is depicted in Fig. 2, displaying a snapshot of a GEANT4 simulation run. The radiation converter consists of a stack of capillaries, made of Teflon (index of refraction $n=1.34$), filled with LXe ($n=1.61$). The 50mm long capillaries, arranged in a hexagonal pattern, are of 1mm diameter, with a 1.2 mm pitch. A 1mm gap maintained between the capillaries face and the 10 mm thick (UV-grade synthetic quartz) window (Fig. 2) allows LXe to circulate in the system during purification. The reflective CsI is deposited on the THGEM's top face, placed here 3mm from the other face of the window; the 5mm gap is filled with the Ne-based GPM’s counting-gas mixture (e.g. Ne:10%CF$_4$).
RESULTS

All the results will be provided in the conference presentation. Here we show only the detection and conversion efficiencies as well as the spatial-resolution results.

The detection efficiency of the full imaging system is defined as the number of particles (neutron/gamma) interacting in the LXe converter, including scintillation of which at least one photoelectron is emitted from the GPM's CsI photocathode - normalized to the total number of particles impinging on the detector. The conversion efficiency is the fraction of impinging particle interacting in the LXe capillaries; it depends solely on the elastic/Compton scattering cross section. In this context the main goal is to optimize the ability to focus and guide the scintillation photons along the capillaries, relying on total internal reflection. This ability strictly depends on the geometry, shape and arrangement of the capillaries, on the reflective index of the material of the capillaries support (Teflon in our simulations), as well as on the geometry and material of the UV-window. Notice that, since the geometry and the material of the capillaries affect also other parameters such as conversion efficiency and spatial resolution, the design and the geometrical arrangement of the converter are the result of a trade-off between various performance constraints. Figure 3 illustrates the calculated detection efficiency for neutrons and gammas at different energies; it is rather constant, of the order of 20% and 30% for neutrons and gammas, respectively; the conversion efficiency is shown as well. For gammas, due to the high energy deposition by Compton electrons and their resulting high scintillation yield, the detection efficiency equals to the conversion efficiency; that is to say, every gamma interacting in LXe, produces at least one energetic photoelectron that is detected by the imaging system. For neutrons, significant number of events release very small amount of energy in the liquid; thus, this process generates a small number of scintillation photons with a lower detection probability by the GPM. As a result, the detection efficiency for neutrons is a few percent lower than the conversion efficiency and not all neutrons interacting in the LXe converter are finally detected.
The reconstruction of the particle's impact position is obtained by calculating the center of the gravity (CG) of the cloud of photoelectrons detected by the GPM, originating from photons emitted by the activated capillaries. The detector's spatial resolution, obtained by computing, event-by-event, the CGs of the detected photoelectron distributions at the photocathode from many impinging particles, is shown in Fig. 4. The broad spatial distribution for neutrons (~10 mm FWHM) originates from their multiple-scattering in the converter walls and from low photoelectrons statistics due to the rather low recoil-induced energy-deposits and scintillation yields. The spatial distribution of gamma particles is narrower, of the order of 2-4 mm FWHM (depending on the gamma energy). The spread of the CG increases with increasing gamma energy, due to the energy-range correlation of the Compton electron.

CONCLUSIONS
A new concept is presented for combined neutron and gamma detection. It encompasses an efficient fast liquid-xenon (LXe) converter-scintillator coupled to a UV-sensitive gaseous imaging photomultiplier (GPM), incorporating a CsI photocathode deposited on a thick gas electron multiplier (THGEM). GEANT4 simulations were carried out to evaluate the expected performances of a first radiation-converter concept incorporating LXe-filled capillaries, with fast neutrons and gammas in the relevant energy range of 2 to 14 MeV. The simulated converter consisted of 50mm long, 1mm in diameter Teflon capillaries filled with LXe. Detection efficiencies of ~20% and 30% were calculated for neutrons and gammas, respectively. A spatial resolution of ~10 mm (FWHM) was derived for neutrons, limited in the present converter geometry by multiple-scattering events; the ~4 mm (FWHM) resolution calculated for gammas resulted from the range of energetic Compton electrons. Further simulation studies are performed with other capillary materials and geometries, as well as with plain converters (without fibers). The assembly of a LXe cryostat system is in course; it will permit investigating 100mm diameter detector prototypes with neutrons and gamma radiation.

ACKNOWLEDGMENTS
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Fissile Mass Estimation by Measuring the Number of Neutron Signals in Time Intervals

C. Dubi¹, T. Ridnik¹, I. Israelashvili¹, J. Bagi², and J. Huszti³

¹Nuclear Research Center of the Negev, P.O.B 9001, Beer Sheva, Israel
²Nuclear Security Unit, Institute for Transuranium Elements, Via E. Fermi, 2749 JRC, Ispra, Italy
³Institute of Isotopes, H-1525 Budapest, P.O.B 77, Hungary

Corresponding Author: israelashvili@gmail.com

INTRODUCTION

Spontaneous fission (SF) sources produce a known number of neutrons per second and per unit of mass. Hence, in case of no background, one can estimate the mass of a spontaneously fissile material by counting the average number of neutron detections in a given interval. However, the detection signal may be highly affected by other neutron sources. In particular, the measurements may be affected by the following sources:

1. Noise neutrons: mainly neutrons from \((\alpha, n)\) reactions occurring in the detected material.
2. Induced fission process: neutrons that are generated by induced fissions in the fissile material.

In order to accurately estimate the fissile mass, we must separate the neutron sources and solve equations for 3 unknowns: The fissile mass, the \((\alpha, n)\) rate and the multiplicity factor. Traditionally, the multiplicity distribution is measured using the Multiplicity Count method \(^{(1, 2)}\). The singles, doubles and triples are related to the three moments of this distribution and to the three unknowns. In order to get a small error in this method, the measurement system should have a high efficiency.

The method presented in this work uses the stochastic transport equation to express the mean, the variance and the third moment (skewness) of the number of neutron signals in time intervals, in terms of the time interval, the fissile mass, the \((\alpha, n)\) rate and the multiplicity factor. Each time interval (width of few msec) is a "measurement", so the three moments have relatively small error, thus relatively shorter measurement time is required. The measurement procedure is measuring the number of neutrons in a fixed time interval, for many time intervals, and calculating the three first moments. By solving a set of three equations, one is able to "isolate" the SF source from the other sources.

The idea of characterizing neutron detection signals by the moments of the detection signal was originally introduced by Hage and Cifarelli \(^{(3)}\). Thus, the true novelty of the present study is not in the derivation of the moments, but rather in constructing the appropriate set of closed, explicit and easily solvable equations in terms of the different neutron sources, and in applying a full methodology for mass assay by analyzing the moments.
Full description of the first three moments derivation and their relation to the fissile mass, the $(\alpha,n)$ rate and the multiplicity factor, can be found in (4) and will be presented in the conference presentation. Here we introduce only the resulting set of three equations.

(Eq. 1) \[
\frac{E(X)}{P_d T} = S(U(D_{sf,1} - 1) + 1) M_L
\]

(Eq. 2) \[
\frac{(V(X) - E(X)) \lambda}{P_d^2 (e^{-\lambda T} - 1 + \lambda T)} = M_L^2 \left[ UD_{sf,2} + \frac{M_L}{1 - D_{lf,1}} (U(D_{sf,1} - 1) + 1) D_{lf,2} \right]
\]

(Eq. 3) \[
\frac{(Sk(X) - 3V(X) + E(X)) 2\lambda}{P_d^3 (e^{-2\lambda T} + e^{-\lambda T} - 3 + 2\lambda T)} = M_L^3 \left[ UD_{sf,3} + \frac{M_L - 1}{1 - D_{lf,1}} \left( 3UD_{sf,2}D_{lf,2} + D_{lf,3} (U(D_{sf,1} - 1) + 1) \right) \right] + \left( \frac{M_L - 1}{1 - D_{lf,1}} \right)^2 D_{sf,2}^2 (U(D_{sf,1} - 1) + 1)
\]

Where $E(X)$, $V(X)$ and $Sk(X)$ are the average, variance and skewness of the number of neutrons in time intervals, respectively. $S$ is the source amplitude, $M_L$ is the multiplicity factor, $T$ is the time interval, $\lambda$ is the probability per time unit that the neutron will undergo any event (detected or lost). In the current terminology, the average lifetime of a neutron in the system is given by $1/\lambda$. $P_d$ is the probability that a spontaneous fission neutron is detected. $D_{sf,n}$ and $D_{lf,n}$ are the $n^{th}$ Diven factors for spontaneous and induced fissions, respectively. $U$ is the ratio between the SF source amplitude, $S_{sf}$, and the total source amplitude, $S$.

Equations 1-3 constitute a well defined set of equations in the unknowns $S$, $U$ and $M_L$. Since $S_{sf} = US$, after solving the set of equations one may easily compute $S_{sf}$ - and mass of the fissile material.

**RESULTS**

The method was implemented for measurements of five, well characterized, Pu samples, with various isotopic compositions (see table 1). The measurements made by the nuclear security unit JRC in Ispra, Italy, using PSMC system ($P_d = 0.55 \ (5)$) and by the Institute of isotopes, Budapest, Hungary, using the AWCC and JCC-31 systems ($P_d = 0.33 \ (6)$ and $P_d = 0.165 \ (5)$, respectively). The value of $\lambda$, for each system, was derived by fitting a single decaying exponent for short time intervals of Rossi-$\alpha$ distribution. The mean, variance, and skewness of each measurement, using time interval $T = 500$ micro sec., were calculated and corrected for dead time analysis. Solving equation 1-3 produced the Pu calculated mass. The mean, variance and skewness of every
sample, as well as the calculated $^{240}\text{Pu}$ masses by the SVM and multiplicity methods, are listed in table 2.

### Table 1. Description of the Pu samples.

<table>
<thead>
<tr>
<th>Sample number</th>
<th>Measurement system</th>
<th>Efficiency</th>
<th>Pu mass [gr]</th>
<th>$^{239}\text{Pu}$ [%]</th>
<th>$^{240}\text{Pu}$ effective [%]</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>PSMC</td>
<td>0.546</td>
<td>4.79</td>
<td>75.66</td>
<td>23.11</td>
</tr>
<tr>
<td>2</td>
<td>AWCC</td>
<td>0.33</td>
<td>20.57</td>
<td>69.95</td>
<td>26.3</td>
</tr>
<tr>
<td>3</td>
<td>AWCC</td>
<td>0.33</td>
<td>49.7</td>
<td>85.15</td>
<td>13.3</td>
</tr>
<tr>
<td>4</td>
<td>JCC-31</td>
<td>0.165</td>
<td>6.7</td>
<td>73.4</td>
<td>26.6</td>
</tr>
<tr>
<td>5</td>
<td>JCC-31</td>
<td>0.165</td>
<td>6.7</td>
<td>84.4</td>
<td>15.64</td>
</tr>
</tbody>
</table>

### Table 2. Experimental results.

<table>
<thead>
<tr>
<th>Sample number</th>
<th>Mean</th>
<th>Variance</th>
<th>Skewness</th>
<th>$^{240}\text{Pu}$ effective mass [gr]</th>
<th>Calculated $^{240}\text{Pu}$ mass [gr] (SVM)</th>
<th>Calculated $^{240}\text{Pu}$ mass [gr] (Multiplicity)</th>
<th>Relative error [%] (SVM)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0.313</td>
<td>0.587</td>
<td>1.138</td>
<td>1.108</td>
<td>1.045</td>
<td>1.023</td>
<td>5.68</td>
</tr>
<tr>
<td>2</td>
<td>1.71</td>
<td>2.294</td>
<td>3.462</td>
<td>5.41</td>
<td>5.51</td>
<td>5.43</td>
<td>1.9</td>
</tr>
<tr>
<td>4</td>
<td>0.257</td>
<td>0.289</td>
<td>0.381</td>
<td>1.78</td>
<td>1.47</td>
<td>1.45</td>
<td>17.5</td>
</tr>
<tr>
<td>5</td>
<td>0.121</td>
<td>0.145</td>
<td>0.196</td>
<td>1.045</td>
<td>0.912</td>
<td>0.897</td>
<td>12.7</td>
</tr>
</tbody>
</table>

As can be observed, the differences between the multiplicity and the SVM method are fairly small (less than 2%). It should be stated that the analysis of the measurements was not optimal, due to the fact that not enough calibration measured were supplied. In particular, we believe that the calibration of the neutron average lifetime was not enough accurate- which effected both methods.

### CONCLUSIONS

A new method for the estimation of fissile mass by a measurement of the number of neutron signals in time intervals was presented. The method uses the stochastic transport equation to express the first three central moments (mean, variance and skewness) of the number of neutron detections in a time interval in terms of the fissile mass, the $(\alpha,n)$ rate and the multiplicity factor. Solving the three equations allows estimation of the fissile mass. The two main advantages of the proposed method, compared to the multiplicity method, are the simplicity of implementation and the higher efficiency thus requires relatively shorter measurement time. The method was implemented successfully for several fissile mass measurements, which were conducted by the Nuclear Security Unit of the JRC in Italy and by the Institute of Isotopes, Hungary.
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6. www.antech-inc.com
Determination of 90Sr-90Y Activity in Urine Samples Using Cherenkov Counting

S. Tsroya1, O. Pelled1, U. German1, R. Marco1, E. Katorza1, Z. B. Alfassi2

1 Nuclear Research Center Negev, P.O.B. 9001, 84190 Beer Sheva, Israel
2 Ben-Gurion University of the Negev, 84105 Beer Sheva, Israel

Corresponding Author: tsroya1969@walla.com

INTRODUCTION

90Sr - 90Y is one of the main sources of internal exposure, and the common method used to estimate its uptake is by measuring urine samples. Based on international safety standards (1) the derived reference level for 90Sr-90Y is 2.2 Bq/L. Since the MDA (Minimum Detectable Activity) for direct urine measurement is higher than this level, special treatment of the urine sample must be employed. It mostly includes steps of concentration, chemical treatment, and preparation for counting. The disadvantages of the common methods are elaborate work, use of expensive materials and possible limited accuracy due to usually direct beta counting. The possible contribution of 60Co, fission products as 137Cs, or the natural 40K in urine, may also be significant.

Beta particles with energy in excess of 263 keV produce Cherenkov photons in aqueous solutions, which can be detected and quantified. Cherenkov radiation measurements can be performed by using the light detection features implemented in liquid scintillation counting systems. When measuring 90Sr-90Y activity in urine by Cherenkov counting, most of the Cherenkov radiation is produced by 90Y (about 98.6%, Emax=2280keV) due to the lower energy of the beta particles from 90Sr (Emax = 546 KeV). There is no chemical quenching when employing Cherenkov counting, however, the counting efficiency varies strongly with color quenching, at a greater extent than in standard liquid scintillation counting (2), and therefore proper quench correction techniques must be employed.

In a previous publication (3) it was shown that a quench correction method based on the external source of some Liquid Scintillation systems can be applied. The method (named ESAR- External Source Area Ratio) uses an indicative parameter based on the integral area of the spectra, and it was shown that it is superior to the common methods which use conventional indexes (4). In the present work, the application of the ESAR method for determination of 90Sr-90Y in human urine samples is described.

MATERIALS AND METHODS

A Quantulus 1220TM LSC system containing a 152Eu standard source, operated at Soreq Nuclear Research Center / Israel, was used for the present work. The ESAR method (3) includes two stages. In the first stage, the pulse height spectrum with and without the external source is measured. By subtraction, the net spectrum due to the external source is obtained. This spectrum area, relative to an unquenched sample, is used to determine the quench correction factor. In the second stage, the sample is counted for a preset time, chosen
to obtain the desired statistical accuracy. The activity is calculated from the spectrum obtained in
the second stage, by using the quench correction factor obtained in the first stage.

**Preparation of samples**

Part of the analytical method is based on oxalate precipitation described in the modified Volchok
method (5). The Beta emitting material from 250 ml urine samples was precipitated as calcium
oxalate, using ammonium oxalate in an acid medium. The urine was placed in a glass beaker
and Sulkoviz solution was added. The Sulkovitz solution was prepared from 25 g oxalic acid, 25
g ammonium oxalate and 50 ml conc. acetic acid. The volume was adjusted to 750 ml with
distilled water. After a few minutes of stirring the precipitate was allowed to settle overnight. The
next day, the supernatant was decant and discarded to waste. The precipitate was transferred to a
50 ml centrifuge tube and the centrifuge was operated at 2000 rpm for 5 minutes. The supernatant
was decant and discarded to waste. The sediment was dissolved with 20ml of 8 M HNO3 and
allowed to dissolve totally for 24 hours. After 24 hours the sample was transferred to a
scintillation vial for Cherenkov counting. Because of the solubility of the organic substances in
urine, most of them are removed during the separation process of the liquid residual urine from
the precipitation. The efficiency curve was prepared by adding varying amounts of sediments up
to four times the normal sediments content in 250ml urine. All reagents used were of analytical
grade. A calibrated liquid standard with specific activity of 4070 Bq/g of 90Sr in 0.1 M HCl
(Amersham Int.) was used.

**RESULTS**

**The contribution of 40K**

40K is present in the natural potassium in urine and produces Cherenkov counts. Therefore, its
contribution must be known and minimized. In order to evaluate the efficiency of 40K removal
by the different procedure steps, six "clean man" urine samples were prepared. Two samples of 20
mL volume from the original urine were counted directly, before performing any separation
procedure. Four samples of urine volume of 250 ml were passed the separation process as
described before, but two of them did not pass the centrifugation step. Table 1 shows the results
for the three phases.

<table>
<thead>
<tr>
<th>Phase</th>
<th>Sample</th>
<th>Count Rate [cpm]</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>A</td>
<td>1</td>
<td>60.62±8.75</td>
<td>*250 ml Urine sample without treatment.</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>58.12±8.50</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Average</td>
<td>59.37±14.25</td>
<td></td>
</tr>
<tr>
<td>B</td>
<td>1</td>
<td>4.04±0.64</td>
<td>250 ml Urine sample after separation and precipitance.</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>3.23±0.57</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Average</td>
<td>3.63±0.86</td>
<td></td>
</tr>
<tr>
<td>C</td>
<td>1</td>
<td>1.41±0.38</td>
<td>250 ml Urine sample after separation, Precipitance and centrifuge.</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>0.707±0.27</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Average</td>
<td>1.05±0.47</td>
<td></td>
</tr>
</tbody>
</table>

*20 ml measured and normalized to 250ml.

**Table 1.** The contribution of 40K to Cherenkov counting: A-without treatment, B-after
separation and precipitance, C- after separation, precipitance and centrifuge.
The separation after precipitance (phase B) reduces the count rate due to 40K by a factor of ~15. The last phase (C) originates a further reduction by a factor of ~3.5 to 1.05 cpm, which is almost the background level (distilled water counting).

**The efficiency curve for 90Sr-90Y counting**

For color quench correction of an unknown urine sample, a curve relating the efficiency to the change in the ESAR quench parameter due to quenching must be prepared. Samples were prepared in polyethylene vials with different quantities of urine sediments, achieving a wide quenching range. A fixed activity of 30.62 Bq of 90Sr was added to each sample, completing to a total liquid volume of 20 mL in each vial. All samples were counted for 10 minutes. The experimental data and the fitted function of the efficiency curve are shown in Figure 1.

The minimal efficiency is 1.36% sediment amount x4 times higher than in a typical urine. The maximum efficiency of 33.3% was obtained for a discolored sample (without sediment). As the color of urine changes with time, two of the most quenched samples were measured again after a number of days. The measurements are shown by the colored points in Figure 1, and it can be seen that they remain on the efficiency curve, thus time is not a factor in the preparation of the sample.

These additional points were not used for calculation of the fitted function, represented by a third order polynomial (equation 1), where I is the ESAR quench parameter (the integral count rate of the external source).

\[
\text{Eff}(\%) = 7.34 \times 10^{10} \times I^3 - 6.227 \times 10^6 \times I^2 + 0.0237 \times I - 1.0717
\]  

(1)
90Sr-90Y activity determination in urine samples

By using the quenching efficiency curve described above, the 90Sr-90Y activity in urine samples can be determined at any degree of quenching. For checking the accuracy of the method, eight urine samples with different quenching degrees were prepared, containing four values of 90Sr activity, from 2.2 Bq to 8.8 Bq.

As the preparation of the samples includes several chemical and physical processes (precipitation, separation), some activity may be lost in the preparation process and the “chemical efficiency” must be determined as well. It is usually done by using a known amount of carrier. For the process described in the present work, the chemical efficiency was found to be (91.6 ± 5.6)%.

The evaluated activities and the difference from the known values are presented in Table 2. The chemical efficiency was supposed 85% (minimum value) so that the evaluated 90Sr activities will represent upper limits. It can be seen that the activity determination for all activities and quenching values is within several percents from the target values, with a single exception of 25%, which may be due to experimental error.

<table>
<thead>
<tr>
<th>Actual activity [Bq]</th>
<th>Evaluated activity [Bq]</th>
<th>Activity difference [%]</th>
</tr>
</thead>
<tbody>
<tr>
<td>2.20±0.04</td>
<td>2.02±0.39</td>
<td>-8.18</td>
</tr>
<tr>
<td>2.20±0.04</td>
<td>2.18±0.59</td>
<td>-0.91</td>
</tr>
<tr>
<td>3.30±0.07</td>
<td>3.68±0.47</td>
<td>11.52</td>
</tr>
<tr>
<td>3.30±0.07</td>
<td>3.62±0.48</td>
<td>9.70</td>
</tr>
<tr>
<td>4.40±0.09</td>
<td>4.16±0.52</td>
<td>-5.45</td>
</tr>
<tr>
<td>4.40±0.09</td>
<td>4.24±0.49</td>
<td>-3.64</td>
</tr>
<tr>
<td>8.80±0.17</td>
<td>9.36±0.79</td>
<td>6.36</td>
</tr>
<tr>
<td>8.80±0.17</td>
<td>6.52±0.61</td>
<td>-25.91</td>
</tr>
</tbody>
</table>

Table 2. 90Sr activity determination

CONCLUSIONS

Determination of 90Sr-90Y activity in urine by Cherenkov counting produces reliable results, when applying the ESAR method for quenching correction. Thus, the method can be successfully applied in a radiotoxicological laboratory. The accuracy of the proposed method can be further improved by using a carrier for specific chemical efficiency determination, at the expense of additional work. However, it applies only for a few samples with results exceeding the recording level, for which an accurate activity determination is needed.

REFERENCES

Radioactive Content of EQ "Balance" Bracelets

S. Tsroya, O. Pelled, A. Abraham, T. Kravchik, U. German

Nuclear Research Centre Negev, P.O.B 9001, Beer Sheva 84190, Israel

Corresponding Author: tsroya1969@walla.com

INTRODUCTION

According to the regulations of the Nuclear Research Center Negev (NRCN), workers are requested to undergo routine Whole Body Counting (WBC) checks for detection of inner contamination due to possible un-noticed incidents. The results for one of these checks indicated an abnormal high activity of $^{232}\text{Th}$. The worker was called for an additional check to make sure the results are not erroneous, with the same outcome. The laboratory team was puzzled, as $^{232}\text{Th}$ is a natural occurring radioisotope and the detected activity was much over the normal values. The worker was invited to be questioned about his work and his whereabouts, to try to understand the source of the abnormal “contamination”. During the conversation it came out that the worker did not take off his bracelet before entering the WBC chamber (although he had to do it according to the rules) as it was “made of rubber and had no chance to contain any radioactivity”. By closer investigation it came out that the bracelet is an EQ product, which utilizes “holographic and negative ion technologies” to “improve body balances”, as advertized by its producers(1).

The suspicion fell on the bracelet and it was counted directly by a spectrometry system, supporting beyond any doubt the fact that it was the source of the enhanced radioactivity content. A third WBC counting of the worker, this time without the bracelet, gave results lower than the detection levels for thorium. As it turned out, this kind of bracelet is widespread worldwide and is worn also by teenagers and children. An immediate action was undertaken to evaluate the significance of the enhanced radioactivity and the Ministry of Health was informed about the issue. The present work presents the results of the radioactivity analysis of EQ “balance” bracelets.

RESULTS AND DISCUSSION

$^{232}\text{Th}$ is a natural radioisotope present in the human surroundings in rocks, soils and water. It decays by $\alpha$ emission with a very long half life and is the head of a chain on its name containing daughters, which emit also $\gamma$ radiation. In nature, $^{232}\text{Th}$ is in secular equilibrium with its daughters, and its activity can be determined by the daughters activity. The prominent $\gamma$ energy lines and the corresponding daughter nuclides are given in Table 1.
Table 1. Main gamma energy lines of the Th-232 series

<table>
<thead>
<tr>
<th>Parent nuclide</th>
<th>Daughter nuclide</th>
<th>$E_{\gamma}$ keV</th>
<th>$P_{\gamma}$ %</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{232}$Th</td>
<td>$^{228}$Ac</td>
<td>202.39</td>
<td>3.81</td>
</tr>
<tr>
<td></td>
<td>$^{212}$Pb</td>
<td>238.63</td>
<td>43.50</td>
</tr>
<tr>
<td></td>
<td>$^{228}$Ac</td>
<td>338.42</td>
<td>11.26</td>
</tr>
<tr>
<td></td>
<td>$^{228}$Ac</td>
<td>463.10</td>
<td>4.50</td>
</tr>
<tr>
<td></td>
<td>$^{208}$Tl</td>
<td>553.19</td>
<td>30.57</td>
</tr>
<tr>
<td></td>
<td>$^{212}$Bi</td>
<td>727.33</td>
<td>6.64</td>
</tr>
<tr>
<td></td>
<td>$^{208}$Tl</td>
<td>860.56</td>
<td>4.50</td>
</tr>
<tr>
<td></td>
<td>$^{228}$Ac</td>
<td>911.16</td>
<td>26.60</td>
</tr>
<tr>
<td></td>
<td>$^{228}$Ac</td>
<td>964.46</td>
<td>5.05</td>
</tr>
<tr>
<td></td>
<td>$^{208}$Tl</td>
<td>968.97</td>
<td>16.23</td>
</tr>
</tbody>
</table>

The Whole Body Counter of NRCN consists of four Ge detectors mounted in a iron shielded chamber (2). The system is a lung detector and is calibrated using a realistic phantom. The calibration assumes that the detected contamination is located in the lungs, following inhalation of contaminated air. In order to ensure that there is no other contamination sources, the workers undergoing WBC are requested to shower before the check and to wear clean clothes supplied by the laboratory.

When the enhanced activity was first detected, based on the lung calibration factor of the four HPGe detectors of the WBC, the total $^{232}$Th activity deposit estimate in the lungs was 22 Bq. In the second measurement performed to verify the result, a similar activity was obtained. Figure 1 presents the pulse height spectrum accumulated by WBC counting of the worker wearing the bracelet. For comparison, the spectrum obtained after removing the bracelet can be seen in Figure 2. Assuming lung contamination, the internal exposure for the aforementioned $^{232}$Th activity was estimated using the IMIE software taking into account the activity of all daughters in the $^{232}$Th series. The worker committed dose would have been in the range 10mSv to 35mSv, depending on the supposed input definitions to the IMIE program.

After interrogating the worker and suspecting the bracelet, it was measured directly by a gamma spectrometry system, equipped with a high resolution HPGe detector. The activity in the bracelet was estimated as 156.6 Bq $^{232}$Th and additional 82.4 Bq natural Uranium (Unat) were also found, both of them in a secular equilibrium with their daughters. The results are approximate, as there is no exact calibration factor for the bracelet geometry.

Further investigation revealed that the radioactive material is spread uniformly over the wristband rubber. The weight of the wristband is 15 g therefore the activity concentration is evaluated to be about 10 Bq/g and 5.5 Bq/g for the $^{232}$Th and Unat respectively.
In additional measurements of four other wristbands from the same manufacturer, an activity of the same order (sometimes even greater) of $^{232}$Th and $U_{\text{nat}}$ was found. A picture of a typical bracelets and the spectrum of its direct measurement obtained by Ge gamma spectrometry is given in Figure 3.
It is important to emphasize that the total activity of the wristband is greater than the $^{232}$Th and $^{238}$U activities mentioned above as the daughters, that are in secular equilibrium with their parents, contribute to the total activity as well. In order to estimate the dose to the wrist resulting from the bracelet, a wrist phantom and $3 \times 3 \times 0.38$ mm TLD100 chips (LiF:Mg,Ti) were used. The chips were placed between the phantom and the wristband for 66 hours. The dose rate to the wrist was estimated as about $3.4\mu Sv/h$ and the total dose when wearing the bracelet for a whole year is estimated as $30mSv$.

According to the international standards\(^{(3)}\), human radiation exposure must be justified. It is not the scope of the present work to discuss the potential effects of the aforementioned bracelets (the company itself states that there is no current scientific evidence of its effects\(^{(1)}\)) but it seems hardly plausible that the radiation exposure involved can be justified.

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1. Web site: [www.eq-love.com](http://www.eq-love.com)
INTRODUCTION

The experience gained during the activities for the elimination of consequences of the accident at the Chernobyl NPP and the recent accident at the Fukushima NPP has shown that the problem of effective radiation control in the territory contaminated by radionuclides has been and still remains a topical task. An absence of such a control just after the Chernobyl accident resulted in the distribution of the radioactive substances and radioactive contaminated materials far beyond a zone of direct contamination, and the process of the radiation distribution did not stop, but even increased. First of all, this was caused by a special composition of the emitted radioactive materials - mainly Cs-137 and Sr-90. The accumulation of cesium in the top soil layer resulted in its export with various foods, wood, saw-timbers, etc. beyond the evacuation zone and also zones with the increased radiation level for several years after the accident. This created threat to the health of the population of not only Belarus, Ukraine and Russia, but also of the neighboring countries, where these materials could be and were transported.

For this reason, the systems of the radiation control in the territories of countries exposed to contamination were created, and all kinds of products were controlled. The neighboring countries paid more attention to the border, or customs radiation control. This control was carried out at the border check points and was aimed at the prevention of transportation of radioactive substances and radioactive contaminated materials through the state borders. As an example, Poland was the first country which created the radiation control system along its entire borders because of a threat of penetration of the Chernobyl-originated radionuclides into its territory. One of the important results of such an almost total control was the detection of not only the Chernobyl radionuclides, but also the so-called orphan sources of various isotope compositions and activities, including sources of 3, 4 and even 5 categories according to the IAEA classification [1]. Besides, cases of the contraband and illegal transportation of especially dangerous nuclear and radioactive materials were recorded. A threat of possible use of these materials for terrorist purposes forced the states and international organizations to focus on this problem, and the Austrian government under the IAEA direction started the well known ITRAP program in 1996 [2]. The continuation of the international programs in this field is the ITRAP+10 program [3] which started recently and is funded by the EU and the US Department of Homeland Security. As a result, new equipment was developed, and fixed radiation monitors, portable search and spectrometer devices, and also so-called radiation pagers were manufactured.

RESULTS

The recent accident at the Fukushima NPP considerably differs from the Chernobyl one. The emission of radioactive materials as a result of destruction of a reactor was about 40 % of the Chernobyl emission, and the radioactive contamination of the surrounding territory was
incomparably lower. Nevertheless, the task of the radiation control is very important. Both in the most destroyed reactor zone, and in the adjacent territory there were abundant radioactive materials of high activity. These are not only a threat of irradiation of the personnel involved in the elimination of consequences of the accident, but could be also used for the terrorist purposes, for example, in the so-called dirty bombs. For these reasons, at such and similar accidents it is necessary to develop in shortest time the radiation control systems both for the personnel and population evacuated from the contaminated zone, and for all kinds of the transport. The purpose of this control is the prevention of casual and/or intentional distribution of nuclear and radioactive materials.

At present, the main system of the primary radiation control are the stationary portal monitors installed in sites which allow operators to scan effectively moved cargoes and pedestrians. However, to solve some specific problems there is a necessity for mobile monitors which can be quickly installed in different places.

In order to solve above mentioned problems the mobile gamma neutron monitor MDS 134A was developed by TSA Systems Ltd. [4]. This system was created first of all for the control of transportation of nuclear and radioactive materials which could be used for the terrorist purposes. The system can be installed at sites where there is a probability of similar acts, including sites of various competitions, shows, etc. Some pictures of the system are presented in Figures below.

1 – rear view of detector’s assembly 2 – front view of detector’s assembly 3 – position of detector’s assembly in a van
The MDS -134A van is fitted with shielded plastic scintillator detectors and helium-3 detectors for detecting both gamma and neutron radiation. It can detect relatively low energy emissions from plutonium and highly enriched uranium, as well as emissions from potential dirty bombs and isotopes used in medicine or industry. The detection area ranges from ground level to several meters in the air without a gap in detector coverage. Alarm notifications are displayed on a laptop PC located within the MDS van, but can also be displayed on a remote alarm panel placed somewhere outside the van.

Two plastic scintillator detectors provide approximately 28 liters of detector volume. Two He tubes (5 x 91 cm) are used as neutron detectors. Gamma sensitivity: the system will detect 1g of HEU or 10g of Pu-239 in 20 uR/hr background at a passage speed of 8 km/h, at a distance of one meter. Neutron sensitivity: the system will detect less than 200g of plutonium in a shielded container that reduces the gamma flux to 1% of the unshielded gamma flux. The rechargeable batteries provide 16 hours of continuous operation.

To complement the superior detection capabilities of the MDS -134A the laptop PC comes with the TSA Systems software RAVEN (Radiation Alarm and Video Event Notification). The RAVEN software not only alerts the monitoring personnel to an alarm condition, but also will display and archive graphs, data, and video images to assist in determining the specific location of the radioactive item. Analysis of select images and data can subsequently be relayed to response personnel in the field for possible secondary inspection, or other interdiction. These images and data can be shared between 3rd party agencies and/or systems for their records and analysis.

CONCLUSION

This system is designed to automatically scan vehicles, rail cars, air cargo bins and seaport shipping containers. It can be parked beside a road to scan vehicles as they drive by, or it can be driven past items to be scanned. Check points can be quickly established in the event that the threat of potential radioactive material movement has been identified. The MDS -134A can also be modified for special monitoring requirements.

REFERENCES

A Recent Development in the Field of Automated Biodosimetry; Overview of the RABiT System

R. Gonen, O. Pelled, M. Weinstein, U. German
Nuclear Research Center, Negev, P.O.B 9001 Beer Sheva, 84190, Israel.

Corresponding Author: rafigonen@bezeqint.net

INTRODUCTION

A mass casualty event is defined as one that involves injury to a significant number of individuals such that it exceeds the response capability of the local responders (1). When this type of event involves radiation, the result can be a large population, who may have received a range of radiation doses spanning from background levels to those large enough to cause medical consequences. These individuals need to be rapidly assessed for exposure levels to determine whether medical intervention is required (1).

Planning and preparedness is critical for an effective response to a mass casualty event. In the case of a radiation emergency the generic accepted guidelines include: a) establishment and training of local and national response teams equipped with critical equipment and supplies, b) knowledge and application of appropriate and available diagnostic approach for assessing radiation injury and dose, and c) access to reach-back reference laboratories, including expert laboratories for dose assessment by cytogenetic biological dosimetry (1). A critical component in the biological dosimetry ‘concept of operations’ is the process to prioritize the selection of samples for rapid cytogenetic triage-dose assessment that requires dynamic communication between the medical responders and reference cytogenetic biological dosimetry laboratory staff.

After a mass casualty radiation event, physicians are primarily concerned with preserving life and evaluating medical signs and symptoms for early treatment decisions. Several radiation exposure assessments, evaluated by an international consensus of experts, are applicable for early-phase acute radiation (1, 2). Depending on the radiation scenario and available resources, appropriate radiation assessment methods should be implemented in a mass casualty radiation emergency. Generic guidelines for the ‘concept of operations’ for first-responders in a mass casualty radiological incident are well described by IAEA resources (1). The implementation of a multiparameter biological dosimetry assessment approach in a mass casualty radiation emergency, however, can be a significant confounder without access to expert teams (1, 3).

Current radiation exposure assessment methods and emerging technologies can offer a potential to contribute to the dose assessment response. Research and development are needed to establish a diagnostic triage concept to facilitate a functional biological dosimetry concept of operations in a mass casualty radiation emergency (4). The initial screening radiation assay must be rapid (1 assay per min or less), use a hand-held device, and ideally involve a self-use test. Secondary and tertiary radiation assay may require more expertise and take longer (>1 day) for use but have higher radiation specificity.

Once identified as potentially exposed, patients may be recommended for biological dosimetry to provide confirmation of the suspected exposure and to determine a dose level. In the early-response phase of a radiation emergency, the initial purpose of cytogenetic triage is to rapidly estimate the dose for each potential exposure. Although these first dose estimates may not be very accurate, the goal is to quickly place the patient into one of the 4 dose ranges (1 Gy to 2 Gy, 2 Gy to 4 Gy, 4 Gy to 6 Gy and > 6 Gy), in order to provide timely information to the
medical community that can be used for patient treatment\(^{(5)}\). At this stage it is also possible to refute false positive samples due to symptoms such as vomiting from other causes. Partial-body exposures may also be identified at this stage.

Once the initial urgency of the requirement for rapid triage dosimetry has passed, the patients identified as having received significant doses can be further checked to provide more accurate dose estimates. Further follow up will continue also on those individuals who received very low or no doses, but still require reassurance. In addition, follow-up for epidemiological studies employing other techniques such as FISH will be required.

Over the past years, the Center for Minimally Invasive Radiation Biodosimetry at Columbia University has developed the Rapid Automated Biodosimetry Tool (RABiT)\(^{(6)}\). It is a completely automated, ultra-high throughput biodosimetry workstation, designed to analyze up to 30,000 samples per day for biodosimetry. The basic system involves two well-characterized assays with all the processing being carried out *in-situ* in multi-well plates: 1) the \(\gamma\)-H2AX assay is a direct measure of the number DNA double strand breaks (DSB) which are present and 2) the micronucleus assay quantifies radiation-induced chromosome damage expressed as post-mitotic micronuclei. Both assays have been shown to be effective in their biodosimetry characteristics and can be used for fast analysis of large populations in the event of a large scale radiological event.

**THE METHOD**

The \(\gamma\)-H2AX assay is a direct measure of the number of DNA double strand breaks (DSB) which are present (see figure 1). It measures DSB by immune-staining the phosphorylated H2AX histone. \(\gamma\)-H2AX yields are quantified by integrating the fluorescent intensity within each nucleus. This assay gives an instant result, but requires that the blood samples be available within about 36 hours from irradiation.

The micronucleus (MN) assay quantifies radiation-induced chromosome damage expressed as post-mitotic micronuclei (see figure 1). An important advantage of the micronucleus assay is that the signal is comparatively stable for some months post exposure, with a biological half life of about 12 months, so is no need for early acquisition of blood samples. Due to the required culture time, the analysis duration for this assay is approximately 70 hours. Samples collected within 36 hours of irradiation are due to be analyzed using the \(\gamma\)-H2AX assay. After this time, the RABiT can be switched over to micronucleus mode and all subsequent samples will be analyzed using the micronucleus assay.
**THE SYSTEM**

The robotic biodosimetry workstation consists of four main modules: centrifuge, cell harvesting system, liquid/plate handling robot and dedicated image acquisition/processing system (see figure 2).

**High-Throughput Imaging Systems for Biodosimetry.** Current automated imaging systems have limited throughput, mostly due to their non-specificity. The developers have therefore built a dedicated high-throughput imaging system for performing the γ-H2AX and micronucleus assays exclusively, seeking creative solutions for rapid sample manipulation, automated focusing and image acquisition and analysis. The throughput of the current imaging system is 6,000 samples per 20-hours. The developers are currently under advanced stages of design and component testing to upgrade this system to an anticipated throughput of 5-6 minutes/96-well plate or 20,000-30,000 samples/day.
**Image acquisition and processing.** Analysis of the image is performed online, as it is grabbed. By using dichroic mirrors and two cameras, attached to the same frame grabber board the nucleus and cytoplasm can be simultaneously seen and the images can be rapidly analyzed. A third camera, preceded by a cylindrical lens is used for monitoring the focus. Figure 3 shows the components of the image analysis system for the two different types of assays that can be performed.

**SUMMARY**

According to the criteria published by different researchers \(^{4,5}\) for the needs of the initial screening radiation assay rapidity, the amount of as 30,000 biodosimetric determinations per day makes the RABiT as an appropriate tool for the triage of a mass casualty event. The first results from \(\gamma\)-H2AX assay are obtained after 2 hours from the sample collection. The second biomarker tested by the RABiT (e.g. MN) also fits within the general frame, taken into consideration that the secondary and tertiary radiation assay may require more expertise and take longer (>1 day).

**REFERENCES**

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An Experimental Study of Indoor Dispersion of Radioactive Material


Nuclear Research Center Negev (NRCN), P.O.Box 9001, Beer-Sheva, Israel

Corresponding author: avish@nrcn.gov.il, avish@bgu.ac.il

The "Red House" (RH) project, conducted in Israel, aimed at increasing the preparedness for terrorism events where radioactive material is silently released inside a shopping mall or governmental facility. During the tests, a set of releases of 50 mCi's of liquid $^{99m}$Tc were conducted, and radiation levels were measured, on the ground and in the air. Special attention was given to the air ventilation system, and measurements were taken from the air filters. This paper describes the project objectives, the experimental setups, the results obtained during the tests and a preliminary analysis of them.

1. **“RED HOUSE” OBJECTIVES**

The main purpose of the “Red House” project was to be able to estimate the distribution of activity inside the building, on the floor and in the air. This data is important in order to revise the lineup and response to a silent terror activity, including the proper design of cost effective radiation detection systems for such buildings.

2. **EXPERIMENTAL SETUP**

The project included 6 dispersion tests, using 50 mCi’s of liquid $^{99m}$Tc which were released by an electric sprayer, and 4 tests in which TiO$_2$ powder was released either by a manual pump or by hand$^1$. The current paper will mainly concentrate on the $^{99m}$Tc dispersion tests. The tests were conducted inside a building located inside the IDF Home Front Command base. The building is designed as a small size shopping mall with two floors, where the size of the lower floor is 24x24 m$^2$, surrounded by small size shops and offices. The upper floor is designed as an open gallery surrounded by shops and offices, as well (see Figure 1).
The release was done from several different points, located on the first or on the second floor. The radioactive material (\(^{99m}\)Tc) was chosen due to its short half life (6.02hr), is rarely abundance in the background and the detectable gamma photon (141Kev) emitted during his disintegration process. High resolution spatial measurements were taken at the first floor (49 points on the floor and 12-15 in the air at a height of 1.2 m) and at the second floor (40 points on the floor and 4 points in the air). Radiation was measurements were conducted using a PDS (CsI crystal) device and a LnBr detector. Air sampling was done using a high (2 m\(^3\)/min) and low (2lpm) volume samplers.

Humidity, temperature and air ventilation parameters were kept fixed during all of the tests. Radiation measurements were also taken from the filters of the air condition system, at the roof of the building, and at a distance of 1 m away from the building using a high volume sampler. The radiation detection devices and the air volume samplers used during the experiments are shown in Figure 2.
3. RESULTS AND DISCUSSIONS

The spatial distributions of the detector readings on the floor surface deposition [in cps], as measured in two of the release tests done from the upper floor is shown in Figure 3. The point of release is designate by an arrow.

The distribution of the surface deposition [cps] along different cross release direction lines for the same two tests is shown in Figure 4. From these results, one can see that the distribution of activity around the point of release is different for the two tests. The points of release were opposite to each other and the air ventilation system was tuned to be symmetrical about the four sides of the building. While the "hot spot" in the left plot is quite concentrated (about 3800 cps), the one in the right plot is less intense (about 2200 cps) and more elongated. Although a symmetrical picture was expected, some asymmetry was revealed in both cases. The reason for that can be the air flow pattern inside the building. In order to have a picture of the air flow, smoke was released inside the building (as a pre-test) before the radioactive material release. The asymmetry in the air flow was discovered in these tests, and the radioactive liquid dispersion obeys to the same air flow pattern. In Figure 4 one can see the cross release direction of the activity distribution at different points away from the point of release.
Fig. 4: surface deposition [cps] along different cross release direction lines.

In both cases, there is a monotonic decrease from the center line of the plume, to the left and to the right. The lines of \( y = -3 \) (on the left plot) and \( y = 15 \) (on the right plot) are of homogeneous activity (~200 cps) in both of the cases. These lines were, both, 3 m behind the line of released. Measurements taken from the filters of the air condition system revealed high radiation level (~40,000 cps) in all of the tests. High volume air sampling results taken 1 m away from the filters shows almost no radiation (100-200 cps). The reason for that is the high efficiency of the HEPA filters in capturing the activity in the air, although the aerosols size was of the scale of a few microns. Based on that result, it is recommended to include radiation detection device in the ventilation system duct, next to the filters in order to get an early alert in a case of a radioactive material release.

4. SUMMARY

A set of six indoor radioactive material dispersion tests was discussed and the surface deposition pattern and air concentration distribution inside the building is presented. The results obtained in these tests show that the surface deposition is highly dependent on the air flow pattern and that the air condition filters are efficient in capturing the air activity.

REFERENCES

Outdoor Dispersion of Radioactive Material


Nuclear Research center Negev (NRCN), P.O.Box 9001, Beer- Sheva, Israel

Corresponding author: Avi Sharon, avish@nrcn.gov.il, avish@bgu.ac.il

"Green Field II" (GFII) project conducted in Israel during 2010-11 aimed at increasing the preparedness for possible terrorism events, where a radioactive material coupled to an explosive charge might be dispersed. The project included seven tests where ~6Ci of $^{99m}$Tc was dispersed by 0.25-2.5 kg TNT charges. The dispersion was done above clean or dirt surfaces, from the ground level or 1m above it. Comparison between the different situations will be shown as well as new parameterizations suggested to be included within the existing models. The paper will describe the project objectives, the experimental setups, some of the results obtained in these experiments and preliminary analysis of these results.

1. INTRODUCTION

The "Green field" I (GF-I) set of experiments had been conducted during 2008-9. This experimental plan included a set of experiments conducted with 0.5-100 kg of TNT, using radioactive simulants. Based on the results obtained in this plan, a new cloud rise model was written for 0.5-100 kg of TNT$^1$. The objectives of the "Green Field" II (GF-II) set of experiments conducted in 2010-11 are listed below:

- Measuring the surface activity concentrations around ground zero (GZ) and up to a few hundred meters down wind. All of the existing models are weak in very near area predictions.
- Study of the fireball–ground interaction by means of the amount of activity deposited on the ground in the very near area where the fireball “kisses” the ground.
- Measuring the vertical distribution of the activity inside the explosion cloud (up to the effective height).
- Measuring particles' size distribution after the explosion (an aero-solization study).
- Comparison between the predictions of some of the existing dispersion models (HotSpot, ERAD, LODI, RODOS, ARGOS) and the field results.
- Verification of the experimental results obtained during the GF-I plan.

2. EXPERIMENTAL SETUP

The GF-II project included 7 dispersion tests of 5-6 Ci's of $^{99m}$Tc that were dispersed using 0.25 and 2.5 kg of TNT. The shots were done above clean or dirt surfaces in order to study the effect of different amount of dirt entrained into the fireball created in the explosion. While clean surface avoid dirt entrainment into the fireball, dirt surface (packed sand soil) enhanced dirt entrainment and due to the high temperature exist during the early stages, agglomeration processes takes place and effect the final particles' size distribution.
The roll of different heights above ground level was checked and a comparison between ground level shots and shots conducted 1 m above ground was conducted. High resolution sampling process was used in order to get accurate radiation fields. Special attention was given to the very near area around the detonation point, called "Ground Zero" (GZ). About 40 measurement points were taken from the GZ area up to 7 times the fireball radii.

The radioactive concentration in the air was sampled in two points close to the GZ at 10 elevations. These measurements were taken in order to study the activity distribution in the explosive cloud up to the "effective" height.

Particles were collected (by stubs) in the GZ area. The stubs were analyzed by SEM & EDS for size distribution and for the identification of agglomerates of radioactive particles and entrained dirt.

The shots documentation was performed by 3 video cameras, 1 fast camera, 1 thermal camera and 1 stills camera. All of these cameras were remote controlled from the control chamber placed at a distance of about 800 meters from GZ.

Figure 1: The GZ area is surrounded by clean surface (left side) and dirt surface (right side).

Measurements were taken by using LnBr, PDS, HPGE, and beta surface detectors. Both shielded and unshielded measurements were done, in order to filter out the effect of the GZ "hot zone".

Figure 2: Portable LnBr detector with a lead shield plate (right) and a PDS detector (left).
3. RESULTS AND DISCUSSIONS

One of the intriguing questions, raised during this study, was how far and to what extent the influence of a RDD explosion used in this field study may be effective. The results of a shot performed in order to address this question are described in the next section. The shot was documented by video and thermal cameras. Using this documented data, it was shown clearly, that the contamination plume was carried by the wind in the SE direction. The radiation contamination on the ground was monitored by a portable LnBr detector, as seen in Fig.3.

Based on these measurements, a dominant direction of the ground contamination for that shot is clearly recognized in the SE direction. All of the data plotted in Figure 3 was corrected according to the decaying of the radioactivity. As shown in Figure 3, a significant reading is recorded 50 m upwind from GZ. The wind was clearly in the opposite direction for this shot, and the fireball size was only about 2 m so the entrainment zone due to the explosion was not exceeding 10 meters. Micro shield calculations, done in order to explain the high "upwind" radiation field, revealed that the GZ hot zone is responsible to this effect, by direct radiation or by the sky shine effect. This measured field reflects the radiation level that the first responders will be exposed to, but not the local ground contamination that has to be decontaminated in the later phases of such an event. For that reason, shielded measurements were included in all of the shots. A comparison between shielded and unshielded measurements is shown in Figure 4.
Fig. 4: Both, shielded and unshielded measurements for shots performed at ground level and at a height of 1 m above a surface covered by dirt.

4. SUMMARY

The results of several outdoor dispersion tests are presented and discussed. The results showed that the high level of radioactive material dispersed around the GZ area increases the detectors readings collected down and upwind from GZ. In order to filter out this influence, shielding was used around the detectors. Both, shielded and unshielded measurements are important in order to calculate the first responders' dose, and in order to plan the decontamination process, respectively.

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Rapid Methods for Screening Potentially Internal Contaminated Individuals in a Radiological Dispersion Event

E. Peri, T. Kravchik, N. Vainblat, I. Yaar

Nuclear Research Centre Negev (NRCN), P.O.B. 9001, Beer-Sheva, 84190, Israel.

Corresponding Author: Eyalp90@gmail.com

In a Radiological Dispersion Device event (RDD), thousands of people are expected to be internally contaminated. But, only a few of them will be highly internally contaminated and will need further medical treatment. Therefore, a rapid method for screening them is needed. In this paper several rapid screening methods that can be used in the triage phase following a RDD event will be presented and compared.

1. INTRODUCTION

In a Radiological Dispersion Device (RDD) event, thousands of people are expected to be internally contaminated with radioactive isotopes. Therefore, an initial rapid method for screening them is needed. It is assumed that only a few of these people will be highly internally contaminated and will be needed further medical treatment. Generally, internal dose assessment is based on measurement of bioassay samples or whole body counts. These screening methods are not suitable for a RDD event where, in the triage phase, there is a need for a rapid screening of potentially numerous people that were involved, in order to identify those who are highly internally contaminated and need further examination and treatment. In this paper, several alternative methods which enable a rapid individual screening after a RDD event are presented.

2. SCREENING METHODS

2.1 Nasal Swabs

Nasal swab analysis is a common method for rapid assessment of dose from internal exposure to solid radio-nuclides (alpha, beta and gamma emitters). The nasal swab is performed by collecting a smear sample from the anterior portion of the nose using a cotton-tipped swab. Radioactive material collected from the nostrils is assumed to represent ~10% of the inhaled radioactivity. Current models indicate that inhaled material is cleared from the anterior nasal passages with a rate that is represented by a physiological half-life of 17 hours (1). Thus, nasal swabs do not need to be taken immediately and can be taken at the triage phase 4-6 hours after the event (2). Swabs may sometime provide unreliable results, especially if they are collected only a few hours after the contamination event. The reasons for this are: nasal obstruction (stuffy/running nose), nose blow, cross contamination of the swab, person to person deviation in swab collection technique and the influence of different particle size distribution. Accordingly, positive results from the nasal swabs analysis indicate a potential intake of radioactive material by inhalation and may promote additional bioassay measurements. However, negative results from the nasal swabs analysis can be misleading and should be considered with special care (3).
2.2 Chest Screening with a Portable Detectors
Rapid assessment of the Committed Effective Dose Equivalent (CEDE) of individuals that are highly contaminated with gamma emitters, such as $^{137}$Cs, $^{192}$Ir, $^{131}$I or $^{60}$Co, can be performed easily with a simple and common portable equipment, such as gamma spectrometers and Geiger-Mueller detectors. Calibrating the detector reading into CEDE is complicated and requires Monte-Carlo simulation that takes into account the detector characteristics, the radiation attenuation inside the body and the radionuclide distribution inside the body as a function of time. For simplicity in the triage phase, data on the detector respond corresponding to 250 mSv (Clinical Decision Level-CDL) is given in most of the publications and procedures on RDD events. Any individual reading above one CDL requires further medical examination and treatment. Some procedures distinguish between adults and children because of the significant difference between the two in the detector reading for the same CDL.

The dose rate values at discrete times after inhalation of four gamma emitting radio-nuclides, for Micro Rem Survey meter (Plastic Scintillator) corresponding to posterior torso detector placement on male phantom receiving 250 mSv CEDE, is presented in Table 1 (4).

The count rate above background level taken in discrete time points after inhalation of the same gamma emitter radio-nuclides, for Ludlum Model 44-9 G-M Pancake Probe connected to Model 3 Survey Meter corresponding to posterior torso detector placement on male phantom receiving 250 mSv CEDE, is presented in Table 2 (5).

Table 1: Dose rate at discrete times after inhalation of four gamma emitters, for Micro Rem Survey meter (Plastic Scintillator) corresponding to posterior torso detector placement on male phantom receiving 250 mSv CEDE (4).

<table>
<thead>
<tr>
<th>Time (d)</th>
<th>$^{60}$Co ($\mu$Sv h$^{-1}$)</th>
<th>$^{137}$Cs ($\mu$Sv h$^{-1}$)</th>
<th>$^{131}$I ($\mu$Sv h$^{-1}$)</th>
<th>$^{192}$Ir ($\mu$Sv h$^{-1}$)</th>
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Table 2: Count rates taken in discrete time steps after inhalation of four gamma emitters radio-
uclides, for Ludlum Model 44-9 G-M Pancake Probe connected to Model 3 Survey Meter, corresponding to posterior torso detector placement on male phantom receiving 250 mSv CEDE

<table>
<thead>
<tr>
<th></th>
<th>ADULT</th>
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<td>Time</td>
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<td>I-131 (cpm)</td>
<td>Ir-192</td>
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<td>(cpm)</td>
<td>(cpm)</td>
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<td>10.00</td>
<td>170</td>
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2.3 Screening using Gamma Camera

Gamma Camera is an apparatus that can detect radioactivity in the form of gamma rays emitted by radioactive isotopes that have been introduced into the body as tracers. The position of the radioactivity source in this technique can be plotted and displayed on a TV screen. Gamma Camera is also an imaging system commonly found in hospitals and can be used for rapid screening of exposed individuals in a RDD event. The Minimum Detectable Activity (MDA) for five radionuclides \(^{(6)}\), is presented in Table 3. The MDAs of all five nuclides are lower then the corresponding ALIs (50 mSv). Thus, it is possible to use a Gamma camera for CEDE assessment of low gamma emitter like \(^{241}\)Am and pure beta emitters like \(^{90}\)Sr/\(^{90}\)Y (by measuring the Bremsstrahlung radiation).
Table 3: MDA for Siemens e.cam Fixed 18 (Gamma Camera) (6)

<table>
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<tr>
<th>Nuclide</th>
<th>AIf (Bq)</th>
<th>Backgroundb (cps)</th>
<th>Calibration factorc (cps/Bq)</th>
<th>Counting time (min)</th>
<th>MDA (Bq)</th>
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<tbody>
<tr>
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<tr>
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<td>1e+05</td>
<td>1.679</td>
<td>5.64E-03</td>
<td>4,367</td>
<td>3,086</td>
</tr>
<tr>
<td>Cs-137</td>
<td>6e+06</td>
<td>1.690</td>
<td>1.95E-01</td>
<td>127</td>
<td>90</td>
</tr>
<tr>
<td>Ir-192</td>
<td>8e+06</td>
<td>1.679</td>
<td>6.41E-01</td>
<td>38</td>
<td>27</td>
</tr>
<tr>
<td>Am-241</td>
<td>200</td>
<td>296</td>
<td>6.10E-02</td>
<td>171</td>
<td>120</td>
</tr>
</tbody>
</table>

a Annual limits on intake for occupational exposure via inhalation, based on most restrictive chemical form (Eckerman et al. 1988)
b Total counts from both detectors recorded in experiment
c Total counts from both detectors (see Table 3-1)

3. CONCLUSIONS
In a RDD event, it is essential to make an initial rapid assessment of the dose coming from internal radioactive contamination. This task can be achieved by using one of the following methods: nasal swabs, chest monitoring by portable detectors and Gamma Cameras.

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2. Key Element of preparing emergency Responders for Nuclear and Radiological Terrorism, NCRP 19, 2005.
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Mikro-Copter in Indoor and Outdoor Dispersion of Radioactive Material, "Green Field" and "Red House" Projects

I. Halevy, A. Sharon, E. Boublil, A. Hen, M. Ghelman, T. Ridnik and I. Yaar
NRCN, P.O. Box 9001, Beer-Sheva, Israel

Corresponding Author: halevyi@caltech.edu

1. INTRODUCTION -MIKRO-KOPTER

The Mikro-Kopter is the ultimate tool for aerial photography or security missions for short distances, times and low payloads.
An experimental study of indoor and outdoor dispersion of radioactive material was conducted as a part of "Green Field" and "Red House" projects.

The Mikro-Kopter used enables 20 min flights. The large arms angle allows wide angle photography and 300 gr of payload.

It has a stabilizing system which together with a GPS it can keep the location at a certain height, and automatically return to the take-off point. Using a computer interface it can communicate with the ground system at real time, sending it to predetermined landmarks (or enter them in real time). After the flight it returns automatically to the take-off point.

The Mikro-kopter's six powerful electric engines give the ability to simultaneously lift weight and keep high maneuverability.
The ground station includes all the functions needed to fly the Mikro-kopter and controlling the payload on it. No eye contact is needed but it will help for durability in hard missions.

Figure 1: The Mikro-kopter with its batteries and six electric propellers. A small camera is located as is payload.
2. THE "GREEN FIELD" AND "RED HOUSE" PROJECTS

As part of terrorism scenarios response, such as indoor or outdoor dispersion of radioactive material, "green field" and "Red house" projects were initiated.

The Green Field project deals with outdoor dispersion of radioactive material initiated by explosives. The complementary Red House project deals with indoor dispersion of radioactive material inside buildings like shopping malls or governmental offices facilities. A set of experiments was conducted in the Green-Field project, checking the influence of different variables such as: size and type of the explosives, amount of the dispersed material, micro meteorology and much more.

Other set of experiments was conducted during the Red-House project checking the influence of different variables such as: different location of the silent release, amount of radioactivity, micro meteorology (like temperature and airflow) and more.

In the dispersion we used stimulants as a first step, and as a second step $^{99m}$Tc liquid, which is a radioactive material. We choose the $^{99m}$Tc since its half life is only 6.0hr and its main gamma energy is 141KeV. Those parameters allow us measure few experiments in short time without severe contamination and safe work with low exposure to radiation. The Red-house experiments were conducted in the assigned IDF Home Front Command Facility, while the Green-Field experiments were conducted in the open desert.

Our explosion cloud rise model-formulated following the first phase of Green Filed project [1] was implanted in the DOE software HotSpot for Health Physics [2].

![Figure 2: The Mikro-kopter measuring the radioactive field in Green-Field project.](image-url)
2.1 Green-Field results

In one of the radioactive experiments during Green-Field 6Ci's of $^{99m}$Tc were adjacent to 250 gr of TNT.
After the explosion, several techniques of radiation detection were performed.
A PDS detector, CsI(Tl), was attached to the Mikro-kopter for Arial radiation field detection.
Two flights of 10 minutes each were performed in the scene.
The results of the aerial radiation field detection are shown in figure 3.

![Figure 3: The results of the aerial radiation field detection in Green-Field project.](image)

Results of the ground deposition radiation field detection in a different Green-Field test are shown in figure 4.

![Figure 4: Results of the ground deposition radiation field as measured by LnBr$_3$ detector](image)
2.2 Red-House results
In the Red-House experiments a dispersion of liquid material was conducted by using an electric sprayer from the first or second floor of the building into the main hall. The amount of activity dispersed in each test was 50mCi mixed with 3.5 liters of water. The building is designed as a small size, two floors shopping mall, where the lower floor is a 24x24 m² size, surrounding by small size shops/offices. The upper floor is an open gallery surrounded by shops/offices as well. Each set of experiments included three repetitions of dispersion tests from three different release points on the same floor. Two flights of 10 minutes each were preformed in the scene. The results of the aerial radiation field detection and the calculated contamination are shown in figure 5.

![Figure 5: The results of the radiation field detection and the calculated contamination in one of the Red-house experiments.](image)

The contamination was calculated from the measurements of the radiation field at 3m height by Ghelman procedure [3].

3. ADVANTAGES AND DISADVANTAGES IN THE MIKRO-KOPTER:

<table>
<thead>
<tr>
<th>Advantages</th>
<th>disadvantages</th>
</tr>
</thead>
<tbody>
<tr>
<td>• High mobility all-terrain vehicle</td>
<td>• High sensitivity to weather condition</td>
</tr>
<tr>
<td>• Monitoring large area in short time</td>
<td>• Short operation time</td>
</tr>
<tr>
<td>• Low exposure to personnel</td>
<td>• Short operation distances</td>
</tr>
<tr>
<td>• Automated monitoring</td>
<td>• Low weight payload</td>
</tr>
</tbody>
</table>

REFERENCES:
2. Steven G. Homann HotSpot, Health Physics CodeVersion 2.07.2 Lawrence Livermore National Laboratory, LLNL-SM-483991
3. M. Ghelman et. al , Privet communication (2011)
Radioisotopes Identification Algorithm Based on Generic Database

V. Bronfenmakher\textsuperscript{1}, A. Osovizky\textsuperscript{1}, V. Pushkarsky\textsuperscript{1}, D. Ginzburg\textsuperscript{1}, S. Mark\textsuperscript{2}, D. Khankin\textsuperscript{2}, G. Shilon\textsuperscript{2}, U. Wengrowicz\textsuperscript{3}, Y. Ifergan\textsuperscript{3}, S. Levinson\textsuperscript{3}, T. Mazor\textsuperscript{3}, and Y. Kadmon\textsuperscript{3}

\textsuperscript{1}Radiation Detection Department, Rotem Industries Ltd., Rotem Industrial Park, 86800, Israel
\textsuperscript{2}Shamoon College of Engineering, Beer-Sheva, 84100 Israel
\textsuperscript{3}Electronics and Control Laboratories, NRCN, Beer-Sheva 84190, Israel

Corresponding Author: bronfenv@rotemi.co.il

INTRODUCTION

The homeland security market is seeking for advanced radiation detectors with improved isotope identification capability. This need is increasing due to the operational difficulties derived by the high probability of innocent alarm shielding and masking scenarios. The identification algorithm should focus on discrimination between Naturally Occurring radioactive Material (NORM) and medical isotopes from industrial sources or Special Nuclear Materials (SNM)\textsuperscript{(1)} following the standards\textsuperscript{(2)} for HLS. The SpecIdentifier software application developed to provide spectrum analysis and identification solutions for various types of detectors. The key requirement is identification of radio-isotopes included in standards for spectroscopic radiation detectors (SPRD).

The process of isotope identification is based on spectrum analysis leaning on multi-parametric tests\textsuperscript{(3)}. The parameters include Peak-to-Compton ratio, FWHM (resolution), gain and other values. However, many obstacles are in the way to the proper determination of each parameter: Peak-to-Compton ratio varies with scintillator’s geometry; spectrum shape and resolution vary with scintillator’s physical properties; temperature dependency of scintillation light yield result in gain instability; non-proportionality in light yield as function of incident gamma energy; high count rates results in shift the calibration relation (gain)\textsuperscript{(4,5)}. These and other obstacles are considered by the proposed approach to the spectrum analysis and are implemented in the SpecIdentifier software.

METHOD

Isotope identification algorithms measure the degree of similarity between the test spectrum and reference spectra from an isotope database. Essentially all challenges in gamma spectroscopy involve perturbations in test spectra compared to the corresponding reference spectra.

A. Generic Database

In the proposed approach the database of reference spectra is being formed by two alternating processes – measurement or simulation. It is this capability, along with the special factorization procedure of each spectrum into high-important quantitative parameters, which make the database most generic. The generic database enables to overcome the listed obstacles by preliminary algorithmic manipulations on the test spectra. Classification of the spectra according to the specific measurement parameters improves the identification process, decreases the probability of the false isotope identification and reduces the database look-up time. Fig. 1 presents the database formation and spectra manipulation flow. The ability to add simulated spectra to the database enhances the flexibility of the radionuclide library. The
database initially includes list of about 40 isotopes from the SPRD standard\textsuperscript{(1)}. Special correction function is generated and applied on the simulated spectra in order to keep the simulations with agreement to the measurements. This function is generated based on parameters that are derived from comparative observation of measured templates and simulated spectra. Background removal, intensity leveling and spectrum composition enables the creation of possible spectrum patterns with mixed isotopes that are useful to get principal spectrum view.

B. Identification

An accurate peak detection process is of the primary priority for correct isotope identification. Three different peak search algorithms are implemented in the \textit{SpecIdentifier}: the Savitzky-Golay algorithm; derivation in frequency domain; and verbose peak search in frequency domain\textsuperscript{(6)}. Such algorithmic redundancy enables the user to optimize the peak search process and to apply the most suitable algorithm for each particular case. The implementation of the Savitzky-Golay algorithm allows best smoothing of the spectrum, which is required to eliminate statistical fluctuations and electronic noise. Application of Fast Fourier transform to the analyzed spectrum enables to decrease the derivation algorithm complexity and as a result to increase the process speed. The verbose algorithm is intended to prevent from the derivation peak search method to mismatch in case of a presence of low rate peaks adjacent to high rate peaks in the spectrum. In this case the spectrum range was divided into a number of windows with constant size and the derivation process took place in every window discretely. An advanced version of the verbose peak search algorithm with variable window size is also available and currently is being evaluated. The implemented stripping mechanism enables the detection of hidden peaks in case of masking scenario. When the identification process is completed a table of suggested radionuclides with matching characteristics is shown. At this stage the user is given the option to strip the spectrum of each radionuclide according to the templates that exist in the database and then to force a new run of the peak search and identification algorithms on the resulted spectrum.
RESULTS

The preliminary tests of the software included measurements of various radionuclides from different categories that were performed with LaBr₃(Ce) scintillation detector (Fig. 2). Table 1 summarizes the results of the different peak search algorithms. The algorithms detected the necessary peaks in most of the cases without any user interference. As it was expected, the over-smoothing of the spectrum can eliminate the low statistics peaks. The verbose peak search method showed the most significant results.

Figure 2. SpecIdentifier: Spectrum analysis tool
Table 1. Peak detection results

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Reference peaks amount</th>
<th>Savitzky-Golay</th>
<th>Derivative in frequency domain</th>
<th>Verbose</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{57}$Co</td>
<td>2</td>
<td>1/1*</td>
<td>1/0</td>
<td>1/1</td>
</tr>
<tr>
<td>$^{67}$Ga</td>
<td>6</td>
<td>6/2</td>
<td>4/0</td>
<td>6/2</td>
</tr>
<tr>
<td>$^{99m}$Tc</td>
<td>1</td>
<td>1/1</td>
<td>1/0</td>
<td>1/1</td>
</tr>
<tr>
<td>$^{133}$Cs</td>
<td>2</td>
<td>1/4</td>
<td>1/1</td>
<td>1/3</td>
</tr>
<tr>
<td>$^{133}$Ba</td>
<td>8</td>
<td>4/1</td>
<td>3/0</td>
<td>4/1</td>
</tr>
<tr>
<td>$^{60}$Co</td>
<td>2</td>
<td>2/3</td>
<td>2/1</td>
<td>2/3</td>
</tr>
<tr>
<td>$^{226}$Ra</td>
<td>&gt;20</td>
<td>6/1</td>
<td>0/20</td>
<td>6/1</td>
</tr>
<tr>
<td>$^{152}$Eu</td>
<td>13</td>
<td>8/2</td>
<td>3/2</td>
<td>8/2</td>
</tr>
</tbody>
</table>

*True/False amount of peaks detected by the algorithm

Table 2. Radionuclide identification results

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Savitzky-Golay</th>
<th>Common derivative</th>
<th>Verbose</th>
<th>False identified radionuclides</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{57}$Co</td>
<td>√</td>
<td>√</td>
<td>√</td>
<td>$^{99m}$Tc</td>
</tr>
<tr>
<td>$^{67}$Ga</td>
<td>√</td>
<td>-</td>
<td>√</td>
<td>$^{57}$Co, $^{99m}$Tc</td>
</tr>
<tr>
<td>$^{99m}$Tc</td>
<td>√</td>
<td>√</td>
<td>√</td>
<td>$^{57}$Co</td>
</tr>
<tr>
<td>$^{137}$Cs</td>
<td>√</td>
<td>√</td>
<td>√</td>
<td>$^{57}$Co, $^{241}$Am</td>
</tr>
<tr>
<td>$^{133}$Ba</td>
<td>√</td>
<td>√</td>
<td>√</td>
<td>$^{99m}$Tc</td>
</tr>
<tr>
<td>$^{60}$Co</td>
<td>√</td>
<td>√</td>
<td>√</td>
<td>-</td>
</tr>
<tr>
<td>$^{226}$Ra</td>
<td>√</td>
<td>-</td>
<td>√</td>
<td>$^{241}$Am, $^{133}$Ba</td>
</tr>
<tr>
<td>$^{152}$Eu</td>
<td>√</td>
<td>-</td>
<td>√</td>
<td>$^{99m}$Tc, $^{57}$Co, $^{54}$Mn</td>
</tr>
</tbody>
</table>

The identification results that are presented in Table II can vary with number of isotope patterns saved in the database. The full automatic identification algorithm is not implemented yet in the software. Thus the user should give it his considered opinion and activate deliberate selection criteria for eliminating non-possible candidates from the results list. As it is shown in [5], an oversized template library may result in an overload of the identification algorithm and possible algorithm mismatches. At this stage only the linear regression analysis is in use but other possible solutions are being considered as well.

Figure 3. Comprehensive simulation of $^{67}$Ga and $^{235}$U measurements with 3”x3” NaI(Tl)
Additional example of the identification challenge is confronted by masking and shielding scenarios. Masking scenario is practically the identification of isotopes with overlapping peaks that are caused by mixed radionuclides with multiple gamma energy lines. Figure 3 presents comprehensive simulations of $^{67}$Ga and $^{235}$U with 3”x3” NaI(Tl) detector, where possible masking of SNM isotope, $^{235}$U, by medical isotope, $^{67}$Ga is shown. In this particular scenario, the precise peak detection process is of primary priority for correct isotope identification. The peak at 185 keV is typical for both, $^{67}$Ga and $^{235}$U spectra.

II. DISCUSSION

The generic spectrum database along with various peak search and spectra processing algorithms provides a complete solution for analysis of spectra measured by varied detectors and in different measuring terms. The SpecIdentifier enables a non-professional operator to define specific experimental setup and to visualize the measurement configuration. Supplementary interface tool was created to assist non-professional users to build appropriate simulation models. The future work objectives include covering a wider scope of detectors, optimization of the software and database completion.

REFERENCES

Simulation of Gamma Spectra for Spent Fuel

S. Levinson¹, B. Sarusi¹, O. Pelled¹, U. German¹, Z. B. Alfassi²

¹Nuclear Research Center Negev, P.O.B. 9001, 84190 Beer Sheva, Israel
²Ben Gurion University of the Negev, P.O.B. 653, 84105 Beer Sheva, Israel.

Corresponding Author: sea-and-spa@013.net

INTRODUCTION
"GammaGen" is a Windows software, developed at The Nuclear Research Centre Negev (NRCN) to generate synthetic gamma ray spectra obtained with various detectors. The information on a chosen radio-nuclide is generated according to its activity, its photo peak energy and yield (taken from a data library). The detector efficiency and resolution, and peak to Compton dependence are used to generate a pulse height spectrum for a specific detector. The spectra can be displayed in several modes: as energy lines of the photo peaks, or as Gaussian of each photo peak. The Compton continuum can be included as well. A spectra mixture for different sources can be produced for visual analysis. Spectra can be exported to some other formats, to enable input to other commercial spectra analyzing programs. The main purpose of the program is to generate synthetic distributions for predicting complicated pulse height spectra. In the field of radioactive waste follow-up it can be used to evaluate the efficiency of different shields and the time dependant monitor response at the waste site.

THE METHOD
Arbitrary radio-nuclides mixture
For a general application, any radio nuclide mixture can be chosen by manual selection from a nuclides library list. The activity of each radio-nuclide is given as input. The detector type, as NaI(Tl) or Ge, and the display method (energy lines, Gaussian, Compton distribution or full spectra) must be chosen. The results are displayed on the computer screen.

Fission products spectra
Such spectra, characteristic of spent fuel, are complicated and include more than hundred radioisotopes, each emitting mostly several photon energies. The composition of the fission products inventory depends on the operational parameters of the reactor (reactor power and operation time) and on the cooling time. The gamma ray energy spectrum reaching a detector placed near a spent fuel site is dependent also on the distance from the site and on the shield and its thickness. The detector parameters must be defined in the input data for calculation of the detector response.

The total inventory, taking into consideration decays and buildup of the fission products is calculated for the main γ-emitting fission products. The calculations are based on data published in the literature¹. The fission products can be divided into several groups according to their physical properties. The different groups are characterized by release fractions, which are the basis for risk evaluations in case of a nuclear accident.

RESULTS FOR SPENT FUEL SHIELDED SITE
Information on the program operation can be found elsewhere\(^2\). The computer code was applied to predict the NaI(Tl) detector response near a spent fuel site. The gamma radiation reaching the detector is determined by the inventory of the fission products which depends on the reactor operation time, the power level, the cooling time and the shielding. A representative case of one year operation at 1 MW level was taken. A concrete shielding was supposed. Calculations were performed for cooling times of 0, 30 days and one year, and for a concrete shielding thickness of 0, 20 cm and 50 cm.

The response of a virtual Na(Tl) detector after one month and one year cooling times and for 20 cm and 50 cm concrete shielding thickness is presented in Figures 1-3. Although the experimental setup of a non-shielded detector close to the spent fuel site is of no practical meaning due to the high radiation fields, the theoretical shielding efficiency and the time-dependant gamma energy composition can be analyzed based on the results of the simulated response.

![Graph](image)

Figure 1. Simulated NaI(Tl) pulse height spectrum for 30 days cooling time and 20 cm concrete shielding.

The outstanding radio-nuclides in the spectrum after 30 days cooling time for a concrete shielding thickness of 20 cm (Figure 4) are \(^{95}\)Zr and \(^{95}\)Nb. They have main peaks at close energies (at 724 keV, 756 keV and 765 keV) which appear as a common peak due to the low resolution of NaI(Tl) detectors. This common peak includes also a small contribution from \(^{140}\)La, which can be identified better through its isolated peak at 1596 keV. \(^{103}\)Ru, with some added contribution by \(^{140}\)La and \(^{140}\)Ba, is represented by a peak at about 500 keV with a height of about one third of the \(^{95}\)Zr + \(^{95}\)Nb peak. Some contribution in the low energy region can be also observed, by the presence of the peaks from \(^{141}\)Ce and \(^{144}\)Ce. It is to be stressed, that the spectrum in Figure 1, as well in the forthcoming figures, presents only the outstanding peaks which are
observable at the given scale. By increasing the Y scale sensitivity many other peaks will be distinguished.

When increasing the concrete thickness to 50 cm for the same cooling time of 30 days (Figure 2) the maximal peak height ($^{95}\text{Zr} + ^{95}\text{Nb}$) is reduced by a factor of about 150 due to the additional absorption. Besides, there are changes in the spectrum due to the different absorption of higher and lower energy gamma rays. The prevalent contribution is still by the $^{95}\text{Zr}$ and $^{95}\text{Nb}$ radioisotopes, but the higher energy peaks of $^{140}\text{La}$ become much more pronounced. Some additional peaks of $^{140}\text{La}$ can be observed above 2000 keV as well. On the other hand, the peak due to $^{103}\text{Ru}$ at 497 keV becomes relatively smaller, and the peaks due to Cerium are not observed anymore on the given scale.

Figure 2. Simulated NaI(Tl) pulse height spectrum for 30 days cooling time and 50 cm concrete shielding.

Figure 3 presents the spectrum for the cooling time of one year, with a concrete shielding of 20 cm. By comparing this spectrum with the one given in Figure 1, it can be seen that the maximal peak height was reduced by a factor of about 25 due to the additional decay. The peak with maximal height belongs still to $^{95}\text{Zr}$ and $^{95}\text{Nb}$, although a new peak due to $^{137}\text{Cs}$ turns significant due its long half life (about 32.5 yr) relative to that of $^{95}\text{Zr}$ and $^{95}\text{Nb}$ (half times of about 64 d and 35 d respectively). Within the maximal peak, the relative contributions of $^{95}\text{Zr}$ and $^{95}\text{Nb}$ change, as $^{95}\text{Zr}$ undergoes only decay, while $^{95}\text{Nb}$ is continuously produced by the decay of $^{95}\text{Zr}$ in addition to its own decay. Some additional radioisotopes can be observed in the spectrum at the given scale, as $^{106}\text{Ru}$. 
CONCLUSIONS
The developed program is a helpful tool for graphical inspection of complicated spectra, especially for the case of the poor resolution NaI(Tl) detectors. It is especially useful for the case of many photo-peaks present at a radioactive waste site. The present work describes an application to predict prominent peaks in the gamma spectra of spent fuel, for different combinations of reactor operating time, cooling time and shielding. The program is still under development, and additional features are intended to be added, as combining the calculated spectra with measured background, adding sum-peaks and escape peaks, and taking into consideration the scattered radiation.

REFERENCES
An Application for User-Friendly MCNP4 Operation

B. Sarusi¹, S. Levinson¹, V. Pushkarsky², A. Osovizky², I. Orion³, U. German¹

¹ Nuclear Research Centre Negev, P.O.B. 9001, Beer Sheva 84190, Israel
² Health Physics Instrumentation Department, Rotem Industries Ltd, Beer Sheva, Israel
³ Ben Gurion University of the Negev, Beer Sheva 84105, Israel

Corresponding Author: Benjamin.sarusi@gmail.com

INTRODUCTION

The MCNP¹ is a popular program used in the field of nuclear physics and radiation detection. One of its widespread uses is to predict detectors response to different radiations for various geometric configurations. An input file containing the geometric data and materials specification must be prepared for running the code. This input file can be quite complex, depending on the needed configuration, and must be carefully checked to avoid erroneous results, requiring a certain amount of expertise.

In order to make the data preparation more user-friendly and to prepare automatically the input file, thus avoiding errors and facilitating the needed work, a special application was developed. It is intended for radiation detectors response calculation for a variety of detector types, configurations and geometries, including shielding options. The detector type and configuration and the radionuclide combination can be easily defined through a user-friendly screen, while the associated basic data (as energies, cross sections, etc.) are automatically added to the input file from a proper database, defined in the input screen. After the MCNP run was ended, the application handles the results file as well. The calculated spectrum is presented on the screen for visual inspection and/or use. The developed application is especially useful when a series of MCNP runs is needed to check the sensitivity of the detector response to a certain parameter, which may require a lot of work and checks for preparation of the input file to ensure an error-free and complete set of results.

THE METHOD

The information needed for preparing the input file is extracted from the following sources:

1. Radioisotopes database consisting of:
   a. A library containing energies and yields of the various radioisotopes.
   b. Specific definition of groups or clusters of radioisotopes, like fission products, NORM, etc.

2. Operating templates which prepare a customized form needed for the input data to MCNP. The template provides the following functions:
   a. A short description of the input parameters.
b. Equation for calculation of different data needed for input.
c. The format needed for input to MCNP.
3. Constants and fixed data to be uploaded.
4. Configuration files which define the parameters to be uploaded.

THE OPERATION OF THE APPLICATION

The operator has to choose the template and to fill the information in the different active windows. The library of isotopes and the specific group definition are to be chosen, and the radioisotopes relative activities are to be defined. After running the MCNP the resulting spectrum can be viewed and output to EXCEL format. The operation is through a main screen, an example of which is given in Figure 1. The prepared MCNP input file is also displayed on the screen for checks, if needed.

![Figure 1](https://via.placeholder.com/150)

Figure 1. The main input screen of the application showing also the resulting MCNP input file.

An additional geometry window is intended to input the dimensions and form of the source. Calculations are performed to determine auxiliary dimensions and different physical parameters.

The templates data can be viewed before running MCNP to ensure correct values. The information is concentrated on a screen, split into three sub-screens, as seen in Figure 2. The left sub-screen shows the configuration. The middle sub-screen shows the equations employed in the templates and their resulting values. The right sub-screen shows the geometrical parameters given in the input.
RESULTS OBTAINED BY USING THE APPLICATION

Figure 4 presents an example of using the application for a great amount of calculations. It describes the dependence of the counting efficiency on the volume of a cylindrical source with a constant diameter of 13.5 cm. The source density was supposed 2 g/cm³ with a fixed radioisotopes concentration. The calculations were performed for three representative energies. It can be seen that by increasing the volume/height, the increased absorption and the higher distances of the added source volumes cause a reduced counting efficiency, which is energy
dependent. This kind of data is useful for optimizing the containers geometry when counting volume sources.

![Graph showing the counting efficiency for a cylindrical vessel of 13.5 cm diameter, containing different amounts of a homogeneous radioactive source emitting different γ energies (source density 2 g/cm³)](image)

**Figure 4.** The counting efficiency for a cylindrical vessel of 13.5 cm diameter, containing different amounts of a homogeneous radioactive source emitting different γ energies (source density 2 g/cm³)

**CONCLUSIONS**

A user-friendly application intended to facilitate preparation of input data to MCNP was developed and used. It is based on templates specific for given geometrical systems, which prepare the input for MCNP in the required format. It is easy to operate, as the parameters are defined in simple screens, and allows direct display on the computer screen.

**REFERENCES**

An Ab Initio Calculation of Exclusive $^4$He photodisintegration

N. Nevo Dinur$^1$, N. Barnea$^1$

$^1$Racah Institute of Physics, Hebrew University, Jerusalem, Israel

Corresponding Author: nir.nevo@mail.huji.ac.il

INTRODUCTION
We present a recently developed computer code that allows for ab initio calculations of inelastic reactions with $A=3+1$ nucleons, as applied to the calculation of the exclusive two-body cross section of $^4$He photodisintegration, i.e., the reactions $^4$He$(\gamma,p)^3$H and $^4$He$(\gamma,n)^3$He, with the full final state interaction explicitly accounted for. The calculation is not limited to the energy region below three- and four-body breakup thresholds.

$^4$He PHOTODISINTEGRATION
The $A=3,4$ systems are excellent candidates as testing grounds for one to study state-of-the-art nuclear models. These include not only nucleon-nucleon force models with dozens of free parameters, which must be accompanied with an appropriate three-nucleon force, but also the nuclear currents, through which the nucleus interacts with external electroweak probes. Using several sophisticated few-body techniques, which have been perfected over the last two decades, one can aspire to make almost exact ab initio calculations in systems with such a small number of particles, yet with a wide range of phenomena to be explored. One of the methods that were benchmarked on the $A=4$ system is the EIHH method. In this work we applied a particular version of it, written especially to allow for the calculation of $A=3+1$ reactions. The most demanding few-body calculations are undoubtedly those of scattering reactions, where at least one of the initial or final nuclear wave functions lies in the continuum. The LIT method, which we used for the current work, has already been successfully applied for exclusive (inclusive) scattering cross sections of various electroweak probes in systems with up to four (seven) nucleons.

The case of $^4$He photodisintegration is of particular importance due to the long standing conflicting experimental results (especially in its peak region, which lies well below the pion-production threshold, thus justifying our use of the unretarded dipole approximation). In order to make a significant contribution to our understanding of this case, a calculation which uses realistic models was needed. This required explicitly including all internal degrees of freedom of all the nucleons, as was done in our code. Our calculation also takes into account the final state interaction (FSI) contribution that was found to be dominant over the Born term in the peak region. We also treated the Coulomb part of the FSI, using in the asymptotic wave function the average Coulomb interaction between the fragments.

Including the Coulomb interaction in the FSI required isospin non-conservation. This was also done consistently throughout the entire code, and we successfully benchmarked the high isospin part(s) of the ground-state wave function of the $A=3,4$ nuclei (less than 0.01%).

RESULTS
Preliminary results will be presented and discussed.
CONCLUSIONS
A computer code was written and applied to the calculation of the exclusive two body \(^4\)He photodisintegration cross section, for the first time using realistic nucleon-nucleon force models. Preliminary results will be presented and discussed.

REFERENCES
Carnallite Density Measurement System for the Dead Sea Evaporation Ponds Based on Gamma Radiation

Yagel Achrak¹, Moti Aharoni², Itzhak Orion¹ and Yael Peleg²

¹Department of Nuclear Engineering, Ben-Gurion University of the Negev, Beer-Sheva, 84105 Israel
²Dead Sea Works LTD, Potash House POB 75 Beer-Sheva, 84100 Israel

Corresponding Author: iorion@bgu.ac.il

INTRODUCTION
The method of gamma ray attenuation for density measurements of variety liquid and material is a well known nondestructive technique. Carnallite is an evaporate mineral, a hydrated potassium magnesium chloride with formula: KCl•MgCl₂•6(H2O), which is mined for both potassium and magnesium.

Evaporation ponds at the Dead Sea contain Carnallite layer below the salty solution level. The thickness of the Carnallite layer under the solution is varying depended on the pond position and the environmental conditions. The Carnallite density is varying along the evaporation ponds properties. Measurements of the Carnallite density are very important for harvesting assessments and for commercial predictions. This Technique based on the principle of a direct dependence between the radiation absorption ratio and the density of the medium for a known material, is well known.

In this paper we report on Monte Carlo simulation study for density measurements system development. The system principle performance and main parameters were tested with these simulations.

Monte Carlo simulation using MCNP [1] were executed to estimate various configurations using ⁶⁰Co, ¹³⁷Cs gamma sources including the collimation shape (isotropic, pencil beam), geometry, and media properties. The source-detector distance was varied in order to introduce the system performance.

In case of using ⁶⁰Co source, the spectral resolution must be fairly high in order to separate the ⁴⁰K natural radiation peak of 1460 keV from the ⁶⁰Co 1332 keV peak. BrilLanCe-380 [2] scintillator LaBr₃(Ce) (1.5" X 1.5") was tested for high resolution spectrometry in room standard temperature measurements.

RESULTS
It is essential to examine the dependence counts versus source-detector distance for a given density using Monte Carlo. In order to fit the gamma source activity and energy to the density measurements statistical accuracy requirements, Monte Carlo simulations were carried out to assess various experimental configurations of ¹³⁷Cs or ⁶⁰Co gamma sources. Each counting point was simulated using 10⁹ histories. Figs 1-2 shows simulations BrilLanCe -380 detector counts of varied source-detector distance for ¹³⁷Cs and ⁶⁰Co. We found out from the results that it is
preferable to use $^{60}$Co source, since the detector counts using the $^{60}$Co source are higher more than one order of magnitude compare to $^{137}$Cs source, in range of 70 cm.

Figure 1: Simulation results with $^{137}$Cs source.

Figure 2: Simulation results with $^{60}$Co source.

One of the Carnallite components is the $^{40}$K, a radioisotope with energy of 1460 keV, which appears in the spectrum. The proximity between the $^{60}$Co peak at 1332 keV to the $^{40}$K peak is shown in spectral measurements using NaI(Tl) scintillator (Figure 3).
CONCLUSIONS

Figure 3 shows the NaI(Tl) detector spectral results, where overlap appears between the $^{60}$Co peak and the $^{40}$K peak. Higher spectral resolution is needed to assist the accuracy increasing of detection threshold.

Figure 4 shows the capability of the BrilLanCe-380 detector to distinguish between the $^{60}$Co peaks and the $^{40}$K peak.

CONCLUSIONS

The Monte Carlo simulations results indicated that using $^{60}$Co as a source is worthwhile than $^{137}$Cs source, since the counts were found to be in one order of magnitude higher. In conclusion,
for the case of Carnallite density measurements, the use of BrilLanCe-380 spectrometer is mandatory.

REFERENCES
Analysis of the Electron Average Energy Shift for Layer Thickness Estimation

A. Givon¹, E. Tiferet², I. Orion¹

¹ Ben-Gurion University of the Negev (BGU), P.O. Box 653, Beer-Sheva 84105, Israel.
² Nuclear Research Center-Negev (NRCN), P.O. Box 9001, Beer-Sheva 84190, Israel.

Corresponding Author: givonal@bgu.ac.il

INTRODUCTION

Thickness measurements of thin films are essential for industrial fields such as semiconductor, microelectronics, photonics and Microsystems (1) as well as for scientific research. Electron spectroscopy techniques (2-5) are one of the ways to perform non destructive thickness measurement on thin films. In this paper, an analysis is performed on an electron spectrum emerging from a sample, bombarded by 10 keV electrons. Typically, most electron surface analysis techniques uses electrons from a few eV up to 2000 eV. However, in recent decades, the improved performance of third generation synchrotrons enables the use of high intensity and more energetic electrons then before (6). Therefore, data of elastic and inelastic interactions at higher energies are available (7).

The basic type of thickness measurement, using electrons, is to measure the signal electron after it had interacted with the sample. A common technique is the X-ray photoelectron spectroscopy (XPS) (2). In XPS an incident photon beam interacts with a reference substrate and emits photoelectrons creating the reference detected signal. Then, an overlayered sample is subjected to the same procedure. When the electrons pass through the overlayer, some of the electrons will lose energy, hence the signal intensity will be decreased. The overlayer thickness can then be calculated using tabulated electron inelastic mean free path (IMFP) values and the ratio between the intensities of the reference and the overlayered samples.

In order to elucidate the signal electron from the background, noises such as bulk inelastic scattering, surface excitations, X-ray satellites and intrinsic losses (5), that occur both at the signal generation in the substrate and in the overlayer, must be taken into consideration. Extensive work was carried out over the years to eliminate these disturbances from the electron spectra measurements (8-14) and to evaluate the electron IMFP. However, there are still many assumptions in the field of electron transport.

When calculating layer thickness using the signal electron, as mentioned above, the measurement of the reduction in signal intensity due to inelastic collisions determines the layer thickness. The bulk inelastic scattering is one of the main contributors to the disturbance that affects the electron spectra. That is so, since the majority of the energy loss occurs in the bulk material. However, as the electron trajectory length is dominated by the elastic scattering effect (8), the electron projection length from a given angle may differ from its actual path length. A correction to the electron projection length due to consideration of the electron elastic scattering is given by the electron effective attenuation length (EAL) (15-17).

The aim of this paper is to present a quantitative analytic algorithm to evaluate the thickness of a layer. It uses the correlation between the electron path length and the electron average energy shift. This correlation is encoded in the inelastic scattering part of the electron spectra. This algorithm is based on the analysis of multiple inelastic scattering.

Multiple inelastic scattering spectra are determined by the probability for n-fold scatterings, and the energy loss distribution after n inelastic collisions (8,9). The calculation of multiple inelastic scatterings is used for deconvolution procedure in various electron spectroscopes, subtracting it from the electron spectrum to isolate the signal electron peak. Here we use the
Deconvolution of the bulk losses requires a calculated spectrum that considers only the electron inelastic effects. To generate these spectra, the multiple inelastic scattering effect should be considered with three possible options: The electron had no interaction in the layer, one interaction in the layer or multiple interactions in the layer. In the first case, the electron retains its original energy. In the second case, the electron energy depends on its probability to lose a certain amount of energy during that interaction. This probability is described by the differential inverse inelastic mean free path (DIIMFP)\(^{(18)}\). When looking at the graph of DIIMFP vs. energy\(^{(8)}\), the probability for a collision is different for different initial electron energies. However, the energy loss probability in a collision remains the same for different initial electron energies. And since the energy loss probability is independent of the initial electron energy but depends on the property of the material, if the DIIMFP is normalized the energy loss distribution for an individual collision in the layer is obtained. Finally, the energy loss distribution for an electron that had undergone more than one collision can be obtained by self convolution on the normalized DIIMFP. To generate the final spectrum, the energy loss distribution obtained from the self convolution on the normalized DIIMFP has to be multiplied by the electron probability to undergone that number of interactions. Summation on all the possible number of interactions yields the requested spectrum.

**THEORY**

The average energy shift \((E_{Av})\) for electrons with initial energy \(E_0\), which have undergone multiple inelastic scatterings in a specific layer, can be calculated from spectrum measurements. The average energy shift caused by the multiple inelastic scattering, can also be defined as

\[
E_{Av} = \bar{n} \cdot T
\]  

(1)

Where \(\bar{n}\) is the average number of inelastic interactions an electron undergoes in a layer with thickness \(x\), and \(T\) is the mean energy-loss per collision. This relation is possible since the mean energy-loss is independent on the incident electron energy\(^{(8)}\). When analyzing the electron inelastic data, the energy dependence of the inelastic mean free path (IMFP) is

\[
\lambda(E) = \lambda(E_0) \cdot (E/E_0)^\alpha
\]  

(2)

\[
E = E_0 - E_{Av}
\]  

(3)

Where \(\lambda(E)\) denote the energy dependent IMFP, and \(\alpha\) is a fit coefficient. Inserting Eqns. 1&3 into Eqn. 2, a collision dependence of the IMFP is obtained

\[
\lambda(E) = \lambda_0 \cdot [(E_0 - n \cdot T)/E_0]^\alpha = \lambda_0 \cdot (1 - R \cdot n)^\alpha
\]  

(4)

Where the reduction ratio \(R\) is a property of the material and it is define as

\[
R \equiv T/E_0
\]  

(5)

Hence, Eqn. 4 depends only on the properties of the material, and the number of interactions an electron has undergone. For clarity, the switch between \(\bar{n}\) in Eqn. 1 to \(n\) in Eqn. 4 is justified since Eqn. 4 gives the IMFP for a given electron, and is not an average over large amount of electrons as in Eqn. 1. The electron inelastic mean free path is the distance at which the electrons have undergone, on average, one inelastic collision. At distance \(x\) the electrons has, on average, undergone between \(n-1\) and \(n\) inelastic collisions. After every collision the inelastic mean free path changes (due to energy-loss). The layer thickness \((x)\) can be defined as an interpolation between Eqn. 6 and Eqn. 7.

\[
\sum_{n=0}^{\infty} \lambda_k
\]  

(6)

\[
\sum_{n=0}^{\infty} \lambda_k
\]  

(7)
Were $\lambda_K$ is the collision dependent IMFP, and $n$ is the number of collisions. The electron average energy-loss per collision for electrons with initial energy $E_0$ in a certain material is known (18,19). Therefore, the average number of interactions $\bar{n}$ can be calculated from Eqn. 1. By interpolating Eqns. 6&7 to $\bar{n}$ the layer thickness can be obtained.

RESULTS and DISCUSSION
The spectra in Fig. 1 are the calculated results from multiple inelastic interaction of a 10 keV electron beam parallel to the surface normal, passing through each pure element film. To analyze these spectra using the algorithm above, the IMFP values as a function of energy have to be fitted to Eqn. 2. The values were calculated using the NIST IMFP database (20) for $E_0 = 10$ keV. The spectra were analyzed in order to obtain the average energy shift ($E_{Av}$) of the multiple inelastic scatterings for each of the four elements, and for four thicknesses per element. The average energy shift was converted using Eqn. 1 to average number of inelastic interactions ($\bar{n}$), were $\bar{T}$ was calculated from DIIMFP values (21). The results are summarized in Table 1 and in Table 2.

![Figure 1: Energy-loss distributions for Ag, Au, Ni and Cu at four different thicknesses (50, 100, 150 and 200 nm).](image)

Table 1: Average energy shift in eV of the multiple inelastic scatterings for Au, Ag, Cu and Ni at four different thickness.

<table>
<thead>
<tr>
<th>Element</th>
<th>500 Å</th>
<th>1000 Å</th>
<th>1500 Å</th>
<th>2000 Å</th>
</tr>
</thead>
<tbody>
<tr>
<td>Au</td>
<td>333.06</td>
<td>675.00</td>
<td>1026.65</td>
<td>1386.47</td>
</tr>
<tr>
<td>Ag</td>
<td>300.46</td>
<td>608.39</td>
<td>924.42</td>
<td>1246.46</td>
</tr>
<tr>
<td>Cu</td>
<td>256.29</td>
<td>517.97</td>
<td>785.43</td>
<td>1056.12</td>
</tr>
<tr>
<td>Ni</td>
<td>306.63</td>
<td>621.13</td>
<td>944.18</td>
<td>1271.94</td>
</tr>
</tbody>
</table>

Table 2: Average energy loss per collision, initial IMFP and $\alpha$ from the fit, reduction ratio R, and the average number of inelastic interactions.

<table>
<thead>
<tr>
<th>Element</th>
<th>$\bar{T}$ [eV]</th>
<th>$\lambda_0$ [Å]</th>
<th>$\alpha$</th>
<th>R</th>
<th>$\bar{n}$</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>500 Å</td>
<td>1000 Å</td>
<td>1500 Å</td>
<td>2000 Å</td>
<td>500 Å</td>
</tr>
<tr>
<td>Au</td>
<td>51.79</td>
<td>78.74</td>
<td>0.7524</td>
<td>0.005179</td>
<td>6.4307</td>
</tr>
<tr>
<td>Ag</td>
<td>53.30</td>
<td>89.76</td>
<td>0.7823</td>
<td>0.00533</td>
<td>5.6370</td>
</tr>
<tr>
<td>Cu</td>
<td>52.66</td>
<td>103.78</td>
<td>0.7829</td>
<td>0.005266</td>
<td>4.8668</td>
</tr>
<tr>
<td>Ni</td>
<td>52.86</td>
<td>87.26</td>
<td>0.7903</td>
<td>0.005286</td>
<td>5.8007</td>
</tr>
</tbody>
</table>
Using the summation of Eqns. 6&7, the length of n-1 interactions and n interactions were calculated. The n-1 and n in table 3 are the lower and upper integer of $\bar{n}$ from table 2 respectively. The $x(n-1)$ and $x(n)$ are the electron average path length after n-1 and n interactions respectively. This summation (table 3) takes into consideration the change in the electron mean free path after each interaction due to the electron energy loss. The interpolation of $x(n-1)$ and $x(n)$ to $x(\bar{n})$ produces the thickness estimation in table 4.

Table 3: n-1 and n inelastic interaction for each material and thickness, and the corresponding path length for each material and thickness.

<table>
<thead>
<tr>
<th>Element</th>
<th>500 Å</th>
<th>1000 Å</th>
<th>1500 Å</th>
<th>2000 Å</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>n-1</td>
<td>n</td>
<td>n-1</td>
<td>n</td>
</tr>
<tr>
<td>Au</td>
<td>6</td>
<td>7</td>
<td>13</td>
<td>14</td>
</tr>
<tr>
<td>Ag</td>
<td>5</td>
<td>6</td>
<td>11</td>
<td>12</td>
</tr>
<tr>
<td>Cu</td>
<td>4</td>
<td>5</td>
<td>9</td>
<td>10</td>
</tr>
<tr>
<td>Ni</td>
<td>5</td>
<td>6</td>
<td>11</td>
<td>12</td>
</tr>
<tr>
<td></td>
<td>x(n-1)</td>
<td>x(n)</td>
<td>x(n-1)</td>
<td>x(n)</td>
</tr>
<tr>
<td>Au</td>
<td>467.8</td>
<td>544.7</td>
<td>999.6</td>
<td>1074.3</td>
</tr>
<tr>
<td>Ag</td>
<td>445.0</td>
<td>532.9</td>
<td>966.7</td>
<td>1052.3</td>
</tr>
<tr>
<td>Cu</td>
<td>412.5</td>
<td>514.6</td>
<td>918.6</td>
<td>1018.5</td>
</tr>
<tr>
<td>Ni</td>
<td>432.6</td>
<td>518.1</td>
<td>939.7</td>
<td>1022.9</td>
</tr>
</tbody>
</table>

Table 4: Estimation results of the layer thickness and deviation from real value.

<table>
<thead>
<tr>
<th>Element</th>
<th>Thickness estimation [Å]</th>
<th>Deviation from real value [%]</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>500 Å</td>
<td>1000 Å</td>
</tr>
<tr>
<td>Au</td>
<td>501.0</td>
<td>1001.9</td>
</tr>
<tr>
<td>Ag</td>
<td>501.0</td>
<td>1002.2</td>
</tr>
<tr>
<td>Cu</td>
<td>501.1</td>
<td>1002.0</td>
</tr>
<tr>
<td>Ni</td>
<td>501.0</td>
<td>1002.3</td>
</tr>
</tbody>
</table>

The results of the thickness estimation using this algorithm are in good agreement with the actual layer thickness used for the calculation of the electron bulk inelastic spectra. Average deviation from the real layer thickness is 0.17 percent. The algorithm presented here relies on accurate spectrum measurements that produce the average energy shift of the measured electrons, and also on electron inelastic properties for the values of IMFP and average energy-loss per collision.

**CONCLUSIONS**

The traditional approach of analyzing the signal electron is extended here, and takes into consideration the total electron energy spectrum. In this approach, the tendency of the multiple inelastic scattering spectra to increase the average energy shift, depending on the number of inelastic interactions, is utilized to estimate the thickness of a layer. The algorithm outlined here gives an accurate, straightforward approach for the estimation of layer thickness. The power equation used for the IMFP has been fitted with NIST database\(^{(20)}\), and applies to the four materials in inspection for this work.
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Monte Carlo Simulations of the Response of a Germanium Detector to Gamma Rays

O. Aviv, Y. Nir-El, G. Haquin

Soreq Nuclear Research Center, Yavne 81800, Israel

Corresponding Author: oferav@soreq.gov.il

INTRODUCTION

In the purpose of better understanding the operation and the complete characterization of a High-Purity Germanium (HPGe) detector (response to gamma radiation), Monte Carlo simulations using GEANT4 (GEometry ANd Tracking) were employed. GEANT\(^{(1)}\) is an object oriented program used to study interactions between low and high energy particles with matter by Monte Carlo methods. These interaction processes depend on the particle’s type, charge state, energy, momentum and the composition of the medium that it passes through. In the last decade several studies dealing with the modeling of passage of Gamma rays through Germanium detectors using GEANT have been published\(^{(2,3)}\). In this paper we present preliminarily results for the calculation of the absolute efficiency of a HPGe detector to gamma rays through Monte Carlo simulations.

The studied HPGe detector (model 65-83, GEM series, ORTEC), is a p-type coaxial detector having ~70% relative efficiency for 1332 keV gamma rays. The experimental system\(^{(4)}\) shown in Figure 1, includes the HPGe located inside a rectangular castle having inner dimensions of 28 cm x 28 cm x 48 cm. The detector is surrounded by several layers of shielding consisting of 1 cm of Copper, 50 mm of low background Lead (\(^{210}\)Pb 2.8 Bq/kg) and 100 mm of ordinary Lead. Outside the detector castle and above the lead shielding exists an active shielding consisting of a rectangular (100 cm x 100 cm x 5 cm) BC408 plastic scintillator (manufactured by Saint-Gobain). The Ge detector is triggered in anti-coincidence (veto) with the scintillator detector to attenuate contributions from cosmic background radiation (e.g. muon flux). The data acquisition and analysis is controlled by Genie2000 software. The assembly of passive and active shielding decreases the background radiation in the energy range of 40-2800 keV to an average count rate of 1.2 events/second\(^{(4)}\).

The Ge crystal detector configuration type is closed-ended co-axial (bulletized). Figure 2 shows the detector geometry. The active detection volume is a Germanium crystal with a cylindrical shape (Ø69.8 mm X 80 mm) having a central hole (Ø9.2 mm X 66.3 mm) for charge collection and also for cryogenic cooling.
Figure 1. Scheme of the experimental setup.

The crystal is coated by layers of Boron (0.003 mm), Lithium (0.7 mm), Mylar and Aluminum (0.06 mm) and another thicker layer of Aluminum (0.7 mm). The crystal and layers are surrounded by an endcap of Carbon fiber (0.76 mm).

Figure 2. HPGe Detector geometry.

The geometry that was used in the simulations was identical to manufacturer specifications with the exception of the Boron layer which was ignored due to its minor effect. The detector and the gamma source were positioned inside the passive shielding consisting of Copper and Lead (similar configuration shown in Figure 1). We benchmark our simulations using 1332 keV and 1173 keV Gamma ray energies (i.e. identical energies to the gamma rays emitted from $^{60}$Co) positioned 15 cm from the endcap.

The package which was considered in the simulations of GEANT4 was CERN’s “LHEP_EMV” (low energy parameterized for high performance) which includes advanced models describing inelastic interactions for all hadrons at the limit of low energy.

RESULTS

Figure 3 shows the simulated spectrum from a point source of $^{60}$Co positioned 15 cm with respect to the detector together with experimental results. The trends are very similar and many of the experimental effect (backscattered gamma, Compton continuum, Compton edges, full-energy-peaks, single escape peaks, and annihilation events) are reconstructed by the simulations.
Events within the full-energy-peak (fep) are those in which the gamma energy is fully deposited in the crystal volume. The absolute detection efficiency for the fep events is defined as the ratio of the net integral under the fep divided by the number of total gamma emission. The calculated detection efficiency for 1332.5 keV gammas was found to be 0.0020±0.00005 [counts/gammas], which is in excellent agreement with the measured value being 0.00195±0.00003 [counts/gammas].

The study of the detector efficiency using MC simulation was extended to gamma rays in the energy range of 10 keV – 5000 keV. Here we considered a mono-energetic point sources, positioned at a distance of 15 cm from the detector front face. Figure 4 shows the calculated absolute efficiency for detecting gamma rays against their initial energy (black circles), presented in double logarithmic scale. Each calculated point was obtained by simulation of 10^6 events. The general trend of the calculated detection efficiency is similar to previous works\(^{(2)}\). The absolute efficiency for several gamma energies was determined by measuring several point sources (\(^{241}\)Am, \(^{57}\)Co, \(^{133}\)Ba, \(^{137}\)Cs, \(^{60}\)Co, \(^{88}\)Y) with well known activities, positioned 15 cm away from the HPGe detector. The experimental data points are also presented in Figure 4 (blue squares). In the limit of high energies (\(E_\gamma>200\) keV), the calculated efficiencies are in relatively good (<5%) agreement with experimental values and within the experimental uncertainty. For lower energies there is an increasing discrepancy with decreasing energy between the experimental results and the calculated values. These deviations are still under investigation. A possible reason for these large discrepancies may be attributed to inaccurate dimensions of the detector assembly, e.g. underestimation of the dead layer leading to less absorption of gammas at lower energies which is translated to more gamma absorption by the Germanium crystal.
CONCLUSIONS
Monte Carlo simulations of the passage of gamma rays through a Germanium detector were performed. Among others, the absolute detection efficiency was studied giving a typical value of $2 \cdot 10^{-3}$ counts/gamma for a point source of 1.3 MeV gammas located 15 cm from the detector. Simulations are in good agreement with experimental values, although further work is required to improve the accuracy as well as the compatibility with experimental findings especially in the lower energy range. We plan to extend our work by studying other Germanium detectors existing in our labs in order to calibrate their absolute detection efficiency under different experimental configurations (source geometry and distance to detector) as well as consideration of complex decay patterns.

REFERENCES
Kinetic Modeling for LiF:Mg,Ti Incorporating Creation of Defects in the Irradiation Stage.

I. Eliyahu¹,², Y. Horowitz¹ and L. Oster³

¹ Ben Gurion University of the Negev, Beersheva, 84105, Israel
² Soreq Nuclear Research Center, Yavne, 81800, Israel
³ Sami Shamoon College of Engineering, Beersheva, Israel

Corresponding Author: ilan@soreq.gov.il

ABSTRACT

A kinetic model is described for LiF:Mg,Ti which incorporates creation of fluorine vacancies via irradiation and their subsequent filling by electrons leading to the creation of F centers. With the appropriate choice of parameters, the model successfully predicts the dose response behavior of the 4 eV and 5.45 eV trapping centers.

INTRODUCTION

The calculation of heavy charged particle (HCP) relative TL efficiencies using track structure theory (TST) is important to the accurate dosimetry of complex/mixed radiation fields in order to establish a sound theoretical basis for the applied techniques and aid in the interpretation and analysis of dosimetric data. A major assumption of TST is that the radiation effects of high linear energy transfer (LET) HCPs arise exclusively from the contribution of the secondary and higher order electrons generated by the HCP slowing down. Direct atomic displacements arising from HCP nucleus elastic scattering which lead to the enhanced creation of F centers and other defects and to the possible alteration of the TL mechanisms are assumed to have a negligible effect compared to the radiation action of the ejected secondary electrons. The validity of this major premise of TST has yet to be established. In this investigation, a more sophisticated version of earlier kinetic models (1), is developed which predicts, as a function of dose, the effect of negative ion vacancy (F center) creation and recombination on the population of the 4 eV trapping center giving rise to composite glow peak 5 in the glow curve of LiF:Mg,Ti.

The Kinetic Model

The details of the model are shown in Figure 1 and immediately below, the coupled differential equations describing the traffic of electron and hole charge carriers in the irradiation stage. These were solved numerically using the fifth order Runge Kutta method (matlab code 23s). The excitation dose, D, is defined as D = X*t, where t is the total length of excitation time and X is the production rate of the electron and holes. In our simulation X was taken as equal to 3.7e20 s⁻¹m⁻³ and the initial values of mHT, mLC, nTc, ncc, nF, nTc², nc, nv were zero. TC₁ is intended to correspond to the 4 eV optical absorption band and is identified with composite glow peak 5 in the glow curve of LiF:Mg,Ti and CC is identified with the 5.45 eV electron trapping competitive center. The dose response of these traps following electron/photon irradiation have been experimentally measured as linear/exponentially saturating with dose filling constants of 10⁻³ Gy⁻¹ and 4 x 10⁻⁴ Gy⁻¹ respectively (2). In LiF:Mg,Ti the F center is associated with an OA band at 5 eV with a dose filling constant of 10⁻⁴ Gy⁻¹ (2). However, various types of dose response characteristics have been reported in other alkali halides (pure and doped) as well as in LiF, including supralinear behaviour at low dose levels, which indicates a complicated somewhat uncharted situation (3). TC₂ and HT₁
are intended as "catch-all" defects which describe the over-all trapping characteristics of the remaining hole and electron trapping centers in the LiF:Mg,Ti complex system.

**Fig 1 : Energy level scheme for the model**

1. \[
\frac{dm_{LC}}{dt} = -A_{mLC} \cdot m_{LC} \cdot n_v + B_{LC} (M_{LC} - m_{LC}) \cdot n_v
\]

2. \[
\frac{dm_{HT}}{dt} = -A_{mHT} \cdot m_{HT} \cdot n_v + B_{HT} (M_{HT} - m_{HT}) \cdot n_v - P_{mf} \cdot (M_{HT} - m_{HT}) \cdot n_F
\]

3. \[
\frac{dn_{TC}}{dt} = A_{TC} \cdot (N_{TC} - n_{TC}) \cdot n_e
\]

4. \[
\frac{dn_{CC}}{dt} = A_{CC} \cdot (N_{CC} - n_{CC}) \cdot n_e
\]

5. \[
\frac{dn_{F}}{dt} = A_{F} \cdot n_v \cdot [(N_{vacS} (1 - e^{-β_F \cdot t_{Dose}}) + N_{vacS0}) - n_F] - P_{mf} \cdot (M_{HT} - m_{HT}) \cdot n_F
\]

6. \[
\frac{dn_{TC2}}{dt} = A_{TC2} \cdot (N_{TC2} - n_{TC2}) \cdot n_e
\]

7. \[
\frac{dn_{TC3}}{dt} = A_{TC3} \cdot (N_{TC3} - n_{TC3}) \cdot n_e
\]

8. \[
\frac{dn_{TC4}}{dt} = A_{TC4} \cdot (N_{TC4} - n_{TC4}) \cdot n_e
\]

9. \[
\frac{dn}{dt} = X - n \cdot [B_{LC} \cdot (M_{LC} - m_{LC}) + B_{HT} \cdot (M_{HT} - m_{HT})]
\]

10. \[
\frac{dn}{dt} = X - n \cdot [A_{mLC} \cdot m_{LC} \cdot n_v + A_{mHT} \cdot m_{HT} \cdot n_v \ldots + A_{TC} \cdot (N_{TC} - n_{TC}) + A_{CC} \cdot (N_{CC} - n_{CC}) + A_{F} \cdot [(N_{vacS} (1 - e^{-β_F \cdot t_{Dose}}) + N_{vacS0}) - n_F]^{-} + A_{TC2} \cdot (N_{TC2} - n_{TC2}) + A_{TC3} \cdot (N_{TC3} - n_{TC3}) + A_{TC4} \cdot (N_{TC4} - n_{TC4})]
\]
A description of the various defect states is shown in Table 1. The behaviour of the F center is described in equation 5 which incorporates the creation of vacancies proportional to $D^a$ (3) as well as direct recombination of F center electrons to the hole center via geminate recombination which has been described by Mayhugh(4) as part of the recombination process during heating and was included herein since our model will be extended to the heating stage to describe TL processes as well. Using the values of the parameters shown in Table 1 leads to dose filling of the 4 eV and 5.45 eV electron trapping centers in agreement with experimental data (2) as shown in Figure 2.

<table>
<thead>
<tr>
<th>Center Name</th>
<th>Symbol</th>
<th>Center parameters</th>
<th>Irradiation value</th>
</tr>
</thead>
<tbody>
<tr>
<td>LC1 – luminescent center</td>
<td>$M_{LC}$</td>
<td>Center concentration</td>
<td>$1 \cdot 10^{10} [m^{-3}]$</td>
</tr>
<tr>
<td></td>
<td>$\lambda_{\text{mLC}}$</td>
<td>Electron trapping probability</td>
<td>$1 \cdot 10^{-20} [m^1s^{-1}]$</td>
</tr>
<tr>
<td></td>
<td>$B_{\text{LC}}$</td>
<td>Hole trapping probability</td>
<td>$1.4 \cdot 10^{-18} [m^3s^{-1}]$</td>
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<tr>
<td>HT1 – hole trapping center</td>
<td>$M_{HT}$</td>
<td>Center concentration</td>
<td>$4 \cdot 10^{22} [m^{-3}]$</td>
</tr>
<tr>
<td></td>
<td>$\lambda_{\text{mHT}}$</td>
<td>Electron trapping probability</td>
<td>$1 \cdot 10^{-24} [m^1s^{-1}]$</td>
</tr>
<tr>
<td></td>
<td>$B_{\text{HT}}$</td>
<td>Hole trapping probability</td>
<td>$3 \cdot 10^{-22} [m^3s^{-1}]$</td>
</tr>
<tr>
<td>TC1 – Trapping center</td>
<td>$N_{TC}$</td>
<td>Center concentration</td>
<td>$8 \cdot 10^{19} [m^{-3}]$</td>
</tr>
<tr>
<td></td>
<td>$\lambda_{\text{mTC}}$</td>
<td>Electron trapping probability</td>
<td>$2 \cdot 10^{-19} [m^3s^{-1}]$</td>
</tr>
<tr>
<td>CC – Competitive center</td>
<td>$N_{CC}$</td>
<td>Center concentration</td>
<td>$6 \cdot 10^{20} [m^{-3}]$</td>
</tr>
<tr>
<td></td>
<td>$\lambda_{\text{mCC}}$</td>
<td>Electron trapping probability</td>
<td>$7.5 \cdot 10^{19} [m^3s^{-1}]$</td>
</tr>
<tr>
<td>F – F center</td>
<td>$N_{\text{Vac0}}$</td>
<td>Initialize Center concentration</td>
<td>$5 \cdot 10^{22} [m^{-3}]$ – $5 \cdot 10^{26}$</td>
</tr>
<tr>
<td></td>
<td>$\lambda_{\text{mF}}$</td>
<td>Electron trapping probability</td>
<td>$5 \cdot 10^{-16} [m^3s^{-1}]$</td>
</tr>
<tr>
<td></td>
<td>$N_{\text{VacS}}$</td>
<td>Center concentration reservoir</td>
<td>$1 \cdot 10^{10} [m^{-3}]$</td>
</tr>
<tr>
<td></td>
<td>$\beta_F$</td>
<td>dose filling constant</td>
<td>$1 \cdot 10^{-6} [Gy^{-1}]$</td>
</tr>
<tr>
<td></td>
<td>$P_{\text{mF}}$</td>
<td>geminate recombination probability</td>
<td>$1 \cdot 10^{-22} [m^3s^{-1}]$</td>
</tr>
<tr>
<td>TC2 – Trapping center 2</td>
<td>$N_{\text{TC2}}$</td>
<td>Center concentration</td>
<td>$9 \cdot 10^{22} [m^{-3}]$</td>
</tr>
<tr>
<td></td>
<td>$\lambda_{\text{mTC2}}$</td>
<td>Electron trapping probability</td>
<td>$4.5 \cdot 10^{-22} [m^3s^{-1}]$</td>
</tr>
<tr>
<td></td>
<td>$X$</td>
<td></td>
<td>$3.7 \cdot 10^{26} [m^3s^{-1}]$</td>
</tr>
</tbody>
</table>

Tabl1 1 definition of the parameters
CONCLUSIONS
F centers can play a very significant role on the 4 eV- TC characteristics in both the irradiation/absorption stage and the recombination stage of the TL mechanism in LiF:Mg,Ti. In this investigation a kinetic model has been constructed which incorporates vacancy/F center creation in the irradiation stage which is successful in predicting the dose filling rate of the 4 eV TC giving rise to peak 5 and the 5.45 eV TC which serves as the competitor in the recombination stage. Creation of vacancies/F centers by the irradiation alters the population characteristics of the 4 eV TC. These calculations will be incorporated in a qualitative/theoretical estimate of the role of enhanced F center production in HCP radiation effects. The need for this estimate comes from the increasing evidence that TST in its current/conventional forms fails to accurately predict relative HCP TL efficiencies.

REFERENCES
Experimental Examination of Holt Expression for TLD Response to Electron Beams

S. Daniel , R. Bar-Deroma
Oncology, Rambam Medical Center, Haifa, Israel

Corresponding Author: Daniel-s@013.net.il

INTRODUCTION
The response of thermoluminescent dosimeters (TLD100) to electron beams with various energies, at different depths and dosimeters size was investigated. Holt et.al.(1) proposed a semi-empirical expression in order to explain fluence reduction inside the TLD however; it was tested in limited conditions. In this work we expanded the measurement conditions and we have compared the results with the predictions of Holt's expression.

EXPERIMENT
40 TLD 100 chips (3.1X3.1X0.9 mm³, Harshaw) were calibrated for dose response and used in our experiment. The annealing process (pre irradiation) included heating the dosimeters to the temperature of 400°C for 30 min.

The dosimeters were then irradiated: A. with various electron nominal energies ranging 2 to 18MeV at the depth of maximum dose (results are shown in Figure 1). B. electron beams of 18MeV at different depths (results are shown in Figure 2).

The dosimeters were read 24 h after irradiation using a TLD reader (Harshaw 3500 reader). TLD response was defined as the area of peaks 4 and 5 calculated using GlowFit (Puchalska, Krakow, Poland) – peak separation technique.

RESULTS
The TLD response for the different nominal electron energies is shown in figure 1. Figure 2 shows the response of TLD to 18MeV electron beams at different depths. Results are normalized to TLD response when irradiated with 18MeV electron beams at the depth of maximum dose for both figures.

Figure 1. Comparison between calculated (Holt) and measured response of TLD100 irradiated with electron beam of various nominal energies.

Figure 2. Comparison between calculated (Holt) and measured response of TLD100 irradiated with 18MeV electron beam at various depths.
Figure 1 and 2 show reduction in TLD sensitivity for low electron beam energies. However, while the results of the first experiment show good agreement with Holt's theory, for the second experiment the agreement is rather poor (for the low electron energies). Further investigation is planned: TLD response to 9MeV electron beams at different depths and response of different TLD layers.

REFERENCES
The Effect of Sample/Planchet Geometry on Precision and Glow Peak Shapes in LiF-TLD-100 Thermoluminescent Dosimetry

D. Sibony\textsuperscript{1}, Y. Horowitz\textsuperscript{2} and L. Oster\textsuperscript{3}

\textsuperscript{1}Energy Engineering Unit, Ben Gurion University, Beersheva, Israel
\textsuperscript{2}Physics Department, Ben Gurion University, Beersheva, Israel
\textsuperscript{3}Physics Unit, Sami Shamoon College of Engineering, Beersheva, Israel

Corresponding Author: sibonyd@bgu.ac.il

INTRODUCTION

Precision of dose measurement of ± 5% (2 SD) is of great importance in many clinical radiotherapy applications and is a standard requirement (1). The use of LiF:Mg,Ti (TLD-100) is widespread (2) and it is acknowledged that the precision of measurement is protocol and user dependent and that many details in the dosimetric protocol must be adhered to achieve the required results.

In this paper we report on the effect of positioning of the sample in the planchet depression on both precision and glow peak shape parameters. The results reported herein arose from an intensive investigation into the reasons leading to the relatively poor precision of 15-20% (1 SD) we have previously obtained in the measurement of the ratio of peaks $5a/5$ (3). Peak $5a$ is a low temperature satellite of peak 5 whose intensity relative to peak 5 is influenced by ionization density (Linear Energy Transfer – LET).

RESULTS

Ten samples of LiF:Mg,Ti (TLD-100) 3mm×3mm×0.89mm chips were irradiated with a $^{90}$Sr/$^{90}$Y beta source to a dose level of 0.8 Gy. The chips were pre-irradiation- annealed in air at 400°C for 1h followed by natural cooling. Glow curve readout was carried out on a Harshaw 3500 manual reader at a heating rate of 1°C/s with reading resolution of 0.475°C/channel. The readout was carried out in two steps: (i) from an initial temperature of 50°C to a maximum temperature of 155°C, to remove the low-temperature component of the glow curves thereby simplifying the deconvolution procedure, (ii) readout from an initial temperature of 155°C to a maximum temperature of 250°C.
Figure 1. The effect of sample position on the glow curve.

Figure 1 illustrates the effect of sample position on the glow curve of LiF:Mg,Ti when placed on a Pt planchet with a depression 5 mm x 5 mm in dimension. This allows a 2 mm deviation in the horizontal direction which was found to have a very significant influence on \( T_{\text{max}} \) (the temperature of maximum intensity in the glow curve). As can be seen in Figure 1, \( T_{\text{max}} \) shifts from 203°C to 208.4°C from left to right positioning on the planchet. The precision of measurement of \( T_{\text{max}} \) was 0.6-0.9°C (1 SD). This variation was found to be of extreme importance in the estimation of peak 5a/5 and indeed careful positioning in the center of the planchet resulted in a factor 4-5 improvement in precision (5). In figure 2 we show an example of deconvolution which includes peak 5a.

Figure 2. Example of glow curve deconvolution.

It was also observed that positioning on the planchet has a significant effect on the precision of measurement of the intensity of the glow peaks as well as on the kinetic parameters.
deduced from the deconvolution analysis. The deconvolution is based on the computerized glow curve deconvolution code developed at Ben Gurion University (3). The results are shown in Tables 1 and 2 for peaks 4 and 5. In this analysis peak 5a was not introduced in order to simplify the deconvolution procedure and as well to ease comparison with the literature. All the parameters in the deconvolution procedure were allowed to vary. The most dramatic difference can be seen in the activation energy and s values of peak 4. The activation energy varies from 1.60 ± 0.03 eV (left) to 1.22 ± 0.03 eV (right), a difference of 25% and can certainly be one of the reasons explaining the great variation in E for this peak reported in the literature. The variations in s are even greater. The activation energy of peak 5, however, is well behaved and agrees with the value of 1.9-2.1 eV quoted in the literature.

Table 1. Kinetic parameters of peak 4

<table>
<thead>
<tr>
<th>Peak 4 - Average Results</th>
<th>Left side</th>
<th>Center</th>
<th>Right side</th>
</tr>
</thead>
<tbody>
<tr>
<td>Temperature [°C]</td>
<td>176.1 ± 0.5</td>
<td>177.2 ± 0.6</td>
<td>180.4 ± 0.5</td>
</tr>
<tr>
<td>Activation Energy [eV]</td>
<td>1.60 ± 0.03</td>
<td>1.46 ± 0.02</td>
<td>1.22 ± 0.03</td>
</tr>
<tr>
<td>Preexponential Factor [s⁻¹]</td>
<td>1.16E+17</td>
<td>2.85E+15</td>
<td>3.76E+12</td>
</tr>
</tbody>
</table>

Table 2. Kinetic parameters of peak 5

<table>
<thead>
<tr>
<th>Peak 5 - Average Results</th>
<th>Left side</th>
<th>Center</th>
<th>Right side</th>
</tr>
</thead>
<tbody>
<tr>
<td>Temperature [°C]</td>
<td>203.0 ± 0.4</td>
<td>204.7 ± 0.7</td>
<td>209.0 ± 0.5</td>
</tr>
<tr>
<td>Activation Energy [eV]</td>
<td>1.96 ± 0.01</td>
<td>1.98 ± 0.01</td>
<td>2.06 ± 0.01</td>
</tr>
<tr>
<td>Preexponential Factor [s⁻¹]</td>
<td>5.94E+19</td>
<td>9.14E+19</td>
<td>3.19E+20</td>
</tr>
</tbody>
</table>

Dosimetric analysis protocols are usually based on one of three alternatives; (i) measurement of the integral intensity of peaks 4 and 5 via region of interest (ROI) (ii) measurement of the peak height of peak 5, (iii) measurement of the intensity of peak 5 using deconvolution. Table 3 illustrates the changes in precision observed using two different methods of measurement for the three alternatives.

A. 8 measurements; in which 4 were carried out in the center of the planchet, 2 on the left and 2 on the right.

B. 4 measurements; all carried out with the sample positioned in the center.
Table 3. Changes in precision

<table>
<thead>
<tr>
<th>Chip No.</th>
<th>Measurement of the intensity of peak 5 using deconvolution</th>
<th>Measurement of the integral intensity of peaks 4 and 5</th>
<th>Measurement of the peak height of peak 5</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Method A</td>
<td>Method B</td>
<td>Method A</td>
</tr>
<tr>
<td>1</td>
<td>5.4%</td>
<td>4.0%</td>
<td>3.0%</td>
</tr>
<tr>
<td>2</td>
<td>4.1%</td>
<td>1.2%</td>
<td>2.6%</td>
</tr>
<tr>
<td>3</td>
<td>5.1%</td>
<td>2.4%</td>
<td>2.6%</td>
</tr>
<tr>
<td>4</td>
<td>4.8%</td>
<td>0.7%</td>
<td>3.2%</td>
</tr>
<tr>
<td>5</td>
<td>4.7%</td>
<td>3.2%</td>
<td>2.7%</td>
</tr>
<tr>
<td>6</td>
<td>4.7%</td>
<td>3.1%</td>
<td>1.9%</td>
</tr>
<tr>
<td>7</td>
<td>3.7%</td>
<td>2.0%</td>
<td>2.4%</td>
</tr>
<tr>
<td>8</td>
<td>4.7%</td>
<td>4.7%</td>
<td>2.5%</td>
</tr>
<tr>
<td>9</td>
<td>4.8%</td>
<td>3.4%</td>
<td>2.2%</td>
</tr>
<tr>
<td>10</td>
<td>3.8%</td>
<td>2.0%</td>
<td>2.6%</td>
</tr>
<tr>
<td>Average</td>
<td>4.6 ± 0.6%</td>
<td>2.7 ± 1.2%</td>
<td>2.6 ± 0.4%</td>
</tr>
</tbody>
</table>

CONCLUSIONS:
Careful positioning of the sample in the planchet depression can lead to a very significant improvement in the calculation of the kinetic parameters of individual peaks as well as to a significant improvement in precision of dose measurement.

REFERENCES:
Minimizing the residual signal of LiF:Mg,Ti readouts

A. Abraham¹, M. Weinstein¹, O. Pelled¹, U. German¹, Z.B. Alfassi²

¹ Nuclear Research Centre Negev, P.O.B 9001, Beer Sheva 84190, Israel
² Ben Gurion University of the Negev, Beer Sheva 84105, Israel

Corresponding Author: adlioh@gmail.com

INTRODUCTION

It is generally assumed that after the readout of TLD chips, all the traps are almost completely emptied and the residual readout is negligible. However, when performing routine readings (short readout times and high heating rates with no annealing), the deep traps are not completely cleared, and therefore, an additional readout of the crystal will indicate a residual dose. The residual dose for a normal additional readout (without any special treatment to enhance the residual dose, like UV irradiation) is about 0.2% of the accumulated historical dose of the TLD chip⁴ and could affect the result of the following readouts, especially when estimating low level doses.

The background signal in a TLD chip glow curve is a combination of the contributions from three main sources. The first one is the accumulated natural background dose for the period of time since its last readout. The second is the residual signal originated from historical irradiations of the TLD chip⁵. The third background source is the noise signal originated in the TLD chip and the TLD reader. For a well maintained operational system the value of the third contribution is low, constant and known, but the first two contributions are history dependent and may be quite significant.

Not taking into consideration the effect of the background contributed by the residual dose could lead to wrong dose evaluation especially when measuring low doses. In many works reported in the literature, the background contribution is evaluated by performing one additional reading after the first dose readout⁶,⁷. This procedure can induce significant errors, depending on the TLD history. In the present study we investigated the subject in more depth. A method for a more accurate evaluation of the residual background dose is presented.

MATERIALS AND METHODS

The experiments were performed with 15 older LG1110 LiF:Mg,Ti Thermo Inc. cards which were in use (cards with different irradiation histories, randomly chosen at the dosimetry laboratory of Nuclear Research Center Negev). These TLD cards contain three crystals 3 × 3 × 0.38 mm (TLD-100) at positions 1 to 3 and only Teflon (no TLD crystal) at position 4. The only precondition was that the residual dose of all un-irradiated chips should be less than 0.02 mSv. In order to get this low residual dose values the TLD cards were irradiated by UV (using a 15 W - 254 nm UV lamp) and a series of following readouts were performed until the reading was less the 0.02 mSv. This initial readout was marked as readout number -1, and was regarded as the baseline value.

The cards were separated into 3 groups and each group was irradiated to a different dose (50, 100 and 500 mSv). The irradiated cards were read eleven consecutive readouts. The first
readout after the irradiation (dose readout) was marked as readout number 0 and the following residual readouts where marked as readout number 1 to 10 respectively. All readouts where performed by a TLD6600 reader (Thermo Inc.). The heating of the TLD crystals is by a hot nitrogen gas jet. The pre-heat conditions, heating rate and maximum temperature are determined through a PC that controls the reader. The time temperature profile (TTP) used includes a pre-heat to 50°C (0.5 s), after which a linear heating of 25°C/s is applied up to a maximum temperature of 300°C, then keeping this maximum temperature up to the predetermined total time of 13.3 s. After each reading, the crystals were immediately cooled to room temperature. This temperature profile is applied usually in routine dose evaluations. The result of the readout process is a digitized 200 channel glow curve.

RESULTS AND DISCUSSION

The way to minimize the effect of the historical accumulated residual dose is to empty the deep traps (before the irradiation) as much as possible. However, this method is practical mostly in research situations whereas in day to day routine work there is neither time nor justification for this additional work. Therefore, in special cases where there is a need to investigate in more depth the residual dose contribution, it is to be evaluated in retrospect.

Table 1 shows the value of the average initial residual doses (read -1) and the ratios of the average residuals to these values after the corresponding consecutive readout. The table shows that there is a difference in the initial dose values and the behavior of the residual doses between the different groups, as a result of the difference in the use and exposure history of the different TLD chips. After 5 residual readouts the average readout of the cards irradiated to a dose of 50 mSv had a similar value as the initial residual dose, while for the two other groups (100 mSv and 500 mSv) the average value after 10 readout cycles was higher than the initial residual dose by 26% and 23% respectively. However, it should be mentioned that the absolute value of these differences is about 13 µSv and 20 µSv, which are not significant from a practical point of view.

Table 1. The average background doses and the ratios of the residual readouts to the background doses (read -1).

<table>
<thead>
<tr>
<th>dose [mSv]</th>
<th>background dose (read -1) [µSv]</th>
<th>Ratio of the average residual readout to the background dose for the different readout cycles</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>average</td>
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</tr>
<tr>
<td>50</td>
<td>99.8</td>
<td>1.83</td>
</tr>
<tr>
<td>100</td>
<td>50.8</td>
<td>5.06</td>
</tr>
<tr>
<td>500</td>
<td>86.9</td>
<td>3.87</td>
</tr>
</tbody>
</table>

Figure 1 shows the average ratio of the residual doses to the initial residual dose (readout -1) after respective readout cycle for all the TLD chips irradiated to the different doses. It can be seen that the uncertainty value becomes smaller with the number of residual readouts. The average ratio of the tenth residual readout is 1.09 with an uncertainty of ~23%. Therefore, this average value could be used to estimate the magnitude of the residual background dose for the presented dose range.
Figure 1. The ratio of the residual dose from the repetitive readouts, as a function of the readout cycle number.

A function was fitted to the experimental ratio of the N-th readout to the pre-irradiation background value (denoted as R). Equation 1 is the formula of the fitted function, which is shown in figure 1 ($R^2=0.996$). $D_{Bg}$ is the value of the measured initial residual dose (read-1) and N is the readout cycle.

$$R = \frac{D_{Bg}}{23} \times N^{-0.5}$$  \hspace{1cm} (1)

In practice, $D_{Bg}$ is the pre-irradiation background and is not known, if dedicated measurements are not performed before irradiation. If several consecutive measurements of the residual dose ($D_R$) will be made after the irradiation, the pre-irradiation background value ($D_{Bg}$) can be evaluated from equation 2, which is based on equation 1. $D_{Bg}^2$ is the parameter of the fitted function of the residual dose vs. the readout cycle given in equation (2).

$$D_R = \frac{(D_{Bg})^2}{23} \times N^{-0.5}$$  \hspace{1cm} (2)

CONCLUSIONS

The background during the readout of a TLD chip is a combination of signals from different sources. It is defined as the signal of a non-irradiated TLD chip, but the dosimeters in use have an irradiation history, which may contribute significantly by an unknown residual dose. The common practice of a single additional readout after the dose evaluation readout may not be accurate, depending on the readout history of the dosimeter. Consecutive readings after the dose evaluation readout produce diminishing residual dose values, as the deep traps are gradually emptied.

A function was fitted to the relative residual dose values vs. the readout cycle, which can be employed to evaluate the pre-irradiation background. It was shown that this value is not dependent on the dose the TLD crystal was exposed to up to at least 500 mSv, thus it is a more accurate background evaluation as the practice of a single additional readout.
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Glow Curve Prediction Based on the Thermal Behavior of a TLD Detector Heated by a Routine Temperature Profile

R. Rozenfeld\textsuperscript{2}, T. Bar-Kohany\textsuperscript{2*}, M. Weinstein\textsuperscript{1}, A. Abraham\textsuperscript{1}, U. German\textsuperscript{1}, Z.B. Alfassi\textsuperscript{2}, G. Ziskind\textsuperscript{2}

\textsuperscript{1} Nuclear Research Center Negev, P.O.B 9001, 84190 Beer Sheva, Israel.
\textsuperscript{2} Ben-Gurion University of the Negev, 84105 Beer Sheva, Israel.

Corresponding Author: kahany@bgu.ac.il

INTRODUCTION

An axisymmetric 2-dimensional numerical model was developed to predict the actual temperature profile evolution in a crystal contained in a TLD card that is used in advanced gas heating TLD reader systems\textsuperscript{(1, 2)}. A commercial TLD dosimeter card consists of a LiF crystal covered by Teflon (PTFE) foils. The heat transfer from the gas jet to the TL crystal is time-dependent, therefore there is a time lag between the jet temperature and the actual crystal temperature. This lag is influenced by geometrical parameters and material properties, including the Teflon protecting layer.

The jet temperature is monitored continuously, but the crystal time-dependent temperature is not known. The glow curve obtained during the heating of the TLD crystal contains the basic information needed to evaluate the radiation exposure and the validity of the readout. Its shape is correlated to the actual temperature of the TLD crystal, therefore its prediction can be made only by evaluating the crystal thermal behavior. In a previous work\textsuperscript{(1)} the evolution of the LiF temperature was calculated for a step temperature function. The most widespread use of TLD is in routine monitoring, where a vast number of cards and limited readout time are the major constraints. As a result, most of the cards must be read employing a high-rate heating profile. Additional constraints, such as the Teflon cover, further limit the temperature for maximum readout, since Teflon can be damaged at temperatures above 575K. The usual temperature profile used in a routine dosimetry laboratory is a preheat to 323K (less than 0.5 s) and a linear heating rate of 25K s\textsuperscript{-1} up to 575K, then continuing at that temperature for a preset time.

The present study deals with evaluation of the temperature evolution in a typical TLD package when applying the routine temperature profile mentioned above and with prediction of the resulting glow curve.

MATERIALS AND METHODS

The numerical calculations

The numerical approach used to evaluate the time-dependent temperature behavior in the system is based on the solution of transient two-dimensional conservation equations and is described in detail elsewhere\textsuperscript{(1, 2)}. ANSYS®-12.1 (formerly Fluent) software\textsuperscript{(3)} was used as the platform. A comprehensive validation work has been done by solving several models by the present numerical method in order to ensure the reliability of its predictions\textsuperscript{(1)}. In particular, the numerical results were compared with an analytical solution for the plane stagnation flow\textsuperscript{(4)}, and with the works which investigated transient confined jets\textsuperscript{(5)}.

The glow curve simulation
A first-order kinetic model \(^{(6)}\) was applied to describe each peak in the glow curve. In the temperature range to which the crystal is exposed during the read-out, mainly peaks no. 1 through 5 are expected. However, since peak 1 decays very fast at room temperature, its contribution will be negligible. For gamma irradiation, the peaks higher than peak 5 are also not significant.

In order to facilitate calculations, the original Randall-Wilkins equation is substituted with an approximate expression – Eq. (5-16) from Pagonis et al. \(^{(7)}\):

\[
\frac{I(T)}{n_0} = s \cdot \exp \left(-\frac{E}{k_B T}\right) \cdot \exp \left(-\frac{s_k B T^2}{\beta E}\right) \cdot \exp \left(-\frac{E}{k_B T}\right) \left(1 - \frac{2k_B T}{E}\right)
\]

where \(I\) is the intensity; \(E\) and \(S\) are the activation energy and frequency of the specific peak; \(n_0\) is the density of the trapped electrons for that peak; \(\beta\) is the actual heating rate of the crystal, thus being the result of the heating profile used; \(k_B\) is the Boltzmann constant; and \(T\) is the time-dependent crystal temperature (in Kelvin).

The values of \(n_0\) are dependent on the radiation dose absorbed by the crystal. This parameter is different for the different peaks.

The heating rate, \(\beta\), is determined based on the simulation results, as:

\[
\beta(t) = \frac{\Delta T}{\Delta t} = \frac{T_{\text{final}} - T_i}{\Delta t}
\]

where \(\Delta T\) is the simulated increase in the crystal temperature over the time period \(\Delta t\). Typically, a value of \(\Delta t=0.02\)s was used in the present calculations.

The intensity for each peak, \(I_n\), normalized by the density of the trapped electrons for that peak, \(n_0\), is obtained. Thus, while the exact shape of the glow curve depends on \(n_{02}, n_{03}, n_{04},\) and \(n_{05}\), the position of each peak (maximum) in time can be determined by the calculation. Since the relative intensity of the different peaks depends on experimental characteristics, experimental data is necessary to fit the complete curve. The experimental data is in the form of intensity-time dependence, which is an experimentally obtained glow curve.

The experimental system
Glow curves were obtained experimentally with LiF:Mg,Ti Bicron/Harshaw (now Thermo Inc.) standard cards containing crystals of dimensions 3 x 3 x 0.9 mm. The readout was performed by a 6600 TLD reader (Bicron/Harshaw) employing hot nitrogen heating. The system is operated routinely at the dosimetry laboratory of the Nuclear Research Center-Negev (NRCN). The pre-heat conditions, heating rate and maximum temperature are determined through a PC computer which controls the reader. The glow curve of each chip is digitized to a 200 channel spectrum.

RESULTS
The temperature evolution
Figure 1 presents the simulated minimum and maximum crystal temperature evolution for the routine temperature profile of the jet (a linear heating rate of 25K s\(^{-1}\) up to 575K, then continuing at that temperature for a total time of 30 seconds). The temperature difference across the crystal in this case reaches a maximum of 13K during the readout process, and the time needed for the crystal to reach the temperature of the jet is about 30 seconds.
The glow curve

As mentioned before, the glow curve can be predicted based on the computed crystal temperature profile. A comparison between the experimental results and results that were obtained by the numerical model, using equations (1) and (2), is presented in Figure 2. Results from numerical model were fitted to match the total area of the glow curve. The good agreement between experimental and simulated locations of the peaks is due to a rather accurate prediction of the lag between the jet and crystal temperatures.

Figure 1 – Simulated temperature evolution in a 0.9 mm crystal and its Teflon cover for a linear jet heating profile (a heating rate of 25K/s up to 573K).

Figure 2 – Glow curve of a typical TLD card – comparison between experimental results and numerical model's results.
Conclusions
The temperature profile evolution in a LiF crystal contained in a TLD card was evaluated by numerical modeling of the flow field of the gas jet and the heat transfer from the jet to the crystal covered with a Teflon layer. The LiF simulated temperature profile has been substituted into a first-order kinetic model equation to obtain the glow curve. The results have been compared to experimental findings.
A good agreement was obtained between the simulated locations of the peaks in the predicted glow curve and the location in experimentally measured glow curves. This agreement indicates an accurate evaluation of the time dependent temperature evolution in the TLD crystal.
The tool which was developed in this work is versatile and can be employed to check the influence of different experimental parameters (like the Teflon thickness, chip thickness, jet position etc.) on the shape of the glow curve.

REFERENCES
Parametric Analysis of Temperature Profiles due to a Nitrogen Jet Impinging on a TLD Detector

I. Cohen\textsuperscript{1,2}, U. German\textsuperscript{1}, G. Ziskind\textsuperscript{2}, T. Bar-Kohany\textsuperscript{2}

\textsuperscript{1} Nuclear Research Center Negev, P.O.B 9001, 84190 Beer Sheva, Israel.
\textsuperscript{2} Ben-Gurion University of the Negev, 84105 Beer Sheva, Israel.

Corresponding Author: kahany@bgu.ac.il

INTRODUCTION

A typical commercial thermo-luminescence dosimeter (TLD) card consists of a LiF crystal covered by Teflon (PTFE) foils. In the modern TLD readers, the TLD crystals are heated by a hot nitrogen jet. The jet temperature is controlled, whereas the temperature of the crystal itself is not measured. Hence, the temperature presented on the glow curve is the jet temperature, rather than the crystal's temperature.

The usual jet temperature profile is preset to a linear section followed by a constant temperature that is limited to 573K by the melting temperature of the Teflon. The crystal's temperature, however, is far from linear during the first couple of seconds at the beginning of the jet linear section and during the transition from the linear section to the constant section.

The present work focuses on reducing the degree of deviation from linearity of the crystal's temperature profile. This can be done by controlled variation of the temperature profile of the nitrogen jet. A numerical model was developed to predict the actual temperature profile evolution in a crystal contained in a TLD \cite{1, 2}. This model was used to calculate the crystal temperature due to the hot jet temperature variation.

PHYSICAL AND NUMERICAL MODEL

The TLD system investigated herein is a commercially available one. Figure 1 presents a schematic description of the heating chamber, which preserves its relevant physical features and dimensions. The LiF crystal considered is $3 \times 3 \times 0.9 \text{mm}^3$, is fully coated by a 0.06mm Teflon layer, and bordered by a "plastic" insulator which was modeled with Teflon thermal properties.

Thermal behavior of both the LiF crystal and the Teflon layer, as a result of heating by several nitrogen jet temperature profiles, was modeled. A typical velocity of 100m/s jet entering from a single round nozzle (SRN) was studied.

The results were obtained with a CFD solver (ANSYS-FLUENT\textsuperscript{®}) \cite{3}.
RESULTS AND DISCUSSION

Figure 2 presents the temperature evolution of the LiF crystal and Teflon as a result of a conventional jet profile used in routine work (a 25K/s linear heating to 573K after a fast preheat to 323K). These results were obtained in a previous work (1,2).

Heat transfer from the jet to the crystal is time dependent. Furthermore, the Teflon cover leads to a non negligible temperature lag between the jet temperature and that of the crystal, thus for the deviation from linearity. For the conventional jet temperature profile mentioned above, the temperature lag after 15sec is about 50K and the LiF crystal reaches the jet’s temperature after about 30 seconds of heating.
In the present work the thermal behavior of the LiF crystal and the Teflon layer, as a result of heating by non conventional nitrogen jet temperature profiles was checked. The purpose was to improve the temperature profile in the crystal in the following respects:

- to increase the linearity of the crystal’s temperature.
- to shorten the heating period.

These goals, however, should be achieved without damaging the Teflon cover. Hence, temperature profiles for the nitrogen jet were designed such that the Teflon temperature will not exceed 573K at any point, to avoid its damage.

**Figure 3** presents an example of temperature profiles of the LiF and of the PTFE (Teflon) for 0.9 mm LiF chips due to a varied jet temperature profile.

The above suggested temperature profile for the nitrogen jet leads to a temperature lag of ~23K between the LiF averaged temperature to the jet’s as compared to the traditional heating profile (see Figure 2).

Moreover, the LiF reaches the jet’s temperature after about 18 second, as compared to ~30 seconds for the traditional heating profile.

Thirdly, the linearity of the crystal’s temperature profile is improved.

**Conclusions**

A parametric analysis of the temperature evolution in a LiF crystal and its PTFE cover when applying different jet profiles was conducted.

All profiles were chosen so that the PTFE cover will not reach its melting temperature.
Jet profiles were designed to reduce deviation from linearity for the crystal's temperature profile. The deviation from linearity occurs especially during the first stage of the heating process and during the transition between the linear section of the jet profile and the constant section.

As the transition time to reach the maximum temperature is reduced, the light emission efficiency from the crystal during the readout is expected to be improved, as compared to that when using a conventional temperature profile.

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The Energy Dependence of the Supralinearity of Peaks 7 and 8 in LiF:Mg,Ti (TLD-100) Can Lead to Incorrect Evaluation of the Radiation Dose to Cancer Patients

H. Datz1, Y.S. Horowitz2, L. Epstein1, L. Oster3 and M. Margaliot1

1 Radiation Safety Division, Soreq Nuclear Research Center, Yavne, Israel
2 Physics Department, Ben Gurion University of the Negev, Beer-Sheva, Israel
3 Physics Unit, Sami Shamoon College of Engineering, Beer-Sheva, Israel

Corresponding Author: datz@soreq.gov.il

INTRODUCTION

LiF:Mg,Ti (TLD-100 in one of its commercial forms) continues to be one of the major thermoluminescence (TL) detectors of choice in many radiation dosimetry applications. As in most measurement modalities, the ease or difficulty in using TLD-100 depends on the degree of precision required in the dosimetric application. For most applications in personnel and environmental dosimetry an accuracy of ~30% (95% CI) is considered acceptable. However, if higher levels of precision of the order of a few percent at the same confidence level are desired, e.g. in clinical radiotherapy/brachytherapy, an intimate knowledge of the intricate behaviour of this material is necessary. Illustrative examples can be seen in the application of LiF-TLDs to low-energy x-ray interstitial brachytherapy using 125I (27.4 keV) and 103Pd (20.2 keV) x-rays and to high-energy 6 to 18 MeV x-rays for radiotherapy. According to international recommendations, the precision of the delivered dose to tumors should be at least ± 5% (95% CI) (1, 2). The extended range of energies used in these various clinical applications coupled with the required high precision raises certain questions concerning the errors introduced by inappropriate/non-universal readout, material annealing and handling procedures.

Dose response supralinearity

One of the bothersome aspects of the characteristics of LiF-TLD is the onset of non-linear dose response (supralinearity) at dose levels of ~ 1 Gy. At dose levels of clinical significance the correction for non-linearity can be of the order of ~ 5-10%, and, therefore, when high precision is required the non-linearity is of very significant importance. The dose response non-linearity can be characterized by the following parameters: (i) critical dose threshold, Dc, at which supralinearity begins (3),(ii) maximum value of the supralinearity (f(D)max), and (iii) value of the dose, Dmax, at which f(D)max occurs.

The normalized TL dose response, f(D), is given by:

\[ f_i(D) = \frac{F_i(D)}{F_i(D^\ast)} / D/D^\ast \]

where \( F_i(D) \) is the TL signal intensity of the i’th glow peak at dose D and \( F_i(D^\ast) \) is the TL signal of the i’th glow peak at a standard or low dose preferably in the region of linear dose response. For peak 5 one finds \( D_c = \sim 1 \) Gy for 60Co and 90Sr/90Y irradiation with \( f(D)_{max} = 3-4 \) at \( D_{max} \) of \( \sim 200-400 \) Gy (4). \( f(D)_{max} \) decreases with decreasing photon energy reaching values of \( \sim 1.7 \) for x-rays of 8.1 keV effective energy (5). Relatively little is known about the behavior of the energy response of the supralinearity of the high temperature thermoluminescence (HTTL) in LiF:Mg,Ti, composed mainly of peak 7 and peak 8 occurring at temperatures of \( \sim 270 \) °C and \( \sim 310 \) °C respectively, when a heating rate of 1°C s\(^{-1}\) is employed. A deconvoluted glow curve of the HTTL following a three-stage background subtraction
protocol (6) is shown in Figure 1. These peaks appear with roughly 5-15%, of the TL signal intensity of peak 5 at dose levels of ~ 5 Gy (depending on annealing protocol (7)) and inclusion of even part of their intensity in the measured TL signal can, therefore, impact the dose response supralinearity at the level of a few percent.

![Deconvoluted glow curve based on first-order kinetics following 90Sr/90Y irradiation to 500 mGy after background subtraction. The inset shows the HTTL peaks.](image1)

Figure 1: Deconvoluted glow curve based on first-order kinetics following $^{90}$Sr/$^{90}$Y irradiation to 500 mGy after background subtraction. The insert shows the HTTL peaks.

The present work describes experimental measurements of the dose response of the HTTL glow peaks following $^{90}$Sr/$^{90}$Y beta ray irradiations at an average energy of 540±50 keV (as shown in Figure 2) and their incorporation with other dose response results at 100 keV x-ray energy (irradiated at the European Synchrotron Radiation Facility (ESRF)) (8), $^{60}$Co (1250 keV) (9) and 8.1 keV effective energy x-rays (5).

![Theoretical and FLUKA version calculated 90Sr/90Y energy spectra. The average energy taking into account multiple scattering is ~ 540 keV due to the enhanced presence of low energy electrons.](image2)

Figure 2: Theoretical (11) and FLUKA version (2011.2.3) (12, 13) calculated $^{90}$Sr/$^{90}$Y energy spectra. The average energy taking into account multiple scattering is ~ 540 keV due to the enhanced presence of low energy electrons.

**METHODS**

The experiments were carried out on 3.2 mm x 3.2 mm x 0.89 mm TLD-100 samples. The samples were pre-irradiation annealed at 400 °C for one hour. The irradiations were carried out at levels of dose between 0.01 and 5000 Gy using $^{90}$Sr/$^{90}$Y source and read-out using a Thermo 3500 manual reader.
RESULTS

Figure 3 illustrates the energy dependence of $f(D)_{\text{max}}$ for peaks 5, 7 and 8 for several radiation sources and particle energies: 8.1 keV effective energy\(^5\) and 100 keV x-rays\(^8\), \(^{90}\)Sr/\(^{90}\)Y beta ray with average energy of 540 keV (present work) and \(^{60}\)Co gamma ray with average energy of 1250 keV\(^9\). The values of $f(D)_{\text{max}}$ are summarised in Table 1 and show a very strong dependence of $f(D)_{\text{max}}$ for both peaks 7 and 8 on energy (Figure 3).

![Figure 3: The dependence of $f(D)_{\text{max}}$ on particle energy for peaks 5, 7 and 8](image)

**Table 1:** Energy dependence of the supralinearity, $f(D)_{\text{max}}$, of Peaks 4, 5, 7 and 8 for several radiation sources and particle energies

<table>
<thead>
<tr>
<th>Radiation Source</th>
<th>Energy [keV]</th>
<th>Peak 4</th>
<th>Peak 5</th>
<th>Peak 7</th>
<th>Peak 8</th>
</tr>
</thead>
<tbody>
<tr>
<td>(5) Gamboa-deBuen et al., 1998</td>
<td>8.1</td>
<td>1.7 ± 0.1</td>
<td>3.3 ± 3.0</td>
<td>7.5 ± 0.8</td>
<td></td>
</tr>
<tr>
<td>(8) Livingstone et al., 2010</td>
<td>100</td>
<td>2.6 ± 0.6</td>
<td>3.2 ± 0.3</td>
<td>9.6 ± 1.0</td>
<td>38 ± 8</td>
</tr>
<tr>
<td>Datz et al. (this work)</td>
<td>540</td>
<td>2.2 ± 0.5</td>
<td>3.3 ± 0.3</td>
<td>17.9 ± 2.7</td>
<td>133 ± 27</td>
</tr>
<tr>
<td>(9) Massillon et al., 2006</td>
<td>1250</td>
<td>2.3 ± 0.1</td>
<td>3.6 ± 0.3</td>
<td>31 ± 8</td>
<td>207 ± 52</td>
</tr>
</tbody>
</table>

* Radiation Source: (5) X-Ray; (8) \(^{90}\)Sr/\(^{90}\)Y; (9) \(^{60}\)Co

CONCLUSION

The dose responses of glow peaks 7, 8 in the glow curve of TLD-100 are very strongly dependent on photon/electron energy. Previously published data on $f(D)_{\text{max}}$ at photon energies of 1250 keV, 100 keV and 8.1 keV effective energy coupled with new data at 540 keV using \(^{90}\)Sr/\(^{90}\)Y beta rays reveals that the maximum supralinearity $f(D)_{\text{max}}$ decreases from values of ~200 and ~30 at 1250 keV, through intermediate values at 540 keV and 100 keV, to values of ~30 and ~3 at 8.1 keV effective energy.

In clinical radiation therapy, in which high precision < ± 5% (95% CI) is required to optimize tumor killing and minimize damage to surrounding tissues, the applied particle energy ranges over three orders of magnitude, from keV to MeV energies. It follows that the LiF-TLD dose response correction required for these various radiation modalities may be significantly different (at the few percent level) if all or part of glow peaks 7 and 8 are included in the TL signal. To emphasize this point, Figure 3 and Table 1 show that at 8.1 keV the value of $f(D)_{\text{max}}$ of peak 7 is twice that of peak 5 and at 1250 keV this difference has
increased to a factor of 9!. Even much greater differences are observed for peak 8. Thus, although the contribution of the total HTTL intensity is only 5 - 15% at dose levels of ~1 to 2 Gy it follows that inclusion of even a part of the HTTL can significantly affect dose estimations/corrections in this range at the few percent level and hence to influence the desired goals of the treatment. It is important to note that the ratio HTTL to peak 5 increases when the particle energy decreases which can also lead to erroneous dose estimations. Of additional significance is the great variability of the behavior of peak 8 from sample to sample (7). It is therefore highly recommended that peaks 7 and especially peak 8 be excluded from the dosimetric TL signal either by limiting the max temperature of readout to ~ 240˚C (at a glow curve heating rate of 1˚C s\(^{-1}\)) or by glow curve deconvolution.

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External Dosimetry of Persons Occupationally Exposed to Ionizing Radiation in Israel During the Years 2004-2010

S. Primo, H. Datz and G. Haquin
Radiation Safety Division, Soreq Nuclear Research Center, Yavne, Israel

Corresponding Author: datz@soreq.gov.il

INTRODUCTION

This report presents the personal dosimetry for external radiation exposure of workers in Israel during the years 2004-2010. The dose data were obtained from the External Dosimetry Lab (EDL) and the National Dosimetry Archive at the Radiation Safety Division at Soreq Nuclear Research Center (SNRC).

The EDL is ISO 17025 certified and provides its services to approximately 13,000 users throughout the country from various sectors such as medical, industrial and research. The workers are monitored mainly monthly for x-rays, γ radiation and where necessary to fast and thermal neutrons. About 95% of the monitored workers who might be exposed to x-rays, γ or β, are using Thermoluminescence Dosimeter (TLD) badges which contains 3 TLD chips type LiF:Mg,Ti as known as TLD-100. Those workers who might be exposed also to thermal or fast neutrons are using LiF:Mg,Ti badges which contains 4 TLD chips : 3 chips which are not sensitive to thermal neutrons (TLD-700) and a forth chip which is very sensitive to thermal neutrons (TLD-600). For fast neutron dosimetry a polyallydglycol carbonate (CR-39) foil is added. Workers who might be exposed to x-rays, γ or β at their finger tips are using a ring which contains one TLD-100 chip.

The TLDs are used to determine the personal dose equivalent H_{p}(10) after a reduction of the TLD reader photomultiplier noise and natural background during the exposure period. Doses below 0.1 mSv are considered as zero doses. It is important to note that all doses below and above 0.1 mSv are stored for at least 50 years in the National Dosimetry Archive. Individual surface doses (skin doses) are normally measured using the same TLDs but the results are given as H_{p}(0.07). TLDs in the form of rings are wearing on the fingers to measure the irradiation of exposed extremities. Here, doses below 0.5 mSv are consider as zero doses.

RESULTS

The evolution of the personal dosimetry during the years 2004 to 2010 in Israel is shown in Table 1. Its shows the number of total body badge (chest) monitored workers, the number of measurably exposed workers (monthly dose above 0.1 mSv), the collective dose which is the sum of the doses off all measurably exposed workers, the annual average individual dose per monitored worker which is the collective dose divided to total body badge monitored workers and the annual average individual dose per measurably exposed worker which is the collective dose divided to all the measurably exposed workers. The table represents data for all workers in Israel.

Table 1 - Monitored workers in Israel during the years 2004 to 2010.
The number of individuals occupationally monitored as the result of external irradiation showed an average annual increase of 1%. The number of measurably exposed workers showed an average annual decrease of 5%. The collective dose, showed an average annual decrease of 7% and it is 1.69 man-Sv in 2010 (2.80 man-Sv in 2004). The annual average individual dose per monitored worker and per measurably exposed worker showed an average annual decrease of 8% and 3%, respectively.

The distribution of the individuals occupationally monitored in the different working sectors were as follows: medical - 69% (2004) to 73% (2010), industry - 22% (2004) to 21% (2010) and research - 9% (2004) to 6% (2010).

In 2010, the distributions of the various sectors to the collective dose were as follows: medical 84%, industry 15% and research 1%. The same distribution was recorded in 2004.

At the major monitored and exposed sector - the medical sector, the number of individuals occupationally monitored showed an average annual increase of 2%. The number of measurably exposed workers showed an average annual decrease of 5%. The collective dose, showed an average annual decrease of 7%. The annual average individual dose per monitored worker and per measurably exposed worker showed an average annual decrease of 9% and 5%, respectively. The contributions of the different units at the medical sector to the number of individuals occupationally monitored and exposed were as follows: x-ray and angiography 47%, surgery and catheterization 35%, nuclear medicine 9%, oncology 5%, and dentistry 4%.

Figure 1 shows the number of monitored and measurably exposed workers together with the collective dose for nuclear medicine. It can be observed that since 2005-2006 there is a sharp decrease of the collective dose (average annual decrease of 3%). The decrease can be explained as a result of changes in working procedures in Israel. Along these years the Radiopharmaceutical Division of SNRC started to supply nuclear medicine products in the form of unit doses instead of bulk batches, reducing to minimum the preparation work by nuclear medicine staff, which brought to radiation doses reduction. It is important to note, that those saved doses disappeared and were not added e.g. to SNRC Radiopharmacy staff. From the other hand the percentage of the measurably exposed workers in nuclear medicine is relatively high being 30% in 2010 (40% in 2004) from the total monitored workers while the average annual dose per monitored worker is higher than 2 mSv. The average dose in the
nuclear medicine field is about 3.5 times higher than the doses at the other units in the medical sector.

![Graph showing the number of monitored and measurably exposed workers and collective dose for nuclear medicine clinics during 2004-2010.](image)

**Figure 1** - The number of monitored and measurably exposed workers together with the collective dose for nuclear medicine clinics during the years 2004-2010.

At dental clinics the number of individuals occupationally monitored showed an average annual increase of 1%. The number of measurably exposed workers showed an average annual increase of 13%. The collective dose, showed an average annual increase of 42%. The annual average individual dose per monitored worker and per measurably exposed worker showed an average annual increase of 41% and 23%, respectively. Although the total number of measurably exposed workers is 17 workers, the results demonstrate that there are less exposed workers who exposed to higher doses. The increase in the average doses can be explained by the fact that dental clinics are not a well controlled field as other units in the medical sector.

Collected data for extremity doses during 2004 to 2010 indicate an increase in the number of monitored workers wearing TLD rings and an increase at the collective dose. In 2010, the contributions of the various clinics\units\sectors to the extremity collective dose were as follows: nuclear medicine 49%, surgery and catheterization 30%, x-ray and angiography 15%, industry 5% and research 1%.

**CONCLUSION**

This study summarizes a small part of the data and trends regarding the external occupational doses in Israel during the years 2004 to 2010. The findings show a general increase trend in the number of monitored workers along a decrease trend in the collective dose and the number of the measurably exposed workers.
These trends can be explained by technological and practices improvements, appropriate safeguards in the workplace in parallel with an increased supervision and its enforcement which led to heightened awareness and wider use of TLDs for monitoring. The trends and findings emphasize the need of training the workers for aright and radiation safer work, supervision and enforcement and transition for "radiation saver" technology as much as possible.

ACKNOWLEDGMENTS
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Detector Design Considerations for a Beta Aerosol Monitoring System

E. Gonen¹, U. Wengrowicz¹, A. Osovizky², J. Nir¹, B. Sarusi¹, I. Levin¹, T. Mazor¹, Y. Kadmon¹, I. Orion³

¹ Nuclear Research Center - Negev, Beer-Sheva, Israel
² Health Physics Instrumentation Department, Rotem Industries Ltd, Israel
³ Department of Nuclear Engineering, Ben Gurion University of the Negev, Israel

INTRODUCTION
Detector design considerations for a continuous air monitoring system are presented. The system, called “BASMT” (Beta Aerosols Sampling Monitor), is developed for monitoring and measuring the concentration of airborne beta emitting radionuclides such as $^{137}$Cs. The beta emitters concentration is estimated by discrimination of beta from Radon and the Gamma background.

The system includes a processing and display unit, a sampling unit and an air pump. The “BASMT” is a stationary system; the pumped air passes through a 2.0 cm diameter filter. The airborne aerosols, including radon radioactive aerosols, accumulate on it. This filter is monitored by a beta and alpha sensitive detector. The instrument discriminates the alpha radon background by means of pulse height and pulse shape analysis. According to the alpha/beta decay ratio of radon, an accurate measurement of the alpha radon enables a good estimation of the beta radon measurement. For precise estimation of the $^{137}$Cs airborne concentration an algorithm was implemented for decreasing gamma background radiation and beta from radon emissions from the aerosols accumulated on the filter.

METHODS
Beta detectors are sensitive to gamma radiation as well. In order to improve the sensitivity and to decrease the detection concentration limits, the gamma background should be reduced to minimum. This is achieved by gamma compensation detector and a lead shield. This shield enclosures both detectors and the filter.

The lead shield reduces the gamma background by more than one order of magnitude. The compensation detector is identical to the filter monitor detector. Statistically, Gamma rays contribute the same signal on both detectors. Therefore; for gamma compensation, the system subtracts the pulses measured in the compensation detector from these measured by the filter detector.

Traditionally, this type of systems requires a large and heavy lead shielding. An improved design on the detector configuration was implemented in order to maintain the same detection limits, but reducing the dimensions and weight of the lead shielding. Several types of detection technologies and configurations were investigated. G.M (Geiger Muller tubes) based technologies, offer no particles recognition, hence no beta radon subtraction can be achieved. Since G.M sensors are sensitive to gamma rays, gamma compensation is also required. Silicon solid state PIN based technologies have excellent energy resolution for alpha and beta discrimination, but they are too vulnerable for applications such as continuous air monitoring in harsh environment. Furthermore, the signal from Compton electrons originated from gamma interaction, has similar energy distribution to the $^{137}$Cs beta, so good gamma discrimination can’t be achieved by spectrum analysis using a single detector.
The selected detection configuration consists of two detectors. One detector faces the filter for alpha and beta measurement, the second detector is employed for Gamma compensation measurements. Each sensor based on a dual phosphor (ZnS and Plastic) scintillation detector. The ZnS coat is a thin phosphor layer that is highly sensitive to alpha radiation. By selecting a thin plastic scintillator, high beta sensitivity is achieved, while keeping low gamma sensitivity. For similar gamma response, both detectors composed in the same mode. The width of the plastic scintillator was selected in order to achieve the optimum beta efficiency with minimum gamma sensitivity. Figure 1 shows the sensor unit of the “BASMT”, the filter detector (behind the filter removal handle), assembled above the compensation detector. The lead shield, colored red in figure 1, enclosures both detectors.

Figure 1: Detector Unit of the "BASMT"

The radon radioisotope $^{212}$Rn, decays to by emission of three alpha and two beta particles. The energies of the alpha particles are 6.0MeV, 5.49MeV and 7.68MeV; the maximum energies of the particles 0.67 and 3.26MeV;

The range of alpha radon particles in a homogeneous ZnS layer was calculated SRIM program [1] of J. Ziegler; this program enables range calculation of particles in matter. The ZnS density is 4.6g/cm$^3$ and the stopping power for Radon, varies from 0.45 to 0.36 MeV/(mg*cm$^{-2}$) according to the alpha energy. For full energy deposition of all the alpha Radon particles, a ZnS coating layer of 32µm depth (~14mg/cm$^2$) is required, this layer absorbs most of the beta particles with energies lower than 100 keV [2]. $^{137}$Cs emits beta particles with maximum energy of 512 keV (94.6%) and 1173 keV (5.4%), the average beta energy is 156 keV. The “BASM” is developed for $^{137}$Cs monitoring measurements, hence a ZnS layer of 14 mg*cm$^{-2}$ can absorb a large quantity of the $^{137}$Cs beta.

Theoretical calculations and MCNP simulations were performed in order to estimate the optimum width of the ZnS; a layer that reduces the $^{137}$Cs beta absorption while still enabling a large energy deposition of the alpha particles there. It was found that by reducing the ZnS coating layer to about 10 mg/cm$^2$, the energy deposition of the alpha particles is large enough, enabling to distinguish alpha/beta particles by means of pulse amplitude discrimination. Only a small fraction of the $^{137}$Cs beta is absorbed in this layer.

Together with the calculation to define the optimum ZnS coating, the depth of the plastic scintillator was also optimized. In order to decrease the gamma interactions, the scintillator should be the thinnest as possible. If the scintillator is too thin, the energy deposition of the beta particles in crystal could be lower than the noise threshold.

The thickness of the plastic scintillator was estimated using MCNP simulations. MCNP has several types of tallies to record different interactions of radiation. In order to estimate the
beta energy deposition in the plastic scintillator, F8, the pulse height distribution tally was selected. An electron source, shaped as thin plate was selected. The source was located at a distance of 3mm from the detector in order to simulate the aerosol filter in the “BASM”. MCNP includes built-in functions for source probability and bias distribution. In order to achieve a typical beta emission spectrum, the built-in evaporation spectrum function was selected. In this function, the energy distribution of the emitted particles are according to the equation: \( P(E) = C \times E \exp(-E/a) \).

Selecting the value of the parameter \( a = 0.105 \), an emission spectrum similar to the \(^{137}\)Cs beta emission was obtained. The simulated average energy of the emitted electrons was 197 keV and the maximum energy of these electrons 550 keV. According to the simulations, it found that a plastic scintillator of 0.25 cm depth has efficiency higher than 70\% \((2\pi)\) for \(^{137}\)Cs beta emission, while decreasing the gamma interactions to less than 1\%.

**CONCLUSIONS**

Detectors considerations for a beta aerosol monitoring system were described. By using a double ZnS/Plastic scintillator, Radon radiation decay is discriminated from artificial \(^{137}\)Cs beta decay. A gamma compensation detector was implemented in order to reduce the gamma background noise. Statistical fluctuations of the gamma interaction between the two detectors can cause erroneous airborne concentration estimation. In order to decrease the statistical fluctuations, traditionally a large lead shielding is implemented. The presented approach optimized the detector configuration, enables high beta sensitivity and large gamma discrimination. This approach reduces drastically the required lead shield to nearly 10 kg instead of more than 50 kg in similar instruments.

**REFERENCES**


Development of Mobile Digital Multichannel Analyzer/Scaling System

Y.Ifergan¹, U.Wengrowicz¹, A.Osovizky², A.Broide¹, B.Sarusi², I.Brandys¹, T.Mazor¹, Y.Cohen¹ and Y.Kadmon¹

¹ Electronic and Control Laboratories, Nuclear Research Center Negev, Beer-sheva, Israel.
² Radiation Detection Department, Rotem Instrument Ltd, Rotem industrial Park, Israel.

Corresponding Author: yairifer@gmail.com

ABSTRACT

The Mobile Digital Multichannel Analyzer (MMCA2010) is a highly integrated portable spectroscopy workstation based on advanced Digital Signal Processing (DSP) technique. The novel MMCA2010 uniqueness lies in its ability to accommodate any chosen Gamma spectrometry detector. The workstation design motivation was to create a compact, mobile, PC-based system, to enable comfortable and flexible measurement process. In this paper, all the Mobile Digital Multichannel Analyzer subsystems will be discussed.

INTRODUCTION

Nowadays, many of the Multichannel Analyzer (MCA) and Multichannel Scaler (MCS) systems found in the market are physically large or plug-in cards, a design which requires addition of a PC and expensive peripheral nuclear instruments, turning the system non-mobile unit. Consequently, it was important to develop an instrument that should include all periphery components, and would be easy to carry. In addition, the MMCA2010 system (see Figure 1) was specially built for field environmental conditions, including a graphical display to show the radiation level trend versus time and spectroscopic data. The computerized system consist of single electronic board that controls all the system logic activated from the PC, as shown in Figure 2. This board, based on MCs51 family of microcontrollers, was designed by the Electronic and Control Laboratories development team, at the NRCN.

Figure 1: Mobile Digital Multi Channel Analyzer/Scaler
ELECTRONIC CIRCUITRY DESCRIPTION

I. Digital Signal Processor Unit (DSP)

The DSP shapes the pulse produced by the detector (such as a photomultiplier in scintillation detector), into a Gaussian or trapezoidal shape, converting the analog signal to a digital signal [1]. The DSP card output is sent to a computer that stores, displays, and analyzes the data. The DSP card can operate in two modes; In the Pulse high analyzer (PHA) mode, the input pulse are sorted into bins (channels) according to their amplitude while in the MCS mode the DSP card records the counting rate of event as function of time.

II. High Voltage Power Supply Unit (H.V.P.S.)

The H.V.P.S. range required in radiation detector varies from -5000V to +5000V. The H.V.P.S. unit consists of four different power supplies which can be programmable via PC: (each voltage is positive and negative polarity)

1) 5000 Volt for High Pure Germanium (HPGe) detectors.
2) 2000 Volt for scintillation detectors such as NaI(Tl), CsI(Tl) and solid state detectors like CdTe, Cd(Zn)Te. The Power Supply (PS) has a low ripple, specially designed for photomultipliers. The power supplies have a high stability and full protected against continuous short circuits.

III. Low Voltage Power Supply Unit (L.V.P.S.)

The L.V.P.S. unit includes two main functions first, providing the DC voltage to the external PreAmp detector and, to all the internal MMCA2010 electronic components. Therefore, this unit consists of five different switching power supply (+5V, ±12V, ±24V). In switching power supply, there is a relation between noise and range voltage, the wider the range, the louder the noise. Therefore, each one of the P.S was filtered before it supplying voltage.

IV. Switching card

The switching card, based on 8051 microcontroller, is the heart o the MMCA2010 system. The electronic card was developed to enable full control on all the MMCA2010 subsystems. The switching card includes four solid state reed-relays for selecting the adequate H.V.P.S. Hence, when the user chooses a specific H.V.P.S via the Human Machine Interface (HMI) the switching card enables +24V through the reed-relay to the H.V.P.S which required to turn-on. As was reported, the H.V.P.S is programmable; to enable the changing of the high voltage value to the detector, the switching card includes a four digital potentiometer. Those detectors which include a PreAmp, require the DC Voltage (±12v, ±24v) from the MMCA2010 system. As mentioned before this P.S has a switching noise. To reduce this noise, the Switching card include a L.C[1] filter [2]. In addition, the switching card protect against warm temperature of the HPGe detector. It is known fact that providing high voltage to HPGe detector while it is warming is very dangerous. The warming is indicated by “0” logic input through the shutdown connector in the left panel of the MMCA2010 system. In order to protect the HPGe detector, the switching card does not supply +24V to the specific PS and the high voltage is shut down.

[1] An LC circuit consists of an inductor, represented by the letter L, and a capacitor, represented by the letter C.
For flexibility operation with different type of detectors, the MMCA2010 accepts analog signals with both, positive and negative polarities.

![Figure 2: The MMCA2010 workstation block diagram.](image)

**DESIGN ENGINEERING**

To keep the temperature of the equipment stable, several experiments were conducted under both natural and forced convection of heat transfer [3]. These experiments show a better performance of MMCA2010 in temperature variation while it is under forced convection. According to these results the configuration to heat transfer will be forced convection by two fans which be in the upper panel of the suitcase. Figure 3 shows the spectrum energy stabilization of $^{137}$Cs and $^{60}$Co sources at 25°C and 50°C temperatures.
APPLICATIONS
A joint research of Rotem Industries with the Faculty of Agriculture an Environmental Quality Sciences in Rehovot concerning “The Radium Transport in Living Tomato Plant” was preformed with the MMCA2010 system, as shown in Figure 4. The research was measuring the gamma emitted from tomato as consequence of $^{226}\text{Ra}$ existence in the living plant.

SUMMARY
In this paper, the development process and resolved engineering challenges for achieving the advantage of multi-purposed system are discussed. The MMCA2010 workstation provides a unique comprehensive solution for gamma spectroscopy and currently is exploiting by several R&D projects as the measuring system.

REFERENCE


Suppression of Afterglow Effect in CsI(Tl) Scintillation Detector by Heat Treatment

Y. Ifergan1, Z.B. Alfassi2, U. Wengrowicz1
1 Nuclear Research Center Negev, P.O.B 9001, Beer-Sheva, 84190, Israel.
2 Department of Nuclear engineering, Ben-Gurion University of the Negev, Beer-Sheva, 84105, Israel.

Email: Yairifergan@gmail.com

INTRODUCTION
For radiation detectors CsI(Tl) crystal is one of the most efficient inorganic scintillator which is used for gamma measurement. However, CsI(Tl) has one large drawback for use in radiation survey meters because of its afterglow effect. This effect manifests itself by the light pulse emitted even after the radiation source is removed. When this crystal is inserted into high dose rate field it cannot be used for photon detection due to the emission of these afterglow light pulses. Even when moving later to a low radiation field, the field cannot be measured due to these afterglow pulses. This delayed emission can cause pulse pileup in high count rate application, reconstruction artifacts in CT application, and reduced contrast and image blurring in high speed X-ray imaging[1, ii, iii].

This research was taken in order to study the effect of raising the temperature of the CsI(Tl) crystal at the afterglow effect. The main purpose of our study was to remove or shorten the duration of the afterglow effect in survey meters.

RESULTS
Up to dose rate of about 100mR/h the PMT output pulses measured with an oscilloscope are clear signal, showing that the crystal is functioning properly. These pulses can be measured with multichannel analyzer to obtain a spectrum of the gamma photons. For dose rate fields in range of 100-300mR/h the obtained signal was distorted and the measured pulses were piled up on the previous signal leading to an increase in the DC level. Above 1R/h the detector is operating with an afterglow as can be seen by removing the radiation source. Even after the removal of the radiation source there are signals from the detector which decay slowly. The afterglow effect can be seen by the level of the DC in the output of the detector, due to high rates of overlapping AC signals. The time required for the afterglow effect depends on the dose rate of radiation field and the length of the time it stays in this radiation field. Table 1 gives the DC voltage obtained at various dose-rates both at 25°C and at 50°C. It is clear from this table that the DC level is lower at 50°C than at room temperature. The lower DC level is due to the faster decay of the afterglow. Figure 1 shows the decay of the afterglow effect following different irradiation periods at a constant field of 2R/h. Similar curves were found for fields with 1R/h and 4R/h, with exponential fitting on Figure 1, we found to have two primary decay time constants of $\tau_1$ and $\tau_2$. Were the total decay time $V_{\text{decay}}(t)$ could be described by the sum of two exponential decays.

Equation 1

$$V(t) = A_1e^{-t/\tau_1} + A_2e^{-t/\tau_2}$$

The long decay component $\tau_2$ has a significant effect on pulse processing which was observed to be populated by an exponential process with a resulting rise time constant of 200s at field of 1R/h after exposure of 30s at room temperature, when the fast decay component $\tau_1$ was found with
9.05s. Table 2 present the two primary decay component \( \tau_1 \) and \( \tau_2 \) dependence of field and exposure time at 25\(^{\circ}\)C. To determine the dependence of decay DC level to afterglow effect duration, we measured \( \tau_1 \) and \( \tau_2 \) at 50\(^{\circ}\)C (Table 3). It is clear from Table 3 that the long decay component \( \tau_2 \) is shortening compared to the one observed at 25\(^{\circ}\)C. Therefore there is an effect of the temperature on the time of the afterglow effect. An additional method to determine the effect of the temperature on the duration of the afterglow was measured with spectroscopic instrument. After removals of the radiation source, the instrument shows a spectrum that decay time. At the end of the afterglow effect, the signal returns to the background. The length of the afterglow effect as a function of dose rate and temperature for irradiation of 120s is given in Table4. It can be seen in Table4 that for 4R/h for120s exposure, the increase in temperature from 25 to 60\(^{\circ}\)C decreases the afterglow effect time from 20 min to 3s. The afterglow effect time at 60\(^{\circ}\)C is short enough to allow the measurement to flow dose rate field just after leaving the high dose rate area. Thus, it seems that the best solution is to keep the survey meter at 60\(^{\circ}\)C.

![Figure 1: The dependence of the DC voltage in the detector output on the irradiation time for dose-rate of 2 R/h.](image)

<table>
<thead>
<tr>
<th>Radiation Field [mR/h]</th>
<th>DC [V] at 25(^{\circ})C Volt</th>
<th>DC [V] at 50(^{\circ})C Volt</th>
</tr>
</thead>
<tbody>
<tr>
<td>Background</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>5</td>
<td>0.01</td>
<td>0.01</td>
</tr>
<tr>
<td>10</td>
<td>0.02</td>
<td>0.015</td>
</tr>
<tr>
<td>100</td>
<td>0.2</td>
<td>0.17</td>
</tr>
<tr>
<td>500</td>
<td>1.1</td>
<td>0.72</td>
</tr>
<tr>
<td>1000</td>
<td>2.36</td>
<td>1.75</td>
</tr>
<tr>
<td>2000</td>
<td>4.12</td>
<td>3.5</td>
</tr>
</tbody>
</table>
Table 2: Summary- DC voltage decay after irradiation doses between 1, 2 and 4 [R/h] at 25°C

<table>
<thead>
<tr>
<th>Radiation Field and time Exposed</th>
<th>$\tau_1[sec]$</th>
<th>$\tau_2[sec]$</th>
<th>Adj. R-Square</th>
</tr>
</thead>
<tbody>
<tr>
<td>1R/h(30sec)</td>
<td>9.05</td>
<td>201.32</td>
<td>0.94</td>
</tr>
<tr>
<td>1R/h(60sec)</td>
<td>11.98</td>
<td>195.52</td>
<td>0.98</td>
</tr>
<tr>
<td>1R/h(120sec)</td>
<td>21.78</td>
<td>617.67</td>
<td>0.96</td>
</tr>
<tr>
<td>2R/h(30sec)</td>
<td>15.02</td>
<td>601.21</td>
<td>0.96</td>
</tr>
<tr>
<td>2R/h(60sec)</td>
<td>17.38</td>
<td>624.79</td>
<td>0.96</td>
</tr>
<tr>
<td>2R/h(120sec)</td>
<td>24.27</td>
<td>661.65</td>
<td>0.98</td>
</tr>
<tr>
<td>4R/h(30sec)</td>
<td>12</td>
<td>608.46</td>
<td>0.96</td>
</tr>
<tr>
<td>4R/h(60sec)</td>
<td>16.8</td>
<td>590.2</td>
<td>0.97</td>
</tr>
<tr>
<td>4R/h(120sec)</td>
<td>20.92</td>
<td>542.47</td>
<td>0.97</td>
</tr>
</tbody>
</table>

Table 3: Summary- DC voltage decay after irradiation doses between 1, 2 and 4 [R/h] at 50°C

<table>
<thead>
<tr>
<th>Radiation Field and time Exposed</th>
<th>$\tau_1[sec]$</th>
<th>$\tau_2[sec]$</th>
<th>Adj. R-Square</th>
</tr>
</thead>
<tbody>
<tr>
<td>1R/h(30sec)</td>
<td>11.41</td>
<td>88.08</td>
<td>0.93</td>
</tr>
<tr>
<td>1R/h(60sec)</td>
<td>14.77</td>
<td>137.47</td>
<td>0.98</td>
</tr>
<tr>
<td>1R/h(120sec)</td>
<td>15.83</td>
<td>152.26</td>
<td>0.99</td>
</tr>
<tr>
<td>2R/h(30sec)</td>
<td>18.33</td>
<td>282.74</td>
<td>0.98</td>
</tr>
<tr>
<td>2R/h(60sec)</td>
<td>14.66</td>
<td>149.02</td>
<td>0.99</td>
</tr>
<tr>
<td>2R/h(120sec)</td>
<td>15.7</td>
<td>150.08</td>
<td>0.99</td>
</tr>
<tr>
<td>4R/h(30sec)</td>
<td>9.86</td>
<td>96.39</td>
<td>0.99</td>
</tr>
<tr>
<td>4R/h(60sec)</td>
<td>13.98</td>
<td>137.6</td>
<td>0.99</td>
</tr>
<tr>
<td>4R/h(120sec)</td>
<td>13.62</td>
<td>127.03</td>
<td>0.99</td>
</tr>
</tbody>
</table>

Table 4: The Afterglow effect time after expose to 120 seconds in fields with dose-rates of 1, 2 and 4 [R/h] at temperatures 25, 40 and 60°C

<table>
<thead>
<tr>
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<tbody>
<tr>
<td>25</td>
<td>1</td>
<td>120</td>
<td>2 min</td>
</tr>
<tr>
<td>25</td>
<td>2</td>
<td>120</td>
<td>10 min</td>
</tr>
<tr>
<td>25</td>
<td>4</td>
<td>120</td>
<td>20 min</td>
</tr>
<tr>
<td>40</td>
<td>1</td>
<td>120</td>
<td>1.5 min</td>
</tr>
<tr>
<td>40</td>
<td>2</td>
<td>120</td>
<td>2 min</td>
</tr>
<tr>
<td>40</td>
<td>4</td>
<td>120</td>
<td>7 min</td>
</tr>
<tr>
<td>60</td>
<td>1</td>
<td>120</td>
<td>3 sec</td>
</tr>
<tr>
<td>60</td>
<td>2</td>
<td>120</td>
<td>3 sec</td>
</tr>
<tr>
<td>60</td>
<td>4</td>
<td>120</td>
<td>3 sec</td>
</tr>
</tbody>
</table>

CONCLUSIONS
It was found that heating the scintillation detector to 60°C shorten considerably the duration of the afterglow effect to 3s, even after exposure of 120s to field of 4R/h. It is suggested to start heating the detector immediately when reaching high dose rate field and adjusting the calibration to the measured temperature.

REFERENCES


A Modular High Sensitive Radiation Detector for Homeland Security and Post Event Applications

D. Ginzburg¹, A. Manor¹, M. Ellenbogen¹, V. Bronfenmakher¹, D. Shmidov¹, Y. Yehuda-Zada², T. Mazo², Y. Kadmon² and A. Osovizky¹

¹ Radiation Detection Department, Rotem Industries Ltd., Israel
² Electronics&Control Laboratories, NRCN, Beer-Sheva, Israel

Corresponding Author: alon@rotemi.co.il

INTRODUCTION

A modular, high sensitive radiation monitoring system designed for the homeland security radiological requirements and radiological post event applications is presented. The prevention of undocumented and potentially threatening shipment of radioactive and nuclear materials is a problem at seaports, border crossings, rail yards, airports and similar locations that requires the use of sensitive radiation detectors. Furthermore; radiological events such as the Fukushima nuclear incident emphasize the need for sensitive detector for monitoring food and commercial products.

During past years (1991-2007) more than 20 incidents of SNM seizures worldwide were registered and published in the media. Several incidents of industrial nuclear materials used as a dirty bomb were reported, too. The IAEA Illicit Trafficking Database¹ reports about 260 confirmed radioactive incidents since July 2006. Hence, the need for an appropriate solution is obvious. Border monitoring equipment plays a key role in combating illicit trafficking.

Several categories of instrumentation were defined for combating with the illicit trafficking. The conveyor monitor is designed to meet the detection level determined by the standard for RPMs², ³. This type of device has greater sensitivity than hand held monitors and maintains low false alarm rate for continuous monitoring of the baggage flow.

The regulation that determines the contamination level of food commodities, commercial products and drinking water are based on IAEA recommendation. Table 1 summarizes these regulations based on the food product and on the contaminating isotope.

Table 1. Guideline for radionuclide's contamination levels & permitted levels in food

<table>
<thead>
<tr>
<th>Radionuclides in Foods</th>
<th>Guideline Level (Bq/kg)</th>
<th>Estimated exposure dose on the thyroid for an year (mSv)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Infants</td>
<td>Other Foods</td>
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METHOD

To cope with the requirements which were specified in the previous section, a Modular Portable Radiation System (MPRS) was developed. The conceptual design of the system is based on
highly sensitive Gamma detector and an optional neutron channel, electronic data-processing unit and computer interface (see Fig. 1).

The first product developed by this modular approach is the Conveyor Radiation Monitor (CRM), which is a smart radiation monitor for baggage screening at airports and at any other type of checkpoints. The CRM system provides very high degree of sensitivity along with low acquisition and maintenance costs. The electronic data-processing unit and the Windows-based software interface used in the CRM are modification of a standard inventory that is part of broad radiation detection solutions provided by Rotem for more than 15 years worldwide. The system is designed to passively screen and detect radioactive radiation from luggage moving on the conveyor belt. The CRM is designed to meet the strict requirements of the ANSI N42.35 and of the parallel IEC 62244. In order to minimize the false alarm rate the CRM system uses special background update algorithm while interfacing the conveyor control system for its status update (moving or stopped). The detection threshold is continuously updated according to the recently acquired background.

In case of an alarm, the CRM uses the output signal to order the control system to stop the conveyor belt. The system's block diagram is described in Figure 2.

Figure 1. Detailed view of the modular system

Figure 2. Conveyor Radiation Monitor block diagram.

The system's electronic unit interfaces with the conveyor control system using two signals, an input signal for the conveyor operation status and an output signal for stopping the conveyor in case of alarm. This interface along with the implementation of special algorithms significantly reduces the number of false alarms and improves the detection level by considering the background variation. Further significant improvement in the detection level is achieved by implementing an advanced algorithm based on the detector reading profile versus time. The
online computer software provides the user with friendly interface for retrieving the archived data and analyzing the history of alarms. Eventually, the fact that Rotem's CRM is especially designed to meet the international standards for RPM is increasing the reliability of the system. Furthermore, the CRM was installed at one of the busiest and most secured international airports and successfully tested for operations during a pilot period.

![Figure 3. A photo of operational Conveyor Radiation Monitor installation.](image)

**RESULTS**

**Gamma sensitivity**

Prior to the measurements, the ANSI42.35 sensitivity requirements were translated into exposure rates, according to the specified distance of 1m of the source from the detection surface. Following the obtained results and the background measurement, the Minimum Detectable Count Rate (MDCR) for the CRM was calculated. The MDCR was calculated according to the ANSI requirements for the false alarm rate (1 in 1000) and the detection reliability (59 out of 60), and for the conditions of 600s of background measurement and 2s of signal measurement. The calculated MDCR was 97 cps.

Further, an electronics optimization was performed and a shielding for background suppression was used. The results that were obtained from these measurements are presented in Table 2.

<table>
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</tr>
</thead>
<tbody>
<tr>
<td>$^{241}$Am</td>
<td>10.28</td>
<td>0.3</td>
<td>35.87</td>
<td>190137</td>
<td>1901</td>
<td>1185</td>
<td></td>
</tr>
<tr>
<td>$^{60}$Co</td>
<td>6.62</td>
<td>0.3</td>
<td>100.78</td>
<td>766921</td>
<td>7669</td>
<td>6953</td>
<td></td>
</tr>
<tr>
<td>$^{137}$Cs</td>
<td>14.57</td>
<td>0.3</td>
<td>61.82</td>
<td>1113529</td>
<td>11135</td>
<td>10419</td>
<td></td>
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<tr>
<td>$^{57}$Co</td>
<td>1.2192</td>
<td>0.3</td>
<td>2.05</td>
<td>154117</td>
<td>1541</td>
<td>825</td>
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<tr>
<td>Background</td>
<td>71654</td>
<td></td>
<td></td>
<td>717</td>
<td></td>
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</table>
The MDCR was calculated again according to the ANSI requirements and based on the background. The MDCR this time was 102 cps. The expected count rates from the ANSI source were assessed and are presented in Table 3.

Table 3. The expected count rate from ANSI sources and the calculated number of Standard Deviations above background for different sources

<table>
<thead>
<tr>
<th>Source</th>
<th>Measured Sensitivity [cps per μR/h]</th>
<th>Expected count rate for required activity [cps]</th>
<th>Number of standard deviations above background</th>
</tr>
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<tbody>
<tr>
<td>$^{241}\text{Am}$</td>
<td>33.04</td>
<td>488</td>
<td>12.9</td>
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<tr>
<td>$^{60}\text{Co}$</td>
<td>68.99</td>
<td>662</td>
<td>17.5</td>
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<tr>
<td>$^{137}\text{Cs}$</td>
<td>168.55</td>
<td>1030</td>
<td>27.2</td>
</tr>
<tr>
<td>$^{57}\text{Co}$</td>
<td>402.56</td>
<td>304</td>
<td>8.0</td>
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CONCLUSIONS
i. The selected plastic detector has an adequate sensitivity for ANSI42.35 requirements and even beyond.
ii. The plastic detector within CRM configuration design is expected to perform beyond the ANSI requirements for the sensitivity.
iii. The performance in higher background rates should be assessed.
iv. The use of more effective shielding (for SNR enhancement) should be considered and the effects should be measured.
v. For this detector configuration, the expected signal from an ANSI42.35 test source is 2.9 to 4.5 times higher (depends on the source-to-distance case scenario) than the MDCR (4.8 cps) that was calculated according to the ANSI sensitivity requirements.

REFERENCES
The Compliance of a New PRD with the ITRAP+10 Testing Program

A. Osovizky1, D. Ginzburg1, A. Manor1, R. Seif2, M. Ghelman2, M. Ellenbogen1, Y Cohen2, V. Pushkarsky1, V. Bronfenmakher1, Y. Knafo2, Y. Ifargan2, D. Sadan2, E. Gonen2, U. Wengrowicz2, R. Atias2, T. Mazor2, Y. Kadmon2 and Y Cohen2

1 Radiation Detection Department, Rotem Industries Ltd, Israel
2 Electronics & Control Laboratories, NRCN, Beer-Sheva, Israel

Corresponding author: dimgiz@rotemi.co.il

INTRODUCTION

For the past two years a global international effort was devoted for developing a testing and evaluation program of radiation detectors for Homeland Security. This effort has lastly yielded an international cooperative program. The ITRAP+10 was launched in 2011. The program is conducted jointly by the IAEA and the US Government, and the tests are being held in several US national laboratories and European accredited laboratories. PDS-GO is a novel personal radiation detector (PRD) that was introduced in 2010 [1]. The PRD is an alarming instrument that is used to detect γ-emitting radioactive materials for the purpose of crime prevention. A scope of specifications that is defined by various standards [2], [3] enables to address and evaluate the performance of the PRD. The ITRAP +10 tests the instrumentation according to these standards.

The PDS-GO is the first commercially available PRD based on a CsI(Tl) scintillation crystal optically coupled with a Silicon Photomultiplier (SiPM) serving as a light sensor. It was developed based on an intensive research for optimizing the utilization of the SiPM in applications of radiation detection that involve scintillation crystal [4].

In this work we present the development progress that was aimed to enhance the performances of the PDS-GO. The electronic design of the preamplifier was optimized to stabilize the sensitivity of the detector for the required range of ambient dose equivalent rate (up to 10 µSv/h), resulting in a linear response. The noise level was reduced by longer integration time of the scintillation light, that is provided by the long decay time of the CsI(Tl) crystal (1 µs). The low noise level enables the operation of the detector over the required temperature range (up to +50 °C) without a decrease in its responsivity for the lower energy range. In addition, special endeavor was devoted for improvement of mechanical design and user interface, in order to assure the sustainability and reliability of the detector in field usage.

These and other aspects of performance advances, as opposed to the ITRAP +10 requirements, are described in this work in order to present the readiness level of the detector for this evaluation program.

METHOD

The PDS-GO-01G is the latest generation of gamma personal radiation detectors.

The alarming personal radiation detector (PRD) is a device intended for Homeland Security (HLS) applications. This portable device is designed to be worn or carried by security personnel to detect photon-emitting radioactive materials for the purpose of crime prevention. A PRD is required to meet the scope of specifications defined by various HLS standards for radiation detection. It is mandatory that the device be sensitive, small, pocket-size, robust mechanical design and carriable on the user’s body. To serve these specialized purposes and requirements, the PDS-GO-01G was developed. A new radiation detector designed to meet the performance criteria established for counterterrorist applications.
PDS-GO-01G is the first commercially available PRD based on a CsI(Tl) scintillation crystal that is optically coupled with a silicon photomultiplier (SiPM) serving as a light sensor. These sensitive pocket-sized instruments are designed to detect, locate and quantify any radioactive materials, such as Special Nuclear Materials (SNM), and to respond to incidents involving Radiological Dispersal Devices (RDD). The PDS-GO-01 includes innovative algorithms in order to minimize false alarms. The PDS-GO-01G instrument is ideally suited for First Responders, Haz-Mat, Border/Inspection personnel and for Security of critical infrastructures. Figure 1 details the PDS-GO-01 physical features:

![PDS-GO-01 (front and rear view) and charging cradle](image)

**PDS-GO-01 (front and rear view) and charging cradle**

- 1- 7-Segment LED display
- 2- Gamma detector reference point
- 3- Push buttons
- 4- Battery compartment
- 5- Charging contacts
- 6- Battery compartment screws
- 7- USB jack location
- 8- LED light indicators

Operator interface is achieved by a 7-Segment LED display and three (2,3,4) push buttons. Radioactivity indication is performed by the display screen on the top of the device (1), an audible beeper, and a vibrator. The audio indication can be configured by the user. The battery compartment (5) is accessible by removing four cross head captive screws at the rear part of the instrument (6). Charger contacts (7) are located on the battery compartment at the bottom of the instrument above the rubber protective cap. Instrument setup and retrieval of stored data is performed via USB connection, while the instrument is mounted inside the charging cradle.

### RESULTS

The following table concentrates the results of the PDS-GO performance against the ITRAP+10 requirements, as tested in our laboratories:

<table>
<thead>
<tr>
<th>ITRAP+10 requirements</th>
<th>PDS-GO performance</th>
<th>Reference</th>
</tr>
</thead>
<tbody>
<tr>
<td>Effective energy range: 60 keV – 1.33 MeV</td>
<td>45 keV – 1.8 MeV</td>
<td>Fig. 1</td>
</tr>
<tr>
<td>FAR – 1 alarm per 10h</td>
<td>Complies</td>
<td></td>
</tr>
</tbody>
</table>
Gamma ambient dose rate indication within ±50%

<table>
<thead>
<tr>
<th>Temperature range</th>
<th>Dose rate within 30%</th>
</tr>
</thead>
<tbody>
<tr>
<td>-20°C - +50°C</td>
<td>Complies</td>
</tr>
</tbody>
</table>

Sensitivity

Complies

Figure 4

Description of results and analysis is to be presented with graphs, figures and tables included in the text, at the appropriate place.

Figure 1. Noise level: background.

Figure 2. Noise Level. Left: $^{241}$Am at 50°C. Right: $^{241}$Am at 24°C
DISCUSSION

A new PRD is presented. The device is based on the novel technology of silicon photomultiplier. The radiation detector has been tested successfully according to the requirements of the ITRAP 10+ program. The technology is investigated for evaluating its potential to upgrade the device for identification application as well.

REFERENCES

INTRODUCTION

Many existing MCA and multi channel scaling (MSC) systems are physically large or plug-in cards thus require both a personal computer and proper measure instruments. Their main disadvantage is large weight and difficult mobility. As so, our motivation was to develop a compact, lightweight & mobile, fully feature PC radioactive spectrometry system.

We have developed a portable spectrometry system which includes MCA, MSC, High Voltage Power Suppliers (HVPS), a laptop and all other components needed for spectroscopy measurement. One of the developed system's advantages is the ability to connect wide range of radiation detectors.

A gamma spectroscopy system usually consists of a radiation detector, proper electronics which collects and processes all the signals produced by the detector, and a computer with processing software, which generates the spectrum, displays and stores data for analysis.

The most important interaction mechanisms in radiation detectors are the photoelectric, Compton or pair production effects. Further explanation of these effects can be found in [1]. Scintillators [2] and PMT [3] are two components of radiation detectors [1]. Their response is temperature dependant. It means that temperature changes cause to spectrum shift and can lead to misidentification of the radioactive material. Since the workstation should operate in wide temperature ranges it is essential to stabilize the spectrum readout, hence identifying correctly the radioactive material. Due to the use of wide range commercial radiation detectors, the spectrum shift phenomena can be prevented by keeping the temperature difference, \( t \), between the MCA and the environment stable during the spectroscopy time.

EXPERIMENTAL SET-UP AND PROCEDURE

The developed workstation undergoes a calibration procedure at 25°C versus a well defined source.

In the present study the effects of temperature changes on the MCA, which is a critical component in identifying the radioactive material, are studied.

Eight K thermocouples were positioned inside the developed workstation. Their position and description can be found in Fig. 1 and table I respectively.
Table 1 – Thermocouple description

<table>
<thead>
<tr>
<th>Thermocouple no.</th>
<th>Location/Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>3 (18)</td>
<td>Center of -5kV power supplier</td>
</tr>
<tr>
<td>4 (4)</td>
<td>Upper left corner of the workstation</td>
</tr>
<tr>
<td>5 (5)</td>
<td>Center of -2kV power supplier</td>
</tr>
<tr>
<td>6 (6)</td>
<td>4 cm above the MCA</td>
</tr>
<tr>
<td>13 (13)</td>
<td>Center of #2 12V DC power supplier</td>
</tr>
<tr>
<td>15 (15)</td>
<td>Center of the workstation</td>
</tr>
<tr>
<td>16 (16)</td>
<td>Center of the MCA (Environment)</td>
</tr>
<tr>
<td>17 (17)</td>
<td>Center of #1 12V DC power supplier</td>
</tr>
</tbody>
</table>

* Note: The number in and the description in parenthesis are related to the forced convection experimental stage

Analyzing the results of natural convection heat transfer experiments (figure 2), which were carried out for two configurations, points that this method cannot keep ΔT stable. Table 2 indicates the 4 configurations which were tested under forced convection. All of these experiments were done inside an inner room without air conditioning, isolated from the environment effects (including neighbor rooms).
Figure 2 – (a) $\Delta T$ as a function of time during natural convection experiments (b) spectrum shift phenomena during natural convection experiments

Table 2 – Forced convection tested configurations

<table>
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<tr>
<th>Configuration</th>
<th>One Fan</th>
<th>Two Fans</th>
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<tr>
<td></td>
<td>Ventilation slots</td>
<td>Temporary internal partition</td>
</tr>
<tr>
<td>A</td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>B</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td>C</td>
<td></td>
<td>X</td>
</tr>
<tr>
<td>D</td>
<td></td>
<td>X</td>
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</table>

Figure 3 shows the results of configurations A-D. Analysis of the results indicates that forced convection can keep the temperature $T$ between the MCA and the environment steady, thus preventing the spectrum shift phenomena. It can be clearly seen that the steady state of the systems, means the time after which the MCA's temperature changes with $\pm 2^\circ C$ is between 30 minutes to 2 hours. Further analysis of the results yielded that at $50^\circ C$, the upper environmental temperature at which the system is designed to operate, the temperature of its internal components does not exceed the operating temperature limit as defined in their technical specification. From an engineering point of view, in terms of benefit versus the required design, manufacturing and the assembly fitness time of the internal partition, configuration C, see Fig. 4, has been chosen. This configuration has the option of adding two outlet fans can be added in order to improve the air outlet. The defined stabilization time of the system, based on the analysis of the experimental results, is 40 minutes.
Figure 3 – Temperature difference of tested configurations at forced convection experimental stage

Figure 4 – Top view of the chosen ventilation configuration

REFERENCES

High Efficiency Power Supply with an Analog Temperature Compensation for Silicon Photomultipliers

V. Berdichevsky¹, M. Ghelman¹, A. Osovitsky², T. Mazor¹, Y. Kadmon¹

¹Nuclear Research Center - Negev, P.O.Box 9001, Beer Sheva 84190, Israel
²Health Physics Instrumentation Department, Rotem Industries Ltd. Israel

Corresponding Author: vadim.berdi@gmail.com

INTRODUCTION

Silicon Photomultipliers (SiPMs) have some advantages over photomultiplier tubes such as low operational voltage, compactness, insensitivity to magnetic fields and low cost. However, SiPMs suffer from significant gain variations with temperature changes. Utilization of SiPMs as light detectors requires an effective gain stabilization method. The SiPM is based on two dimensional array of photocells. Each photocell is an avalanche photo-diode, operated in Geiger-mode with a reverse voltage higher than the breakdown voltage. The gain of each photocell in the SiPM can be expressed as the photocell charge divided by the charge of an electron. The gain is depended on SiPM's over-voltage, which is the difference between the operational voltage and the breakdown voltage (1).

As the temperature rises, the lattice vibrations in the SiPM become stronger. This increases the probability that carriers may strike the structure before reaching the energy needed for ionization to occur. Thus, the break-down voltage is growing linearly with the temperature (2), and the gain decreases when working at the same operational voltage. Increasing the operational voltage as a suitable linear function of the temperature maintains the over-voltage constant and thus the gain of the SiPM remains constant as well. However, this method is not taking into account the influence of the scintillator and the power supply whose characteristics are temperature-depended as well.

In order to overcome the total non-linearity of the detector's gain versus temperature, a new compensation method was developed and presented in this paper. A piecewise linear compensation function is used, dividing the whole temperature range into three segments. This analog compensation circuit is integrated into a highly efficient power supply, which consuming less than 0.2 mA from a 3 V voltage source.

POWER SUPPLY

The power supply was designed for nominal input voltage of 3 V, and is based on a commercial boost converter IC to supply a bias voltage in the range of 27 V to 29 V to the SiPM. Instead of directly using the commercial IC to convert the 3 V input voltage into the required output voltage, we have used three intermediate Cockcroft-Walton voltage-multiplication stages, as shown in Fig 1. Such configuration is much more efficient due to the lower output voltage of the boost converter and thus requires lower quiescent current.
TEMPERATURE COMPENSATION

Basic schematic of the linear temperature compensation circuit is shown in Fig 2. $V_t$ is the output voltage of a linear thermistor IC, and is equal to 800 mV at room temperature (20°C). Moreover, $V_t$ has a 10 mV/°C temperature dependence.

It is possible to set the bias voltage of the SiPM as well as the slope at which this voltage would change versus the temperature by changing the resistors values in Fig 2.

Fig 3 (dotted line) shows the $^{137}$Cs photo-peak at 662 keV dependence on temperature, assuming the bias voltage is changing with 18 mV/°C slope using an external power supply. It is seen from this figure that between 30°C and 50°C the peak is staying almost at the same channel on the MCA and is independent of temperature. Notwithstanding, between 0°C and 30°C there is a 10% change in the location of the peak. This is mainly due to the effect of the CsI(Tl) scintillator whose light emission is temperature depended as well.

The continuous line in Fig 3 shows the $^{137}$Cs photo-peak dependence on temperature while using the power supply in Fig 1 with the temperature compensation circuit from Fig 2, calibrated to 18 mV/°C as well. For this setup, at temperatures above 20°C the gain drops at 20% as the temperature rises to 50°C, due to the limited load regulation of the power supply.
Figure 3. $^{137}$Cs photo-peak channel versus temperature.

To improve the results it is possible to change the bias voltage in a piecewise linear function of temperature. The compensation circuit presented in Fig 4 divides the whole temperature range into three portions (below 20°C, between 20°C and 40°C, and above 40°C), each of the portions is compensated with a different slope.

Figure 4. Feedback configuration for a piecewise linear temperature compensation.
RESULTS
The results with the new piecewise linear compensation method are presented in Fig 5.

![Graph](image)

Figure 5. $^{137}$Cs photo-peak stability versus temperature using the new piecewise linear compensation method.

CONCLUSIONS
A high efficiency power supply with a piecewise linear temperature compensation function for SiPMs was developed.

The new analog compensation method takes into account not only the SiPM’s gain variations versus temperature, but also the effects of the scintillator and the power supply. Using the new method, total gain deviations of only ±4% over a wide range of temperatures were observed, compared to more than ±15% with a simple linear compensation function and the same power supply with a limited load regulation.

REFERENCES
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