Dear Friends,

We are gathering today to our traditional bi-annual meeting of the scientific community in Israel, which deals with nuclear energy and radiation. We decided this time to expand the meeting to three days instead of the usual two days meeting. It was lucky, as we received over 100 papers, which is a significant increase from the last conference. We hope that you will enjoy the venue, but most of all, the collegial atmosphere and the extra knowledge all of us will acquire.

The first day, as in the previous conferences, is devoted to more general lectures. The following two days are dedicated to lectures in specific professional fields.

We hope that all of us will enjoy the meeting,

Prof. Zeev Alfassi

On behalf of the Organizing Committee
Organizing Committee

Prof. Zeev Alfassi – Chairman, Ben Gurion University, Beer Sheva.

Dr. Uzi German - Nuclear Research Center Negev, Beer Sheva.

Dr. Marcelo Weinstein - Nuclear Research Center Negev, Beer Sheva.

Dr. Ilan Yaar - Nuclear Research Center Negev, Beer Sheva.

Mr. Gustavo Haquin - Soreq Nuclear Research Center, Yavne.
List of Exhibitors

**Eldan Electronic Instruments Co.**
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Tel: 03-9371144
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The 24th Conference of the Nuclear Societies in Israel

Program & Papers
14:00 - 17:30  PL1 - Plenary Session
Hall B

Chairs:  U. German
         Y. Ronen
         Y. Laichter

14:00  Welcome Addresses

Uzi German
On Behalf of the Organizing Committee
Ronen Yigal
President of the Israel Nuclear Society
Yaacov Laichter
President of the Israel Society for Radiation Protection

Invited Lectures

14:30  TBA
S. Horev

15:00  Nuclear Renaissance and Non-Proliferation
P. Goldschmidt

15:30  Towards a Nuclear Renaissance
J. Koch

16:00  Radiological Accident in Goiania – Mental and Psychological Effects
J.J. Rozental

16:30  Radiation Protection from a Radiobiological Perspective
Z. Hollander

17:00  Progress in Monte Carlo Simulation Methods for Radiation Applications
I. Orion
08:30 - 10:30 Parallel Sessions (S1-S3)

**08:30 - 10:30 S1 - Accelerators / "SARAF"-A**
Chair: A. Nagler

- **08:30** Overview of the SARAF Project
  A. Nagler

- **08:45** LiLiT - a Liquid Lithium Target as an Intense Neutron Source for Nuclear Astrophysics at SARAF
  G. Feinberg, M. Paul, D. Berkovits, I. Silverman

- **09:00** Towards Accelerator-based Boron Neutron Capture Therapy of Infectious Diseases
  S. Halfon, M. Paul, D. Steinberg, A. Nagler, I. Polacheck, A. Rubinstein, M. Srebnik

- **09:15** Light RIB Production with a 40 MeV Deuteron Beam
  T.Y. Hirsh, D. Berkovits, M. Hass

- **09:30** Neutron Diffraction in Archaeometallurgy: an Example of Possible Use in the SARAF
  E.N. Caspi, O. Rivin, H. Ettedgui, M. Peilstocker, S. Shilstein, S. Shalev

- **09:45** Beam Commissioning of Phase I of the SARAF Accelerator
  K. Dunkel, M. Pekeler, C. Piel, P. vom Stein, H. Vogel

Discussion

**08:30 - 10:30 S2 - Radiation Measurements**
Chair: G. Haquin

- **08:30** Locating, Identifying and Quantifying Radioactive Sources in an Inaccessible Area
  Y. Shamai, Y. Nir-El, G. Haquin, Z. Yungreiss

- **08:45** Gamma Measurement System for Calculation of Calcium Transport in Tomato Plants
  U. Wengrowicz, D. Tirosh, Y. Ifargan, S. Ovad, B. Sarusi, A. Bar-Tal, A. Schwartz, R. Seligmann

- **09:00** Calcium Translocation and Whole Plant Transpiration Noninvasive Measurements using Radio-Strontium as Tracer

- **09:15** Measurement of PET Tracers Low Activity Levels During Plasma Samples Analysis
  A. Osovizky, B. Laster, A. Wilson, T. Harris, B. Sarusi, U. Wengrowicz, D. Tirosh, S. Houle

- **09:30** Laser Ablation-ICPMS of Hair Strands as a Means for Assessment of Internal Exposure to Uranium
  E. Elish, H. Sela, Z. Karpas

- **09:45** Counting Delayed Neutrons in Uranium Analysis- Statistical Interpretation of Experimental Results
  T. Rieper, Z. Yungreiss, G. Haquin, Y. Nir-El
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<td>08:30 - 10:30 S2 - Radiation Measurements (Cont.)</td>
<td>Hall B</td>
<td>Neutron Flux Measurements of a D-T Neutron Generator</td>
<td>R. Ashkenazi, M. Cohen, D. Melnik, H. Ettedgui, A. Beck, Y. Ben-Dov, Y. Nir-El, G. Haquin</td>
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<td>10:00 - 10:30 S3 - Reactor Physics</td>
<td>Hall C</td>
<td>The Haling Power Depletion For Nuclear Core Analysis Including Genetic Algorithm Optimization Methods</td>
<td>S. Levine, K. Ivanov</td>
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<td>08:30</td>
<td></td>
<td>A Safe Solution to World Energy Supply - the Very High Temperature PB Reactor</td>
<td>D. Saphier</td>
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<td>08:45</td>
<td></td>
<td>Verification of Decay Heat Calculation Capabilities of BGCore System</td>
<td>E. Fridman, E. Shwageraus</td>
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<td>09:00</td>
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<td>Seed-Blanket Fuel Concept for Pebble Bed Reactors</td>
<td>E. Plosker, L. Droizman, E. Shwageraus</td>
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<td>09:30</td>
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<td>High Conversion Thorium Fuel Cycle for PWRs</td>
<td>D. Volaski, E. Fridman, E. Shwageraus</td>
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<td>09:45</td>
<td></td>
<td>Breeding of 242mAm in a PWR</td>
<td>L. Golyand, E. Shwageraus, Y. Ronen</td>
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<td>10:00</td>
<td></td>
<td>The Use of 241Am to Enhance Proliferation Resistance of LWR Fuel Cycle</td>
<td>L. Golyand, Y. Ronen, E. Shwageraus</td>
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<td>10:15</td>
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<td>Some Implications of Branching Ratio of the 241Am(n,gamma) Reaction</td>
<td>L. Golyand, E. Shwageraus, Y. Ronen</td>
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<td>10:30 - 11:00</td>
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<td>Coffee Break and Exhibition</td>
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<td>11:00 - 13:00</td>
<td>Parallel Sessions (S4-S6)</td>
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### 11:00 - 13:00 S4 - Calculation Methods (Cont.)  
**Hall A**

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<tr>
<td>11:30</td>
<td>Data Assimilation in the Atmospheric Dispersion Gaussian Plume model: Simulating the Gamma Absorbed Dose by using an Approached 1D Solution</td>
<td>D. Sattinger, A. Sharon, S. Levinson, B. Sarusi, I. Yaar</td>
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<tr>
<td>11:45</td>
<td>Radioactive Contamination Field Simulation for VMS</td>
<td>Y. Kadmon, A. Broide, M. Ghelman, D. Tirosh</td>
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<td>12:00</td>
<td>GammaGen - a Computer Code for Gamma-ray Spectra Generation</td>
<td>S. Levinson, B. Sarusi, O. Pelled, U. German</td>
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<td>12:15</td>
<td>The Benefit of Adding Detectors in the Back when Measuring Heterogeneous and Homogenous Contamination in the Lungs</td>
<td>O. Pelled, U. German, A. Abraham, S. Tsroya, Z.B. Alfassi</td>
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<tr>
<td>12:30</td>
<td>Dual Whole Body Counter: What Average to Use</td>
<td>J. Feldman, M. Segal, Y. Sabov, O. Presler, I. Orion, M. Weinstein, U. German, Z.B. Alfassi</td>
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### 11:00 - 12:45 S5 - Radiation Detectors - A  
**Hall B**

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<tr>
<td>11:15</td>
<td>An Improved Methodology for Local Alarm System</td>
<td>R. Freud, E. Marcus, E. Gonen, A. Levitin</td>
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<td>11:30</td>
<td>Evaluation of Radiation Monitoring System</td>
<td>R. Freud, E. Gonen, E. Marcus, I. Shvetz, R. Harari</td>
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<tr>
<td>11:45</td>
<td>Aerial Radiation Detection Vehicle Manned and Unmanned Concepts</td>
<td>I. Halevy, I. Yaar, A. Broide, A. Manor</td>
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<td>12:00</td>
<td>A Novel Radiation Monitoring System</td>
<td>E. Marcus, E. Gonen, A. Algom, B. Sarusi, A. Schwartz, R. Freud, U. Wengrowicz, D. Tirosh</td>
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<td>12:30</td>
<td>Environmental Radiation Monitoring System Control Center</td>
<td>B. Sarusi, E. Vax, S. Levinson, U. Wengrowicz, D. Tirosh</td>
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11:00 - 13:00  S6 - Medical Physics - A  Hall C
Chairs:  S. Faerman
D. Silvern

11:00  New Scoring System and a Novel Method of Plan Quality Evaluation and Comparison
J. Menhel

11:15  A Comparison Study of 10MV Unflattened and Flattened (therapeutic) Beams
M. Ben Ayun, S. Faermann, A. Tsechanski

11:30  A Genetic Optimizer for Intra-operative Mid-course Correction of a Prostate Implant
D. Silvern, M. Zaider, G. Cohen, J. Xiong, M. Zelefsky

11:45  Target Positioning Accuracy of a Novel Technology for Target Localization in Radiation Therapy
D. Schifter, D. Neustadtter

12:00  Large Feld IMRT for Malignant Plural Mesothelioma: Comparison with Conventional Technique

12:15  Irradiation Accidents in Radiotherapy: Analyze, Manage, Prevent
A. Wygoda

12:30  Can Record and Verify Systems Eliminate Radiotherapy Delivery Errors? The Rambam Medical Center Experience
R. Bar-Deroma, O. Keren, A. Kuten

12:45  Implementing a New Chart in the Radiotherapy Unit at Rambam Medical Center
R. Bar-Deroma, O. Keren, R. Carmi, E. Gez, R. Sela, A. Kuten

13:00 - 14:00  Lunch

14:00 - 15:30  Parallel Sessions (S7-S9)

14:00 - 15:30  S7 - Accelerators / "SARAF"-B  Hall A
Chair:  D. Bercovits

14:00  Operation and Maintenance of SARAF
L. Gertz, I. Mardor, Y. Buzaglo, A. Grin, R. Michael, R. Raizman, L. Weissman, A. Nagler

14:15  First Experience at SARAF Intense Low-energy Beams : Radiological Concern
L. Weissman, D. Berkovits, Y. Grof, Y. Ben-Dov

14:30  Lattice Study for SARAF 40 MeV Linac and Extended for the EURISOL 60 MeV Low Energy Driver
J. Rodnizki, B. Bazak, D. Berkovits, G. Feinberg, A. Shor, Y. Yanay

The 24th Conference of the Nuclear Societies in Israel
14:00 - 15:30 S7 - Accelerators / "SARAF"-B (Cont.)  Hall A

14:45  Ion Beam Loss Calculations for SARAF Accelerator with Emphasis on the Tails of the Particles Distribution  
B. Bazak, A. Shor, D. Berkovits, G. Feinberg, J. Rodnizki, Y. Yanay

15:00  Simulations of Aperture Collimation and Ion Neutralization Effects in the SARAF Low Energy Beam Transport  
Y. Yanay, B. Bazak, D. Berkovits, G. Feinberg, J. Rodnizki, A. Shor, L. Weissman

15:15  Beam Dynamics Simulation of the 1.5 MeV Proton Beam Measured at the SARAF RFQ Exit  

14:00 - 15:45 S8 - Applied Physics  Hall B  
Chair:  E. Yahel

14:00  Cross Section Tuning  
J. Wagschal, T.L.Y. Yeivin

14:15  High Pressure Study of Intermetallic Compound Hf(10)B2 and TDPAC Studies of Radiation Damage  
I. Halevy, A. Beck, E. Auster, O. Levy, H. Ettedgui, E.N. Caspi, O. Rivin, I. Yaar,  
M. Ganor, S. Kahane, Z. Berant, J. Hu

14:30  Absence of Low-Temperature Dependence of the Decay of 7Be and 198Au in Metallic Hosts  
V. Kumar, M. Hass, Y. Nir-El, G. Haquin, Z. Yungreiss

14:45  Radiation Damage Accumulation in Thin Metal Films: A Novel Alpha Emitter Technique  
R. Hevroni, E. Yael, A. Maniv, O. Levy, G. Makov, I. Kelson

15:00  Crystal and Exchange Fields in the Singlet Ground State System TbCo3B2 Studied by Inelastic Neutron Scattering  
O. Rivin, R. Osborn, A. Kolesnikov, E.N. Caspi, H. Shaked

15:15  Magnetic and Crystallographic ND Study of Tb_1-xY_xCo_3B_2  
E.J. Wolfson, E.N. Caspi, H. Ettedgui, H. Shaked

15:30  Possible Liquid-Liquid Phase Transition in Bismuth  

14:00 - 15:45 S9 - Medical Physics - B  Hall C  
Chair:  M. Levita

14:00  Measurements of Residual Radiation Spectra from Neutron Activation in a Medical Linear Accelerator  
D. Schifter, T. Schory

14:15  Automated Calculation of TG-43 Parameters for a Brachytherapy Source  
D. Silvern, A. Ayoub, G. Shani
14:30 A Micro-MOSFET In Vivo Dosimetry System  
L. Cahana, D. Epstein, S. Koren, I. Orion, D. Alezra

14:45 The Use of a Mini-chamber for Small Photons Beam Dosimetry  
Y. Krutman, Y. Tzigelman

15:00 Farmer Ion Chamber Dosimeter Response in Strong Magnetic Fields  
S. Koren, D. Alezra, I. Orion

15:15 Lateral Spine X-Ray Diagnostic Examination: Left or Right Projection?  
A. Ben-Shlomo

15:30 A Three Year Undergraduate Program (B.SC) in Medical Radiation Physics in the Ariel University Center of Samaria  
T. Schlesinger

15:30 - 16:00 Coffee Break and Exhibition

16:00 - 17:30 Parallel Sessions (S10-S12)

16:00 - 17:15 S10 - Non-ionizing Radiation  
Chair: M. Israeli

16:00 AC Powered Linear Power Supply Units ("AC Adaptors") - Magnetic Fields and Unexpected Electric Current Dependence  
R. Hareuveny, I. Ben David

16:15 Magnetic Fields in Vicinity of Stand Alone Transformer Stations  
R. Hareuveny, N.M. Yitzhak, S. Kandel

16:30 Laser Hazard Assessment with an Eye Simulator  
M. Margaliot, A. Amitzi

16:45 Interference of Extremely Low Frequency (50 Hz) Electric and Magnetic Field Emitted by HV Power Lines with RF Broadband Field Meters  
I. Eliyahu, Y. Yaffe, R. Hareuveny

17:00 Acoustic Waves Generation by Pulsed Microwave Radiation  
N.M. Yitzhak, R. Hareuveny, R. Ruppin, M. Margaliot

16:00 - 17:15 S11 - Radiological Aspects  
Chair: Y. Shamai

16:00 Source Term Estimation for a Radiological Dispersive Device Scenario  
I. Yaar, A. Shron

16:15 Health Risks from Exposure to Low Doses of Ionizing Radiation according to BEIR VII  
J. Koch, T. Schlesinger
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<td>16:30</td>
<td>Human Cytoplasmic and Nuclear Non-Exchangeable Organic Bound Tritium (OBT) Increase Due to Exposure to Different Extracellular Media and Potassium Concentrations</td>
<td>R. Gonen, U. German, E. Priel, Z.B. Alfassi</td>
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<td>16:45</td>
<td>High Dose Radiation Facility at Rotem Industries Site</td>
<td>R. Sarusi, Y. Bashiri, N. Ankri, H. Asido</td>
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<td>16:00 - 17:30</td>
<td>S12 - Medical Physics - C</td>
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<td>Medical Physics Society - General Meeting</td>
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<td>08:30</td>
<td>PL2 - Reactor Technology Aspects</td>
<td>Boiling in MTR Fuel Element Channels as Cooling Mechanism During Partial Loss-of-Coolant Accident (LOCA) in Pool Type Reactors</td>
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<td>Core Overheating Event Probability Analysis due Lack of Passive Cooling in a Typical MTR Pool Type Research Reactor by FTA Study</td>
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<td>09:00</td>
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<td>Bottom-Reflooding of PWR Core with Transient Two-Phase Boundaries and Rod Ignition</td>
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<td>09:15</td>
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<td>Validation of BGCore System for Burnup Calculations</td>
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<td>09:30</td>
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<td>Thermal-Hydraulic Feedback Module for BGCore System</td>
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<td>09:45</td>
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<td>Criterion for Energetic Fuel-Coolant Interaction</td>
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<td>11:00</td>
<td>PL3 - Dosimetry</td>
<td>Investigation of the Ionization Density Dependence of the Glow Curve Characteristics of LiF:Mg,Ti</td>
<td>Y. Horowitz, A. Horowitz, L. Oster, S. Marino</td>
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<td>11:15</td>
<td></td>
<td>Applying the High-Temperature TL in LiF:Mg,Ti to Mixed Thermal Neutron-Gamma Dosimetry – a Review</td>
<td>U. German, M. Weinstein, A. Abraham, Z.B. Alfassi</td>
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<td>11:30</td>
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<td>Reassessing TLD100 doses by the Residual dose – a Review</td>
<td>A. Abraham, M. Weinstein, U. German, Z.B. Alfassi, R. Azulay</td>
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<td>The PTTL Effect in Different TL Materials</td>
<td>B. Ben-Shahar, U. German, M. Weinstein, A. Abraham</td>
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<td>11:00</td>
<td>PL3 - Dosimetry (Cont.)</td>
<td>Fast Neutron Dosimetry Using the Glow Peak Separation Method in CaF2:Tm</td>
<td>N. Vainblat, B. Ben-Shahar, M. Weinstein, U. German, A. Abraham</td>
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<td>12:30</td>
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<td>The Distribution of Committed Effective Dose over Fifty Years after Acute Intake of Uranium</td>
<td>T. Kravchik, R. Rabi, N. Dukhan, Y. Gabay, U. German</td>
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<td>13:00</td>
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<td>14:00</td>
<td>PL4 - Radiation Detectors</td>
<td>Adaptation of the Scinti SPEC Multi-Channel Analyzer for Spectrometry of CZT Detectors</td>
<td>A. Edery, I. Orion</td>
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<td>14:15</td>
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<td>Continuous Tracking of RFID Tagged Radioactive Sources</td>
<td>A. Broide, E. Marcus, Y. Gabay, E. Miron, R. Seif, U. Wengrowicz, Y. Kadmon, D. Tiros</td>
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<td>14:45</td>
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<td>Cyclotron Target Monitoring During Bombardment for PET Isotope Production</td>
<td>A. Osovizky, B. Laster, E. Mishani, A. Wilson, A. Garcia, E. Vulaski, B. Ashkenazi, S. Houle</td>
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<tr>
<td>15:30</td>
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16:00  Radionuclides in Air Particulates at Soreq Nuclear Research Centre, Yavne
     G. Haquin, T. Riemer, R. Ashkenazi, C. Tzur

16:15  Natural Radioactivity in the Groundwater in the Negev Desert and Arava Valley - Israel
     G. Haquin, N. Pery, L. Broshi, Z. Yungreiss, A. Paytan, Z. Karpas, O. Paz-Tal,
     S. Elhanani, A. Vengosh

16:30  Sustainable use of Natural Water Sources Containing Ra Activity

16:45  Radiological Considerations in the Production of Lightweight Concrete Based on Coal Ashes
     M. Nisnevich, G. Sirotin, T. Schlesinger, Y. Eshel

17:00  Determination of Natural Radioactivity and Gamma Radiation dose in Ceramic Materials Containing Fly Ash Developed in Israel
     N. Lavi, G. Haquin, Z.B. Alfassi
Towards a Nuclear Renaissance

Jean Koch

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

INTRODUCTION

With the beginning of the Nuclear Age in the 1950s it was believed that nuclear power could produce electricity in such large amounts and at such a low cost ("too cheap to meter") that it would become the main energy source. Nuclear power underwent consequently a rapid development during the 1960s and 1970s. This development was brought to a stop in the U.S. following the 1979 Three Mile Island accident and in several Western European countries following the 1986 Chernobyl accident. Nevertheless, nuclear power has been expanding in other regions of the world, mainly in Eastern Europe and in Asia. Globally, the nuclear share of electricity production has remained constant at about 16% since the mid 1980s. However, this share varies widely among the 31 countries that produced nuclear electricity in 2006, ranging from 2% in China to 78% in France, as can be seen in the following figure.

Nuclear share of electricity generation – 2006 (source: World Nuclear Association, 2007\(^1\))
It seems that the main factor fueling the renewed interest in nuclear energy in the West is the growing awareness that climate change is very likely caused by combustion of fossil fuels to produce energy. Several other factors concur in recent years to promote a revival of nuclear energy.

**CLIMATE CHANGE**

Awareness of the public and decision-makers about climate change and its anthropogenic causes has much increased in recent years. This trend has lately been exemplified by the awarding of the 2007 Nobel Peace Prize to the IPCC (Intergovernmental Panel on Climate Change), the international body of experts established by the UNEP and WMO to assess the scientific, technical and socio-economic research conducted in the field of climate change. IPCC concludes in its just published authoritative Fourth Assessment Report that "most of the observed increase in global average temperatures since the mid-20th century is *very likely* due to the observed increase in anthropogenic greenhouse gas concentrations"\(^{(2)}\) (very likely indicating an over 90% probability of occurrence).

If mankind is to seriously address global warming while increasing its energy consumption, energy efficiency and conservation must be promoted and "carbon-free" energy sources must replace combustion of fossil fuels to a large extent. Renewable energies (wind, solar, hydroelectricity) are to be aggressively advanced. However, since wind energy and solar energy are not expected to make a major contribution in the foreseeable future, it becomes inevitable to recognize that nuclear power must make a significant part of the energy mix.

It seemed that the flexible mechanisms designed in 2001 to facilitate the implementation of the Kyoto Protocol, by lowering the overall costs of reducing the emissions of greenhouse gases, were biased against nuclear power\(^{(3)}\). However, IPCC states in its Fourth Assessment Report that "nuclear power, which accounted for 16% of the electricity supply in 2005, can have a 18% share of the total electricity supply in 2030"\(^{(4)}\). Although the predicted share increase is relatively modest, it must be stressed that it would mean more than a doubling of the nuclear installed capacity. Such a dramatic increase would surely bring about a nuclear renaissance.

On that background, some prominent environmental scientists and environmentalists concerned by climate change are expressing their support for nuclear energy, such as James Lovelock (the atmospheric scientist and father of the Gaia theory) and the late Bishop Hugh Montefiore (founder and director of Friends of the Earth) in the UK and Patrick Moore (a co-founder of Greenpeace) in the US\(^{(5,6)}\). Consequently, it seems that an evolution in the way of considering nuclear power is underway in the "green" establishment.
OTHER FACTORS DRIVING THE NUCLEAR REVIVAL

The World Nuclear Association mentions the following factors, besides climate change, as being instrumental in reviving nuclear energy:

- Increasing energy demand: A doubling of electricity consumption is predicted by 2030 to take into account global population growth, industrial development and living standard rise. Energy-intensive desalination plants will also be needed to cope with the increasing shortage of drinking water.

- Economics: Due to increasing fossil fuel prices, nuclear energy is the most cost-effective of the available baseload technologies. The economic benefits of nuclear power will increase further, as carbon emission reductions are encouraged through various forms of financial incentives. Nuclear power is also less sensitive to fuel price fluctuations, since its fuel costs are a much smaller part of the production costs than for energy from fossil fuels.

- Security of supply: As many countries realize how vulnerable they are to interrupted deliveries of oil and gas, the abundance of uranium makes nuclear energy attractive.

PROSPECTS FOR THE FUTURE

442 nuclear power plants are in operation, as of December 2006, with a total installed capacity of about 370 GWe. New capacity includes:

- 30 reactors being built around the world today.
- 35 reactors planned to come online during the next ten years.
- over 200 planned for a further horizon.

The situation and the outlook in several key countries is described below:

- China has nine reactors in operation, two under construction and an ambitious building program for 28-40 new ones by 2020, bringing to a capacity of 41-46 GWe by 2020.
- India has seven reactors under construction, with plans for 16 more to give 20 GWe of nuclear installed capacity by 2020.
- Russia plans to build 40 GWe of new nuclear power by 2025 and to double its nuclear capacity.
- Finland and France began both to build a 1600 MWe EPR (European Pressurized water Reactor), renewing building of nuclear units in the EU after more than a decade.
- Germany and Sweden are reconsidering their anti-nuclear policies.
- In the USA, the first application for a nuclear power plant in more than 30 years was filed in September 2007 and applications for as many as 29 reactors are expected by the NRC over the next three years. It is also to be mentioned that an impressive improvement in the operational performance of the existing U.S. fleet of 104 reactors occurred between 1980 and 2005, when the average capacity factor increased from less than 60% to more than 90%.
CONCLUSIONS
It seems that the nuclear renaissance is really underway, but for the ambitious plans to materialize public acceptance must be fully recovered. This can only be achieved if the public is confident that the following three main issues have been adequately treated:

- Safety: Existing Generation-II reactors have an impressive record of safety and Generation-III plants that are presently being built incorporate inherently safe features in their design. This is therefore a risk communication issue.
- Nuclear waste: As opposed to the public common view that this is an intractable issue, practical steps are taken to build deep geological disposal facilities for nuclear long-lived waste in several countries, Finland being in the forefront. Risk communication is therefore the main issue.
- Proliferation: The diversion of nuclear fuel to produce nuclear weapons is the most difficult issue to address. However, Generation-IV systems that are presently being designed are proliferation-resistant.

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Radiological Accident in Goiania – Mental and Psychological Effects

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Key words: public and professional perceptions, conflicts of information, distortion and misunderstandings, anxiety, fear, mental health and discrimination.

Adjusting ability (technical competence), capacity (necessary resources) and sensibility (consciousness) are the three most important priorities any staff member of any organization should bear in mind while acting during a nuclear or radiological emergency.

I have no doubt to affirm in the above context -- although two decades have gone last September, many lessons from the radiological accident in Goiania have not been fully learned. The events which unraveled in the fall of 1987 took the population of the city of Goiania completely by surprise: they did not comprehend what had happened and failed to grasp what measures needed to be taken, a situation which precipitated, besides the technical response in a very complex psychological and mental reaction, coupled with discrimination, which is my intention to discuss and to alert authorities.

In September 1987, the removal of the rotating assembly of the shielding head of a Teletherapy unit and the dismantling of the capsule containing 50.9 TBq (1375 Ci) of Cs-137 led to the most serious radiological accident to have occurred to date. It resulted in the injury by radiation of many people, four of them fatally, and in a widespread contamination of the central area of Goiania, a Brazilian City of some one million inhabitants. Goiania is the Capital of the State of Goias, 180 km from Brasilia, the capital of Brazil. The radiological accident in Goiania, with many parameters involved, was unique, because it happened in an urban setting -- center of the city and until now is the more important Laboratory to learn how to deal with in case of similar situation. The accident presents lessons that cover the phase prior to the accident, the emergency phase and the aftermath of the accident up to the present, twenty years later. Each phase was marked by conflicts by individuals and organizations alike.

Today, aside the medical treatment to the patients in observation, the repository where 3,500 cubic meters of waste are stored in more than 6,000 containers, still evokes social and economical concern and mental and psychological emotion, among the victims. The Goiania accident involved all classes of the regional society. Not only were we faced with political and technical problems to deal with, but also social and economic and, simultaneously, the parallel parameter involving psychological and mental health, aspects, affected from apprehension, by society commotion, lack of credibility, perception of risk, misunderstanding of nuclear jargons, economic losses, neighborhood atmosphere, discrimination (by others due to fear, including relatives).

This translated into fear and depression of the population and stigmatization and discrimination against the victims and the main commercial products of the city. This paper also considers the public and professional perceptions and conflicts of information, distortion and misunderstandings, as the main causes for the anxiety and discrimination. Additionally, several unforeseeable psychological issues during the early phase of the accident are mentioned which are not found in the literature. These emerged suddenly through unexpected situations or questions that we never expected and must be analyzed in terms of (a) the identification of problems in safety culture and (b) safety culture and human behavior.
Finally this paper also emphasizes the security point of view, today a matter of IAEA’s high concern, as well as national authorities, taking into account recent IAEA statements and conferences on security of sources, especially in case of illicit traffic and malevolent event, and psychological and mental aspects derived. On these aspects I would like to point out 2 topics and discuss:

1 - “The radioactive materials needed to build a "dirty bomb" can be found in almost any country in the world, and more than 100 countries may have inadequate control and monitoring programs necessary to prevent or even detect the theft of these materials, the International Atomic Energy Agency (IAEA) says.” – “It is difficult to predict the level of exposure of persons, as this would depend on many factors such as the physical and chemical form of the radioactive material, size and type of explosive and proximity of persons to the blast. In all likelihood, the most severe tangible impacts of a dirty bomb would be the social disruption associated with the evacuation, the subsequent clean-up of contaminated property and the associated economic costs.”

2 – “We know from experience with accidental releases of radiological sources that they can cause widespread panic, economic hardship and significant health concerns. Remember Brazil, in 1987. Urban scavengers found a medical teletherapy machine left in an abandoned building. They removed the radioactive source from its shielding, ruptured it and distributed the beautiful blue, glowing powder they found inside to their friends, neighbours and relatives. The powder was 137Cs. Four people died. More than 110 000 people were monitored for radiation exposure at the city’s sports stadium. Scores of buildings were evacuated and some were even demolished. Cleanup costs were enormous. The incident generated about 3500 cubic metres of radioactive waste. Actual fatalities were relatively light in the Brazil incident, but panic was widespread. I can only imagine how much worse the situation would have been had terrorists dispersed the toxic material rather than innocent, uninformed people.


"JOSE JULIO ROZENTAL"  

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INTRODUCTION
This series of three papers (parts A, B and C) is meant primarily for non-radiobiologists professionals working in the area of radiation protection. It aims at pointing out the less than sound scientific foundations of a human safety related profession and the need for a better scientific theory in this area. However, this by itself might not be enough to apply the relevant and much necessary changes and updates to methodologies for radiation risk assessment crucial for the radiation protection profession. Public responsibility and administrative courage to address the emerging new information is not less imperative.

The most important toolbox of every profession that manages safety and purports to set standards is the one holding concepts and methodologies to enable risk definitions and assessments. These, as also for every engineering profession, have to rely on as sound as possible scientific theories. In the ionizing radiation protection case it is crucial to have a basic theory that explains the radiation – living matter interaction as well as a resulting, more practical, set of rules that directs how to calculate the build up of risk to humans as result of the Ionizing Radiation (IR) absorbed dose.

For more than half a century, the worldwide Radiation Protection profession (RP) has been leaning on such set of rules. However, for the last two decades fast and consistence accumulations of empirical data in molecular biology in general and as a result also in radiobiology have been casting an increasing shadow of doubt on the governing, most basic theory relevant to RP. The doubts have been concentrated mainly around the apparent variability and complexity in the rules governing interactions of the living matter with different levels of radiation doses mainly and most importantly the "low doses" (less than 100mGy)\(^1\). It seems that the simple theoretical monotone description of same interactions at all levels of radiation with the living matter is an over simplification that can find only diminishing support from the fast accumulating scientific knowledge.

**Simple mechanistic views and the resulting hypothesis**
In 1926, Muller was the first to publish that X-rays were the cause for mutations in living cells\(^2\). Up to the early 1950s, radiation biologists assumed that there was a threshold levels to almost all health detriments associated with radiation exposures. Moreover, it was believed, for several decades after the discovery of radiation, that small doses were beneficial to health\(^3\). In 1957 Lewis\(^4\) suggested that cancer risks might exist at all doses greater than zero.

Currently, the prevailing consensus held by national and international radiological protection organizations is that for all absorbed doses of IR, including the low and very low, the most appropriate risk model is one in which the risk of radiation induced cancer and hereditary diseases is assumed to increase linearly with increasing dose, with no threshold: "Linear Non Threshold" (LNT)\(^7\).
This is a working hypothesis that was decided upon administratively in the late 1950's (5). This concept stems from the "classic" "target shooting" hypothesis and its fundamental practical conclusions. The basis is the more than half-a-century old "microdosimetric argument" (6). Briefly, this argument holds as follows: a low dose D1 is considered to be low enough so that only one photon is assumed to pass through the cell nucleus. Two assumptions are then made: Firstly, that this D1 dose causes an increase of cancer risk in human population and secondly, that the carcinogenic process is monoclonal (i.e. depends on the transformed cell only). If so, than the same basic cellular responses to radiation damage must exist at say a dose of D1/10 because the proportionally (linear) fewer cells damaged by D1/10 would each be subject to the same single photon damage as the ten times larger effected cell population. It follows that the excess cancer risk must decrease monotonely by the same factor of 10 over the dose range D1 – D1/10. This is the argument underlying the LNT hypothesis and resulting risk extrapolations (6).

The above approach implicitly assumes that any DNA lesion has the same probability of giving rise to cancer irrespective of the number of lesions in the same cell and neighboring cells. Also, as the phenotype of tumors appear to be monoclonal (i.e. originate from one ancestral cell), carcinogenesis has been assumed to be also monoclonal in its nature and its risk proportional linearly to the dose at all of its levels (13) loyal to the "grand" LNT paradigm.

The main body of epidemiological evidence that is serving the current risk model

The data acquired from the epidemiological studies performed on the survivors of the Hiroshima and Nagasaki nuclear bomb attacks currently still serve as the main foundation for the stochastic-risk estimates of Ionizing Radiation (IR) at all dose, dose-rate and radiation energy levels (8, 9, 10, 11). The risks at low doses and/or dose rates are simply linearly extrapolated form the Hiroshima and Nagasaki case (with a one instance factor correction for low Dose and Dose Rates Effect - DDREF). These people were exposed to "very acute" (extremely high dose rate) doses of 2-5 MeV \( \gamma \) radiation with dose-weighted LET spectra similar to those of \( \gamma \) rays at energies around 1 MeV (12).

CONCLUSION

The current perception is that the amount of IR stochastic risk depends linearly, with no threshold on the amount of damage to the DNA. \( \text{RISK} = \alpha \times \text{DAMAGE} \), with \( \alpha \) being a real' positive and constant expression.
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INTRODUCTION

As was shown in part A of this series of three papers, the current working model used by Radiation Protection (RP) professionals is based on a concept that can be conceptually represented by the expression: \( \text{RISK} = \alpha \cdot \text{DAMAGE} \), with RISK being the stochastic risk from Ionizing Radiation (IR), DAMAGE being DNA damage and \( \alpha \) being a real positive and constant expression.

"Confusing" RESULTS

There is a constant flow of significant evidences to a reality of IR – living matter interaction that seems increasingly complicated. Moreover, many results "stubbornly" obtained for over a century, do not fit the main governing hypotheses (1).

Here are most of the "confusing" observations:

1. LET Dependence on IR energy.

It has been recognized in the literature, for quite a long time, that the Relative Biological Effectiveness (RBE) of photons (2) and electrons (3, 4) is not constant at all energy levels on which it depends inversely (5). Accordingly, it has been also shown that the Linear Energy Transfer (LET) of the so called "low-LET" quality of radiation varies with energy to which it is inversely related (6, 7). Moreover, it was shown that these phenomena are not restricted to radiobiological endpoints such as chromosomal aberrations or mutagenicity but also to neoplastic cell transformation and carcinogenicity (9). Currently, for all the so called "low LET radiation" the radiation quality weighting factor (\( W_R \)) used to estimate the effective absorbed radiation dose (in other words, to estimate the risk) is instructed to be \( W_R=1 \) (10). This is true also for 29 kVp X ray radiation that was shown to bear very high RBE quality relative to references such as 200 kVp x-rays or Co-60 (11, 12).

2. IR dose rate importance for risk estimation, direct and inverse rate effects

As reviewed (13) the existence has been known of a directly proportional Dose Rate Effect (DRE) according to which the probability of effects on the genetic material, such as chromosomal aberrations, mutagenicity or neoplastic transformation, increases with the increase of dose rate (14). This has been observed in somatic as well as in germ cells. More recently (15), it was reported that it was possible to show, in both these cell types, also the existence of a general pattern of an inverse DRE. This effect is characterized by greater sensitivity to IR-caused genetic end points such as HPRT mutagenicity at very low Dose Rates (DR) than at low DR. This inverse dependence on rate was approximated by using a parabolic model with a minimum at the 10mGy/min range with sensitivity rising with higher and lower DRs (for low-LET radiation). This DR was named Minimal Mutability Dose Rate (MMDR).

This rate is millions of times higher than the average, world wide background. It seems that exogenous and endogenous damages are sensed and reported in at least partially different cellular signal transduction pathways and are weighted differently as to the final impact on DNA Double Strand Breaks (DSBs) and other radiobiological endpoints production.
Moreover, a comparison with other (than IR) physical, chemical and biological agents indicates that IR is indeed set apart from these by its unique micro- and nano- dosimetric traits (16).

In view of the information presented in the next section, it was suggested (15) that the background to noise ratio is the IR signal measured and that may be mechanisms such as stochastic resonance effect might be of pertinence. The importance of dose rate to the neoplastic transformation of cells was clearly shown (17) with the DDREF approaching infinity at very low dose rates (still orders of magnitude higher that the background IR dose rate) (18).

3. DNA damage by endogenous sources of free radicals

The oxidative phospohilation system in the eukaryotic cell mitochondrion is a source of massive out-fluxes of free radicals into the cytoplasm. DSBs are the lesion type that is widely considered to have the major role, among other possible DNA lesions, in cell transformation. It was calculated (16) that the number of spontaneously generated endogenous DSBs (EDSBs) is equivalent to that caused by no less than 200 mGy/day of low-LET IR (some 8 DSBs/day). Moreover, the total endogenous DNA lesions (all kinds of single strand lesions) production was measured (15) to be equivalent to that caused by about no less than 5mSv/min.

4. Radio-Adaptive Response (AR)

The term was coined in 1984 (19). The existence of an AR is now well established. It is classically described as the induction, by a low dose of radiation (few tens of mGy), of a protective effect against a high challenge dose (hundreds of mGy and higher) (20). This effect has been observed for many endpoints including chromosomal aberration induction, mutation induction, neoplastic transformation, cell survival, fetal death, fetal abnormalities and tumor latency (21). AR effect has been also observed in humans (22).

5. Bystander Effect (BE)

For the purpose of this paper, the radiation BE is defined as previously (23): BE is the ability of cells affected by a harming agent to convey manifestations of damage to other cells not directly targeted by the agent or necessarily susceptible to it themselves (24). BE, so far, was investigated mainly in-vitro. It includes a whole spectrum of radiobiological endpoints some detrimental, others not. Both quality and intensity of the BE depend on the cell types involved in the experimental set up. The various effects observed under the above definition of BE are the result of signal generation by irradiated cell interacting with non irradiated cell and not the results of radiation induced changes in the culture medium (25). For convenience purposes, a review published not long ago (24) had divided the BE into four not necessarily mutually exclusive sub-categories that will be mentioned briefly:

1. BE after cytoplasmic (non-nucleus) irradiation, 2. BE after low fluence of α-particles, 3. BE after irradiation with charged particle microbeam and 4. BE effect after transfer of medium from irradiated cells.

6. Genomic instability (GI)

It is pertinent to understand that the genome of a living organism is constantly "bombarded" by a current of potentially hazardous molecules which are species of free radicals (26). The origin of such stress can be endogenous (mainly oxidative phosphorylation) or exogenous (e.g. background IR). Moreover, the "normal" cell activities such as DNA replication and cell division are serious destabilizing processes. Indeed, thousands of damages are caused to a cell DNA daily by endogenous as well as exogenous sources. Among them DSBs mentioned above. It is clear, therefore, that much elaborated genome stabilizing mechanisms are in place in order to keep genome stability. The actual resulting reality is a balance between harm and defense (27).
IR was shown to be able to cause the genome to lose some of its stability. This is expressed by enhanced rate of mutagenesis and resulting deleterious effects and a variety of cellular changes and, also, the increased sensitivity to genotoxic agents (28). This GI phenotype can persist through many cell divisions in culture and also through murine generations with their progeny exhibiting the results. GI is considered hallmark of the process of carcinogenesis, most of human cancers showing loss of genomic stability (28).

7. Radiation Hyper Sensitivity (RHS)
This phenomenon was demonstrated initially some 14 years ago when mammalian cell line was discovered to be hypersensitive to radiation doses below 250 mGy (29). As the radiation dose is increased to 1 Gy, the cell population becomes increasingly resistant per unite dose. Since then, by 2004, RHS had been reported in cells of more than 40 X-irradiated human cell lines and for different radiation qualities and biological end points (29). The elimination of damaged cells is considered a hallmark of defense against cancer at the cellular level. It is widely considered a very powerful defense strategy against the development of transformed phenotypes out of the damaged genotypes (30). This is true only at low dose levels beyond which the elimination of large number of cells would have become detrimental by itself. At the higher absorbed doses the various repair mechanisms with their auxiliary cell behavior alterations are in place to take care of the genome in order to prevent cell transformation in the face of risk of missed repair.

8. Change in defense strategy according to level of absorbed IR dose.
It was shown, through genetic response analysis (32, 33, 34), that even very low IR doses (2-4 mGy) do provoke cellular (genetic) response as do higher doses. However, both levels of responses did not activate the same sets of genes and also did so in a very different time frame. This shows that different cellular response to IR systems are at work in cases of different levels of absorbed dose. The response, on the genetic level was sometimes of repression and sometimes of activation of the relevant genes.

9. Cancer risk of populations living prolonged time in high levels of background IR.
For space limitations reasons, it will be only mentioned that the risks observed in such numerous cases, clearly do not support the expected risks from an LNT type risk model (39, 44, 45, 46, 47, 48).

10. Observed risks deducted from acute exposure accidents hardly support LNT.
Not only in cases of chronic, low rate exposures, but also in radiological accidents, involving exposure to relatively high dose rates, considerable amounts of observed risks that were retrospectively calculated were quite far from the ones expected from simple extrapolations to low doses suggested by LNT. Examples are results from the Chernobyl (1986), Mexico (1962), China (1991), Brazil (1987) and more (39).

11. Crucial role of the cell environment in the complex malignancy process.
It is now widely accepted that although cancer, in most cases, appears as a monoclonal phenomenon (i.e. the cancerous cells are the progeny of a single cell) the process leading to the outbreak of the disease is far from being dependent on the transformed cell alone. There is a crucial dependence, of carcinogenesis, on the controlling immediate cellular neighborhood (41). Additionally, it depends also on the response of more remote surveillance such as that of secreted cytokines, local aggressive protecting cells and the systemic immunological machinery (39).
12. Tissue versus whole body exposures

Recognizing the importance of absorbed dose rate and of the biological environment on carcinogenesis, it became clear (30) that a calculated dose for the whole body in case of local actual exposure (i.e. "effective dose") can hardly be the same, in terms of cancer risk, as cases when the exposure in reality is of a more dispersed nature. This is so also because of immun-surveillance considerations.

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Radiation Protection from a Radiobiological Perspective

Safety-Related Technological Profession in Need of a Better Scientific Foundation.

Part C – Fresh Concepts are Called for

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INTRODUCTION

As was shown in part A and part B of this series of three papers, the current working model used by Radiation Protection (RP) professionals is based on a concept that can be conceptually represented by the expression: RISK = α * DAMAGE, with RISK being the stochastic risk from Ionizing Radiation (IR), DAMAGE being DNA damage and α being a real positive and constant expression (part A). A list of well documented empirical phenomenology, mainly from the last two decades, was also brought (part B), that show that the natural reality has been, most likely much more complex to be described in a risk model such as the Linear Non Threshold (LNT) one.

The discussion in this part will draw on this phenomenology in order to call for new and fresh design efforts of new risk estimation concepts that will describe the natural reality more accurately. It will reflect the notion that acquires more and more supporters as follows: The LNT guideline for risk estimates in RP is based on a too simplified model. It is based on a damage theory that does not take in consideration the biological inter and intra cellular complex damage response network and the complex carcinogenesis.

DISCUSSION

The above described compelling body of evidence has created unrest among scientists that are concerned with carcinogenicity in general and radiological carcinogenicity in particular. The net risk resulting from low to very low absorbed doses of IR are particularly debated with concentration on the question how to estimate cancer risk at these doses. Controversy over this subject is readily detected in the contemporary scientific literature. The discussion here has not undertaken to support one particular stance or to describe the variety of opinions in the field or theories on the right model to describe low dose IR – living matter interactions. The status of the literature can be learned from the comprehensive referencing.

The "target shooting" assumption ("microdosimetric argument" mentioned in the introduction) that is basic to the LNT model, takes in consideration the magnitude of the damage to the DNA only as function of the radiation flounce. By doing so it completely disregards the other, most pertinent part of the cancer risk-generation equation: the biological response of the insulted cell.

The damage response of the living cell to toxic stresses such as IR is mediated through sensors, signal transaction mechanisms and effectors (such as gene transcription factors). This response system is principally protein based. In general, protein action efficiency, as function of external conditions, shows parabolic-like type of behavior with a maximum (i.e. optimum conditions).
They also show saturation in their activity as function of the concentration of their activity substrate. Thus, it should be expected, from a damage response system, to have function characteristics of a protein-based biochemical system. It is, therefore, reasonable to expect that the damage response characteristics will resemble those of proteins (e.g. saturation). As a result, it seems plausible that the cell will respond optimally to damages to its DNA at a specific range of damage burden. Below and in particular, above this range, reduction in the ability of damage overcoming is expected. It is reasonable to expect that as the dose or dose rate increase, the responses per unite of dose, become less effective.

Even if the response to damage is neglected, linearity of the amount of damage as function of the IR dose can hardly be assumed at all doses. This is because some of the damaging molecules, resulting from the radiation (i.e. free radicals of sorts) might be intercepted by mechanisms (proteins) the efficiency of which might not be the same per unit of dose, at all doses or dose rates.

The existence, mentioned above, of different genetic responses (and hence different mechanisms) to different dose levels, strongly imply that linearity (as function of dose) of the resulting cell defenses across this dose range can be expected only as a rare coincidence.

Even pivotal scientists consistent in their support for the current adherence to the LNT working model for RP practice, include notable reservations such as: "Consequently, we do not know the magnitude, nor even the direction of any such deviation from linearity-the risks should indeed be lower than those predicted by a linear extrapolation, but they could well be higher." Or, "The aim should be to begin to incorporate all the new types of information generated into the risk assessment process for ionizing radiations. This approach will help supplement the available human epidemiological data in setting radiation protection standards" Or, "At present, we cannot be sure of the appropriate dose response relation to use for risk estimation at very low doses- (respectively).

The main organizations that publish reports which have influence on the standards set for RP, have supported, in their reports published in the last few years, the adherence to the LNT risk model. Their reports are, however, more guarded than earlier ones. BEIR VII–phase 2 report that was first publicized in 2005 added the remark concerning the above recommendation: "uncertainties on this judgment are recognized and noted" (pages 50, 466 and 591).

Fundamental practices in RP should be reexamined. For example, in view of the different RBE values for photons as function of their energy, can we still assume Wr=1 for all so called "low LET" qualities of radiations (i.e. γ, X, β) when we purport to calculate "Equivalent dose"?

Or, taking in consideration the supra-cellular immuno-surveillance defenses of the organism against the tumorgenesis of a transformed cell, can we still use the tissue weighting factor (WT) as before? Regarding the immun system reaction, can we freely translate radiation dose received in one concentrated area into a more moderate whole body exposure (effective dose)?

The, by now, obvious fact that the very low doses have effects on the cell does not mean that these effects are detrimental. They could as well be protective. It might well be that "low doses stimulate these systems while high doses inhibit them".
Radiation protection, as a safety related profession, is currently relying on a practical perception that has little scientific support as such. As explained above even those who recommend adherence to the LNT approach, do not believe that it represents the situation in nature \(^{11,8}\). This means that RP is relying on human risk "rulers" and methods of calculations (e.g. equivalent and effective doses) that are believed, by most if not all relevant radiobiological scientific community, to not represent reality, the same reality by which humans "really" become seek or die due to IR exposures.

RP is about risk and not DNA damage.

\[
\text{RISK} = \text{DAMAGE burden} + \text{total bio-RESPONSE}. 
\]

A contradiction exists between the approaches of the pragmatic and conservative radiation protection professionals and the ones of the scientific community. The first group needs clear cut models to conduct the safety management by. Its members need to be able to calculate the stochastic cancer risk emanating from any given exposure situation. Also, they have to be able to counter possible law suits regarding theirs or the supervised employer's alleged safety related misconduct. The second group however is interested in better understanding of nature, refuting hypothesis if possible and embark on newly proved methodologies. The problem is that the legitimacy of the practices of the first group should better rely on the sound findings of the second one.

A solution to the above paradox can hardly be a one that tends to exaggerate in the risk estimate for "just to be on the safe side" reasons (this might be the case in using LNT and ALARA principles when in doubt of the reality). There are claims that should be seriously considered, that such approaches, over the years, have been the cause for the public "radiophobia". Also, it might effect negatively the psychological outcomes of radiological terrorism (e.g."radiological dispersion device") and has been the reason for unwanted medical practices such as abortion panics, too much limited utilization of diagnostic modalities, unnecessary costs and more \(^{6,9,10}\).

It may well be that, in the future, when many non linear models, describing the complex biological responses to IR insult, will be in the scientific consensus, the pragmatic RP practices, as recommended by expert bodies, will still favor LNT approach as the best tool for dealing with the overall summation of the various mechanisms leading to low-dose risk.

At this point it is not possible to rush to conclusions based on the current knowledge and agreement within the scientific community. It is expected that the advising expert bodies such as ICRP, BEIR and NCRP will change their verdicts decisively only when they can replace the current risk estimation set of rules with a new one. This new theory of risk will have to enjoy enough support and agreement from the scientific community to be able to become the new basis for a new practical working risk model for the use of the radiation protection profession so as to be able to withstand legal, ethical and social challenges.
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LiLiT - a Liquid Lithium Target as an Intense Neutron Source for Nuclear Astrophysics at SARAF

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**INTRODUCTION**

The mechanism for production of the majority of heavy elements ($A > 60$) in stars involves neutron-capture processes. Abundances of heavy elements are determined by the stellar rates of the slow (s-)process and rapid (r-)process and the half-life of relevant nuclides. The neutron capture cross section on stable and unstable nuclides in the stellar regimes of energy is consequently an essential parameter in nuclear astrophysics. An intense neutron source based on the $^7$Li$(p,n)^7$Be reaction is planned as a target in SARAF\textsuperscript{(1)} for this purpose. The special kinematics of the reaction near threshold $E_{p,\text{lab}} = 1.881$ MeV happens to generate in a thick lithium target, forward-collimated neutrons which display, when integrating over the whole cone solid angle, an energy spectrum appropriate to simulate the conditions of a stellar interior at a temperature of 350 MK. An intense beam is required in order to generate a high neutron flux which enables measuring new cross sections that are impossible to measure using a solid lithium target. The use of a liquid target enables dissipating the power while irradiating the lithium with an intense proton beam. A conceptual design\textsuperscript{(2)} (figure 1) was proposed for the SARAF Liquid Lithium Target (LiLiT). It includes a lithium jet chamber, an electro-magnetic induction pump loop and a heat exchanger. The detailed design is currently in progress.

**NUCLEAR ASTROPHYSICS**

The goal of the nuclear astrophysical experiment is to make a reliable measurement of the Maxwellian-averaged capture cross section (MACS) $\langle \sigma v \rangle / v_T$ for a stellar temperature of 350 MK ($kT = 30$ keV). Neutrons produced in stellar interior are quickly ($\sim 10^{-11}$ sec) thermalized\textsuperscript{(3)} through elastic scattering. The neutron velocity distribution can be described by the Maxwell-Boltzmann distribution:

![Conceptual design of the SARAF Liquid Lithium Target](image)
where $dn$ is the number of neutrons with velocity between $v$ and $v+dv$ (or between $E$ and $E+dE$), $n_0$ is the number of neutrons per unit volume, $T$ is the temperature of the stellar interior and $m$ is the neutron mass. The MACS in the stellar medium is defined by the expression:

$$
<\sigma v> / v_T = \int n(E)\sigma(E)vdE / v_T.
$$

In order to reproduce the average cross section of a Maxwellian neutron flux in the stellar medium by irradiating a thin target in the laboratory, the required neutron spectrum has an $E^{-\exp(-E/kT)}$ dependency. Calculations were performed for the neutron spectrum produced by the $^7\text{Li}(p,n)^7\text{Be}$ reaction, assuming that the neutrons are emitted isotropically in the center-of-mass frame when the protons energy is above the reaction threshold (1881 keV). If the velocity of the center of mass is greater than the neutron velocity in the center-of-mass frame, the neutrons are emitted in a forward cone. When the proton energy is 1912 keV, neutrons are emitted in a cone of $\pm 60^0$ angular opening and the calculated neutron energies, using the $^7\text{Li}(p,n)^7\text{Be}$ energy dependent cross section$^4$ and integration over all incident energies down to threshold in a thick Li target and over all neutron angles, results in a spectrum very close to the $E^{-\exp(-E/kT)}$ dependency with an effective value $kT = 26$ keV, a cutoff at 107 keV and a neutron yield of $2.4 \times 10^7$ neutrons/mA (figure 2). This calculation is in agreement with measurements that were done using a solid lithium target$^5$ and other calculations$^6$. Cross section measurements at $kT = 26$ keV spectrum can be extrapolated to the required stellar $kT = 30$ keV spectrum.

The secondary experimental target used for activation by the neutron flux will be positioned just downstream of the Li jet on the external side of the jet-support wall. Interaction of the neutrons with the lithium flow and the stainless steel support wall affects to some extent the neutron beam yield and distribution through absorption and scattering. The attenuation may be taken into account while calculating the MACS value in the secondary target but the effect of resonant elastic scattering (mainly the $^{26}\text{Fe}$ 28 keV resonance) is harder to predict. The back wall thickness will be determined after transport calculations and analysis of the expected effects on the activation yield. Our preliminary calculations indicate that a rigid stainless steel back wall will contribute only a small uncertainty to the MACS measurement (few percents for ~1 mm thick wall).
The high neutron flux expected from the SARAF proton beam and the liquid-Li target (estimated to be more than one order of magnitude larger than presently available) is especially valuable for nuclear astrophysics because it would enable laboratory measurements on radioactive nuclides. Such nuclides act often as branching points in the s-process flow between a further neutron capture and radioactive β decay. An interesting case which we plan to study is that of the $^{90}$Sr($n,γ$)$^{91}$Sr reaction. The practical limit in the $^{90}$Sr target mass and the $^{91}$Sr half-life of 9.5 hours makes a high neutron flux of neutrons essential for this study. Off-line γ-spectrometry of the $^{91}$Sr product is planned to be used after activation. There is no data for $^{90}$Sr MACS despite the importance of that isotope for the s-process and understanding nucleosynthesis in this region of the element production chain.

**EXPERIMENTAL SETUP**

The LiLiT target is planned as an intense neutron source for SARAF phase I. At this stage the linear accelerator will include 6 cavities optimized for $β=0.09$ which should provide the required beam for the astrophysical experiment: a high intensity proton beam (2-4 mA) with a low energy spread.

Assuming a smooth lithium surface and a mono-energetic (1912 keV) proton beam, neutrons are created in the first 4 μm, after which the protons' energy becomes smaller than the reaction threshold. A minimum thickness of 160 μm of lithium is required to stop the protons but for safety and hydrodynamical reasons, the lithium jet thickness is designed to be 1800 μm. Leaving safety margins for irradiating by a ~8 mm width (rectangular) proton beam, the jet width is planned to be 15 mm.

In order to dissipate the heat created by the high power proton beam (3.8 kW for 2 mA proton beam) and reduce the risks of boiling or increased evaporation, a jet velocity of 20 m/s is required. The jet velocity is generated using an induction liquid metal pump.

![Figure 3. Depth profile of temperature expected from a 4 kW proton beam (2 MeV, 2 mA) in a Li flow at 20 m/s.](image)

![Figure 4. Water jet simulation, front view (left) and side view (right), proton beam direction indicated by red arrow.](image)
Figure 3 shows the expected depth profile of the temperature for a 4 kW (2 MeV, 2 mA) proton beam and assuming a steady-state temperature of 200°C in the lithium loop. The proton beam radial distribution is assumed to be Gaussian with $\sigma = 2$ mm. The maximum temperature increase – about 5 mm downstream the proton beam center, is approximately 80°C.

Water simulations have been performed to study the hydrodynamic behavior of the jet which is affected by the nozzle and shape of the back wall. Hydrodynamic similarity was conserved by keeping the Reynolds and Weber numbers (60,000 and 2,020 respectively) equal to those of the liquid Li flow. Ratios of 1:6 and 7.3:1 for the hydraulic diameter and velocity of the liquid-Li and water flows, respectively, were derived. Based on a preliminary design for the nozzle and back wall, a Perspex model was built in atmospheric environment.

Reasonable thickness changes and waviness were observed visually (figure 4). Since the kinematic viscosity of water and lithium are similar, a 1:1 scale (for both hydraulic diameter and velocity) water setup is been planned and built presently. Some diagnostics devices will be mounted in the new setup for measurements of quantitative jet parameters.

In order to reduce the risks for both personnel and machine, the LiLiT target will be first tested using an electron gun in a separate laboratory prior to operating in the SARAF beam hall. After accumulating operating experience and proving that the liquid metal loop sustains the electron beam power, the target will be moved to SARAF and irradiated with the proton beam. Thermal, mechanical and vacuum behavior will be tested at proton energies below neutron threshold and neutron flux will be determined in a second stage.

Protecting the superconducting cavities from lithium vapors or splashes is critical. In order to prevent a line of sight, two bending magnets will create an S bend in the SARAF phase I beam line. The beam will be transported and transversally focused by 5 quadrupoles – 2 doublets and a single quadrupole and will pass through an adjustable rectangular collimator (4-jaw) close to the target which will enable controlling the beam geometry (figure 5). Preliminary beam dynamics simulations for the PSM and beam transport to the liquid-Li station were performed using the GPT simulation code(7). The RMS radius of the proton beam at target position is 3.1 mm as required and the energy fluctuation (1 STD) is 7 keV.
Energy is mainly gained using cavities 3 and 4, cavities 1 and 6 operate as bunchers while cavities 2 and 5 are turned off for a 1.912 MeV final energy and low energy spread. The beam line design is not finalized but, based on the simulation (figure 6) the required beam parameters can be achieved with reasonably low energy proton losses.

Figure 6. Beam RMS radius, 5000 protons envelope and bore radius along PSM and beam line. The beam has a waist at target position (~15 m) with $R_{rms} = 3.1$ mm, divergence of the beam after target position is artificial. Up-left: energy gain along PSM.

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Towards Accelerator-based Boron Neutron Capture Therapy of Infectious Diseases

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INTRODUCTION

A new conceptual design for a Boron Neutron Capture Therapy (BNCT) facility, based on the high-current low-energy SARAF\(^{1(1)}\) proton beam and liquid lithium target, is described. A novel approach utilizing BNCT for the treatment of infectious diseases is under investigation.

The principle of using the \(^{10}\text{B}(n,\alpha)^{7}\text{Li}\) reaction induced by thermal neutrons on \(^{10}\text{B}\) (a stable isotope of boron with natural abundance of 20%) was originally suggested by Locher\(^{(2)}\) for cancer therapy. The reaction has a cross section of 3840 barns (whereas the competing reactions of neutrons with \(^{10}\text{B}\) have a cross section of less than 1 barn) and yields highly-ionizing particles (\(\alpha\) and \(^{7}\text{Li}\)) and a \(\gamma\)-ray:

\[
{^{10}\text{B} + n_{th} (\approx 0.025 \text{ eV})} \rightarrow {^{11}\text{B}^*} \rightarrow {^{4}\text{He}_{1.78 \text{ MeV}} + ^{7}\text{Li}_{1.1 \text{ MeV}}} \rightarrow {^{4}\text{He}_{1.47 \text{ MeV}} + ^{7}\text{Li}_{0.84 \text{ MeV}} + \gamma_{0.48 \text{ MeV}}}
\]

The emitted charged particles have a short range in organic tissue (5-9 \(\mu\text{m}\)) which is about the diameter of one cell. Thus, the destructive effect of these high linear-transfer energy (LET) particles is confined only to the cells that contain \(^{10}\text{B}\). In order to utilize these remarkable qualities for successful therapy, a sufficient amount of \(^{10}\text{B}\) must be selectively delivered to the tumor cells or to other targeted areas in the body, and enough thermal neutrons must be absorbed to sustain a lethal effect from the nuclear reaction. At an achievable \(^{10}\text{B}\) concentration of \(\sim 20 \mu\text{g/g in a target cell (or 10}^9\text{ atoms/cell)}\)\(^{(3)}\), the neutron flux that is necessary for reasonable therapy duration (30-90 min) is around \(10^9\) \(\text{s}^{-1} \text{ cm}^{-2}\) on the tissue\(^{(4)}\).

The optimization of the neutron energy spectrum for the maximum benefit to the patient in therapy of deep-seated tumors has been studied in the last fifteen years\(^{(5,7)}\). The epithermal neutron energy range, \(\sim 1\) eV to \(\sim 10\) keV, has been established as ideal for therapy of such tumors.

Accelerator-based BNCT

Suitable neutron sources for BNCT have been limited for many years to nuclear reactors\(^{(8)}\). A reactor can produce a sufficient neutron flux for therapy. However, the energy spectrum is usually moderated to the thermal range, which is efficient only for superficial tumors.

A neutron source based on a low-energy, high-current light-ion accelerator has the potential for meeting the requirements for a clinical BNCT facility. The flexibility to choose the target material and the ion energy and current allows the design of the most suitable neutron field (energy spectrum and flux) for therapy.
Further advantages of accelerator-based neutron sources for BNCT are: (i) Accelerators of small size which can, in principle, be located inside hospitals; (ii) Accelerators can be easily turned off when the neutron field is no longer required (hence licensing and regulations are simplified); (iii) The capital investment of an accelerator-based BNCT system is much lower than the installation of a nuclear reactor; (iv) Accelerators have good public acceptability, and are already being used in hospitals.

The main design effort of accelerator targets for BNCT has been on lithium, with the reaction $^7\text{Li}(p,n)^7\text{Be}$ at proton energy of 1.9-2.5 MeV. A major advantage of this reaction is the low energy neutron spectrum that is naturally produced (mean neutron energy in the range of 34-326 keV). Neutrons of these energies require less moderation in order to get to the BNCT ideal neutron spectrum (1 eV to 10 keV) than those generated in other target materials suggested for this purpose, such as beryllium\(^9\) and carbon\(^10\). Smaller moderator lengths translate to fewer neutron losses and lower the neutron per unit accelerator current at the patient position. The lithium target efficiency will allow the use of smaller and cost-effective accelerators with a current of ~5 mA\(^11\).

Despite the excellent neutronic qualities of the reaction $^7\text{Li}(p,n)^7\text{Be}$, a lithium target has been considered as very complicated to build because of the mechanical, chemical and thermal properties of lithium, the major problem being to sustain or dissipate the power generated by the high-intensity proton beam.

**Liquid-Lithium Target**

A Liquid-Lithium Target (LiLiT), for high-intensity proton beam (1.5-4 MeV, 2 mA) from SARAF Phase I\(^1\), is currently being built at Soreq NRC. The target, based on a forced liquid-Li flow is designed both to produce neutrons through the $^7\text{Li}(p,n)^7\text{Be}$ reaction and to serve as power dump for the proton beam. The design is based on a prototype liquid-lithium loop that has been developed at Argonne National Laboratory and has been successfully tested for power dissipation, using a 1 MeV, 20 kW electron beam\(^12\), and on the intensive research for IFMIF (a 250 mA, 40 MeV deuteron accelerator\(^13\)).

The construction of a liquid-lithium target combined with the high-intensity beam from the SARAF accelerator, presents a remarkable opportunity to create the world's first accelerator-based epithermal neutron source for BNCT. It will be the first lithium target able to absorb such a high power proton beam (up to 20 kW).

For an incident proton energy $E_p \approx 1.91$ MeV (just above threshold of the $^7\text{Li}(p,n)$ reaction at $E_p \approx 1.88$ MeV), the emitted neutrons are forward-collimated into a cone with a full-opening angle of $\pm 60^\circ$, average energy of 42 keV and total flux of $2.4 \times 10^{10}$ n/mA; at higher energies ($E_p \approx 2$ MeV) neutron are also emitted at rear angles, but most of the neutrons are still emitted at angles smaller than $\pm 51^\circ$, the average energy is 75 keV and the total flux is $1.1 \times 10^{11}$ n/mA\(^14\). A moderator/reflector assembly will be built to create a neutron spectrum and intensity maximally effective to the treatment. Thus the assembly is designed to: (i) Moderate the fastest neutrons to below 10 keV; (ii) Filter out the neutrons with energy lower than 1 eV; (iii) Reduce $\gamma$-ray contamination. The design must maintain a number of usable neutrons (1 eV-10 keV) sufficient to provide therapeutic doses in reasonable times (30-90 min).

Water as a moderator and Al$_2$O$_3$ as a reflector have been shown previously to be ideal for $^7\text{Li}(p,n)$ near threshold BNCT\(^15\). A conceptual model of our liquid-lithium target and moderator-reflector assembly is shown on figure 1. The moderator geometry is based on ref. 11, where Monte Carlo calculations showed that near threshold, a 5-mA proton beam can produce therapeutically useful neutron beams with a water moderator. Graphite, LiF, MgF$_2$, CaF$_2$, CsF$_6$ and Teflon are also considered as alternative moderators due to the extreme reactivity of liquid lithium with water and the associated safety requirements.
The photon production in the target include mostly 478-keV $\gamma$ rays from inelastic proton scattering ($p,p'\gamma$). The addition of a layer of lead or bismuth between the target and the moderator is considered to attenuate the $\gamma$-ray intensity at the patient station. As a thermal neutron shield, a $^6$Li thin sheet (0.01-0.03 cm) between the moderator and the phantom would be most effective\(^{(1)}\). Cadmium is not viable for this purpose, despite its high thermal neutron absorption cross section, since it produces large number of thermal-neutron capture $\gamma$-rays.

Figure 1. Conceptual model of our liquid-lithium target (LiLiT) with moderator/ reflector assembly for BNCT.

**BNCT of infectious diseases**

Accelerator-based BNCT can in principle be applied in a wide variety of clinical fields. We are presently investigating a novel approach for the treatment of infectious diseases stemming from biofilms.

Biofilms are composed of microorganisms in a hydrated polymeric matrix of their own synthesis, attached to a biological or non-biological surface such as medical implants or devices. The mode of growth of this microorganism, composed mainly of bacteria and fungi, is not completely understood. Bacteria in biofilms have however different characteristics than in their regular mode of growth. For example, they can be up to 1500 times more resistant to antibiotics and immune chemicals. Biofilms are associated with many infectious diseases and inflammatory processes such as ear, nose and throat (ENT) infections, implant infections, bacterial endocarditis, bacterial kidney stones, candida vaginitis, cystic fibrosis, chronic lung infections, acute pulmonary infection, osteomyelitis, nosocomial sepsis, dental, rheumatoid arthritis associated synovectomy, inflammatory bowel disease (Crohn's disease and colitis), fungal keratitis associated with contact lenses, upper respiratory tract infections, chronic wound infections to name a growing list of maladies.
The BNCT methodology described here can be beneficial in this context. We envision that BNCT will significantly degrade biofilms on medical devices, in cavities and other organs and prosthetic implants and will provide a non-invasive procedure that is less threatening to the patient, cost effective and at the same time will help to reduce resistance to common antibiotics.

**Preliminary experiments**

Feasibility experiments evaluating the effectiveness of boron neutron capture for killing bacteria are taking place at the Soreq NRC nuclear reactor. Previous BNCT efforts for cancer therapy indicate that efficient therapy requires a fluence of $10^{12} \text{n/cm}^2$ on targeted cells and a $^{10}\text{B}$ concentration of $\sim 20 \mu\text{g/g}$ at target cells (or $\sim 10^9$ atoms/cell)\(^{(3)}\). In these conditions, 2 or 3 neutron capture events are occurring very close or within the target cell.

Studies on the radiobiology of $\alpha$-particles\(^{(16)}\) also showed that the number of $\alpha$'s (with energies similar to that produced by $^{10}\text{B}$ neutron capture), required for one lethal lesion per cell, is in the range of 2-6 (depending on the cell type and their nucleus size). Following these results, our experimental conditions of thermal neutron flux (applicable since no tissue penetration is required), irradiation time and boron concentration were set to ensure at least one $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction to occur at the bacterium periphery. The neutron flux was measured at several locations in the reactor experimental hall, using the gold activation method. The reactor tangential tube was found to be the most suitable for the experiment, with a neutron flux $>10^7 \text{n/cm}^2/\text{s}$, a cadmium ratio of 16-18 and relatively low gamma radiation ($<100 \text{R/h}$).

Since the neutron flux is relatively low, the irradiation time was set to 6-8 hours and the $^{10}\text{B}$ concentration was 4.5 $\mu\text{g/g}$. We evaluated that in these conditions there would be around one $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction at a 5 $\mu\text{m}$ periphery of each bacterium. Control samples (no neutron irradiation) were prepared in the same way. Small polyethylene ampoules containing the bacteria *Streptococcus mutans* in Brain Heart Infusion (BHI) media and boron compound (FOX6\(^{(17)}\)) at different concentrations were placed in the tangential tube and exposed to the neutron flux from the Soreq reactor.

**RESULTS AND CONCLUSIONS**

Preliminary results of these experiments are shown in figure 2. The optical density (O.D.) of the solution containing bacteria in aqueous phase is a measure of the amount of bacteria present, when the background sample (BHI) is set as O.D. zero. In figure 2 the average O.D. of samples that were neutron irradiated for 6.25 hours and then incubated at 37 °C for 20 hours is shown together with the O.D. from identical control samples, which were only incubated at the same time. The O.D.'s were measured before and after the incubation.

A minor effect on bacteria growth in the highest $^{10}\text{B}$ concentration was measured in the control group (although the standard deviation is large). The irradiated group had a significant reduction in the O.D. at $^{10}\text{B}$ concentration of 4.5 $\mu\text{g/g}$.

The growth of bacteria was affected by the combined irradiation and $^{10}\text{B}$, but there was no total elimination of the bacteria. It is possible that the damage is not a direct killing of the bacteria, but a damage that is influencing the bacteria growth.
Figure 2. Average optical density (O.D.) as a function of $^{10}$B concentration for a 6.25 hr. neutron irradiation of samples (see text) and for control samples, before and after 20 hr. incubation.

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Light RIB Production with a 40 MeV Deuteron Beam

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INTRODUCTION

Light radioisotope beams such as \textsuperscript{6}He and \textsuperscript{8}Li have uses in basic nuclear physics, astrophysics studies and neutrino physics. Efficient large scale \textsuperscript{6}He production is one of the important milestones for the future "beta beam" facility \cite{1}. The main idea of the "beta-beam" is to produce large amounts of short lived radioisotopes such as \textsuperscript{6}He or \textsuperscript{8}Li that decay through beta decay. The radioisotopes are accelerated to relativistic energies and stored in a hippodrome shaped ring. During the beta decay an electron type neutrino is emitted, and due to the accelerator ring shape and the radioisotopes high velocity, there is a high efficiency of the neutrino to be emitted in one of two directions. Therefore, by accelerating very large amounts of \textsuperscript{6}He, for example, one obtains a directed neutrino source. By pointing the accelerator in the direction of a large neutrino detector some hundred km away, one can measure neutrino properties, such as neutrino oscillations or neutrino cross sections. The beta beam requires production of \(8 \times 10^{11}\) \textsuperscript{6}He atoms per second after extraction.

This work presents optimization calculations for the production of \textsuperscript{6}He and \textsuperscript{8}Li isotopes, through the \textsuperscript{9}Be(n,\alpha)\textsuperscript{6}He and \textsuperscript{11}B(n,\alpha)\textsuperscript{8}Li reactions, using a 40 MeV deuteron beam such as in the future SARAF accelerator at Soreq NRC \cite{2}. The production of \textsuperscript{6}He was found to have very high efficiency by using a two-stage irradiation setup \cite{3}, a primary target generates fast neutrons using a deuteron beam, while the secondary target uses the fast neutrons to produce large amounts of \textsuperscript{6}He. The calculations were performed using experimental data of the neutron spectrum emitted from the primary target \cite{4} and MCNP4b \cite{5}.
THE DOUBLE TARGET SETUP

The proposed setup comprises a primary thick target made of a light isotope such as beryllium, lithium or carbon and a secondary beryllium target, as shown in Figure 1.

![Figure 1 - illustration of the two stage irradiation setup](image)

A 40 MeV, 2 mA beam of deuterons collides with the primary light isotope target and produce fast neutrons (with average energy of ~12 MeV) mainly via the reactions (d,n) and (d,2n). These fast neutrons are then hit the secondary beryllium target, multiply by the reaction $^9\text{Be}(n,2n)^8\text{Be}$ and produce a very high yield of $^6\text{He}$ isotopes by the reaction $^9\text{Be}(n,\alpha)^6\text{He}$, on the order of $10^{13}$ atoms of $^6\text{He}$ per second. The cross sections of $^9\text{Be}(n,\alpha)^6\text{He}$ and $^9\text{Be}(n,2n)^8\text{Be}$ [6], compared to the neutron flux inside the secondary beryllium target, are shown in Figure 2.

![Figure 2 - cross sections of neutrons in $^9\text{Be}$ vs. n flux inside the secondary target [6]](image)

Clearly the overlap of the reactions cross section with the neutron flux is high and thus would give a high yield of $^6\text{He}$. A primary target made of lithium produces neutron yield of ~0.06 n/d for 40 MeV deuterons which corresponds to $8\cdot10^{14}$ neutrons for a 2 mA deuterons current.

MCNP SIMULATIONS

In order to investigate the production of $^6\text{He}$ using the two targets setup and to find the optimum geometry of the targets, we performed Monte-Carlo simulations using MCNP4b [5]. Because this MCNP version does not include charged particles, we took a measured neutron source [4] which stands for the neutrons emitted from the first target as result of the 40 MeV deuterons bombardment. The secondary target is then placed near the neutron source. Because of the forward angular distribution of the neutrons (80% of the neutrons are within an angle of $50^\circ$) the secondary target geometry was chosen to be cylindrical. Additionally, in order to increase the neutron flux inside the secondary target we placed a beryllium reflector around the secondary target which wraps it in a cup-like shape.
Several parameters were considered during this work. We investigated each one separately before choosing the optimum set. The parameters that affect the $^6$He production and that were examined are: the cylindrical target radius and length, distance between the targets, targets density, reflector thickness and the effect of the neutron source diameter.

**PRODUCTION RESULTS**

In order to find the optimal geometry for the secondary target, which best fits the neutron source properties, we calculated with MCNP the mean neutron flux in a 2D slice perpendicular to the target front.

The result in Figure 3 shows that the neutron flux inside the target has the highest value at the middle of the target front, just in front of the neutron source. According to that, it seems that the optimal geometry of the target should be a cone or half a sphere and not cylindrical. However, after some more calculations it was found that the maximum increase in $^6$He production yield for a cone-like target versus a cylindrical target is only few percents. Therefore, we focused mainly on optimizing the parameters for a cylindrical target.

It is clear that the parameter that mostly increases the $^6$He production is the distance between the primary and secondary target. Figure 4 (solid line) shows that one can get about twice more $^6$He by decreasing the distance between the targets from 5 to 1 cm.

![Figure 3 - Mean neutron flux inside a 2D slice within the beryllium secondary target, units are n/s/cm²](image)

![Figure 4 - $^6$He and $^6$Li production yields for different materials vs. L - the target positioning](image)
For the simulation shown in figures 3 and 4 we started with a 1 cm$^2$ neutron source at the back plate of the primary target. We find that increasing the neutron source size to 10 cm$^2$ decreases the radioisotope production by only 3%. For a primary lithium target, it is probably impossible to create a neutron source much smaller than 10 cm$^2$ on the primary target back plate, due to the long deuteron range of 20 mm. However, for a smaller secondary target, it is preferable to have also a small beam radius, at least smaller than the targets cross sectional area.

**EXTRACTION EXPERIMENTS**

After the $^6$He ($^8$Li) is produced one has to extract the $^6$He out of the secondary target. Due to the short half life of $^6$He, the extraction must be a fast process. We suggest a preliminary test of extracting the $^6$He from the target by heating the beryllium target to high temperatures and sweeping the diffusive $^6$He out of the target by flowing helium gas through it [7].

For efficient diffusion we need to reach high temperatures and to have a porous material. For these reasons we consider replacing the secondary beryllium target with a beryllium oxide compound. While pure beryllium melts at 1287°C, beryllium oxide ceramics melt at much higher temperatures of 2530°C. Beryllium oxide can also be produced in form of fibers [8], which should enable efficient extraction due to its porous structure. In order to examine the extraction efficiency we plan to perform an experiment using a BeO target and a fast neutron source. To test the production and extraction of $^8$Li radioisotope a porous boron nitride target is planned.

Results of simulations performed using BeO and BN targets are shown in figure 4. According to these results, the production of $^8$Li using the two stages setup is lower than $^6$He by about one order of magnitude.

**CONCLUSIONS**

During this work we studied the optimization of the two stage target setup from several aspects. Some of the results were expected and some gave new insights.

After careful examination of the results, we reached the following conclusions:

1. The distance between the two targets has a strong effect on $^6$He production, and should be as short as technically possible.
2. The specific shape of the secondary target does not have high effect on the total $^6$He production, and should better be chosen according to production and extraction simplicity considerations.
3. For high production, the secondary target total volume should be as large as reasonably possible, with a slight advantage for taking shorter target lengths rather than larger target radiuses. However, extraction considerations might limit the volume.
4. A reflector may increase the $^6$He production by only ~10%. The reflector contribution reaches saturation at a thickness of 7 cm.
5. The total production of $^6$He depends approximately linearly on the secondary target density and therefore low target densities could be compensated by taking higher volumes.
6. For a 2 mA, 40 MeV deuteron beam, the upper limit for the $^6$He production rate via the two stage targets setup is $\sim 6 \times 10^{13}$ atoms per second.
According to our calculations, we achieve a production rate of the same order of magnitude as the production rate of $^6$He using a 2.2 GeV and 100 $\mu$A proton beam and a spallation convecter proposed as a driver for beta-beam [9,10]. The extraction efficiency and the optimal BeO target density for efficient $^6$He extraction are still unknown. Therefore, depending on the extraction optimization, a 40 MeV (or a bit higher) and several mA deuterons LINAC could match the performance of the beta-beam proposed driver.

In the near future we intend to perform experiments in order to verify the optimization calculations and to investigate $^6$He and $^8$Li extraction methods.

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Neutron Diffraction in Archaeometallurgy: an Example of Possible Use in the SARAF

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INTRODUCTION

Neutron diffraction analyses of ancient metals show that this method is capable of detecting differences in the inner composition and microstructure of ancient metal objects. This gives archaeometallurgists a powerful non-destructive tool to help generate a better understanding of ancient modes of production (i.e., casting direction and mechanical and thermal treatment of selected parts of the original cast). Introducing this method here in Israel will allow us to study unique and precious archaeological metal artefacts that cannot be taken abroad or cut or drilled for sampling.

We carried out neutron analysis (on KARL diffractometer in the IRR-1 at NRC-Soreq) on two Middle Bronze Age I "Eye" axes, one made of bronze and the other of silver (Figure 1) and found recently in the ancient cemetery of Mahzevat Shuni. Both are rare finds and the silver axe is unique in the archaeology of Israel. Preliminary results confirmed its ability to clearly identify and analyze metal compositions and phases, identify and estimate the content of the majority of crystallographic impurities, and obtain information on the texture of the studied alloy. Such analyses can be optimized further by refining the present instrument’s parameters.

Figure 1 – Bronze eye axe (a), silver axe (b), mounted on the diffractometer (c).

We present this study as a good example out of many other possible studies that can be undertaken using the neutron diffraction method. One of our future aims in this study is to establish the limiting spatial resolution and to suggest improvements in the present instrument to achieve the highest resolution possible. Building an improved neutron diffractometer at the SARAF, based on the present instrument will enable us to further establish the neutron diffraction method for materials science studies in Israel.
EXPERIMENTAL CONDITIONS

Room temperature neutron diffraction (ND) measurements were carried out on the KARL diffractometer in the IRR-1 at the Soreq Research Centre. Neutrons from the reactor were monochromatized using a pyro-graphite monochromator, resulting in a 2.421 Å neutron beam. The multi-detector (bank of 14 3He detectors with angle step 3.5°) was used for registration the ND pattern in the scattering angle range of 2θ=20°–120°. ND data were collected separately from the middle and blade parts of both copper and silver axes, and also from pure (3N) silver and copper powders as standards. The primary beam formed by Cd apertures had the sizes 10 mm x 15 mm for the middle part of each axe, and the sizes 10 mm x 10 mm for the blade part. ND data are analyzed using the Rietveld refinement analysis with the FULLPROF program [1].

For XRF analysis of the axes a bench top model EX-310LC (Jordan Valley Co) was used [2]. The conditions used for the analysis were 35 kV of a rhodium tube and a primary beam size of 2 mm in diameter.

Results of XRF analysis and ND analytical data

The results of the XRF analysis show that one axe was made of copper alloyed with tin i.e. bronze (Cu + 5.5%Sn) (Figure 1(a)) and the other axe was made of silver alloyed with copper (Ag + 23%Cu in the corrosion layer) (Figure 1(b)). The results represent corroded surface layers of about 10 microns in depth.

**Bronze axe**

On the ND patterns of the copper powder and the middle and blade part of the bronze axe (Figure 2) the strongest lines are identified as {111} and {200} of a face-centered cubic lattice with cell parameters in the range of 3.64 – 3.68 Å, in agreement with the data for the lattice type and the unit cell parameters of copper element [3] and copper – tin bronze [4]. It means that the bronze axe contains mainly the Cu – Sn solid solution with cell parameter increased in comparison with pure copper (Table 1). This increasing of the cell parameter is caused by substitution of copper atoms in the lattice by the bigger tin atoms.

![Figure 2 - Rietveld refinement analysis of the ND data taken from the middle part of the copper axe. Solid circles, continuous line and dashed line are measured data, refined](image-url)
Table 1. The cubic lattice constants (in Å) for solid solutions found in the bronze axe and in the silver axe. Number in the parentheses is estimated error.

<table>
<thead>
<tr>
<th>Solution</th>
<th>Middle part of the axe</th>
<th>Blade part of the axe</th>
</tr>
</thead>
<tbody>
<tr>
<td>(Cu – Sn)$_1$</td>
<td>3.648(5)</td>
<td>3.629(5)</td>
</tr>
<tr>
<td>(Cu – Sn)$_2$</td>
<td>3.689(5)</td>
<td>3.667(5)</td>
</tr>
<tr>
<td>(Ag – Cu)</td>
<td>4.077(5)</td>
<td>4.076(5)</td>
</tr>
</tbody>
</table>

Lines in the diffraction patterns deduced from both parts of the bronze axe are significantly wider than the corresponding line widths of the copper powder (Table 2). Such a situation is typical for inhomogeneous solid solutions with composition varied from one part of the object to the other. For describing this solution we used the model, where two bronze phases were introduced with different cell parameters, representing two different tin percentages in the bronze. As result we received two values of the tin concentration – about 5% (Cu – Sn)$_1$ and 12% (Cu – Sn)$_2$.

These results are similar for both part of this axe (Table 1). A small decrease in the lattice constant (and correspondingly in the Sn content) is observed when comparing results from the middle part of the axe to results from the blade. Yet, this change is not far from one standard deviation of the experimental method (Table 1). Therefore, we used the averaged data of the lattice constants for the determination of the Sn content.

ND patterns of the middle and blade parts of the bronze axe contain several additional small diffraction lines (Figure 2). These lines belong to impurity phases among which the Cu$_{0.85}$Sn$_{0.15}$ [5], CuBr$_2$ [6], and the Cu$_2$O [7] phases. The existence of the Cu$_{0.85}$Sn$_{0.15}$ and Cu$_2$O impurities is commonly found in bronze artifacts [4]. The CuBr$_2$ phase could be interpreted as corrosion product.

Table 2. The widths, w (in degrees), of the neutron diffraction lines \{111\} and \{200\} of copper powder and bronze axe, for silver powder and silver bronze axe. Values are deduced by a single Gaussian fit to the neutron data.

<table>
<thead>
<tr>
<th>Object</th>
<th>Cu powder</th>
<th>Bronze Axe</th>
<th>Ag powder</th>
<th>Silver Axe</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>middle</td>
<td>blade</td>
<td>middle</td>
<td>blade</td>
</tr>
<tr>
<td>w(111)</td>
<td>0.95(2)</td>
<td>1.40(3)</td>
<td>1.43(2)</td>
<td>0.86(3)</td>
</tr>
<tr>
<td>w(200)</td>
<td>1.31(3)</td>
<td>1.93(5)</td>
<td>1.70(4)</td>
<td>1.06(4)</td>
</tr>
</tbody>
</table>
Figure 3 – Rietveld refinement analysis of the ND data taken from the blade part of the silver axe. Solid circles, continuous line and dashed line are measured data, refined model and the difference between them, respectively.

Ag axe
On the ND patterns of the silver powder and the middle and blade part of the silver axe (Figure 3) the strongest lines are identified as (111), (200), and (220) of a face-centered cubic structure with cell parameters in the range 4.07 – 4.09 Å, in agreement with silver element [8] and silver – copper alloy cell parameters [5]. It means that the silver axe contains mainly the solid solution with cell parameter decreased in comparison with pure silver. This decreasing of the cell parameter is caused by substitution of silver atoms in the lattice by the smaller copper atoms. Assuming limited solubility of copper in the silver metal and the linear dependence of the lattice constant on the concentration for the solid solution (i.e. Vegard’s law) 3 - 4% (atomic) or about 2% (in weight) of copper is estimated to be present in the main part of the silver axe. ND patterns of the middle and blade parts of the silver axes contain also several small diffraction lines in addition to the strong silver lines discussed above (Figure 4). These lines belong to several phases among which Cu is the strongest. The existence of Cu is typical for Ag – Cu alloys with Cu concentration higher than the solubility limit (about 9% in weight) [9]. Additional observed phases - AgCl [10] (in both part of the silver axe), and CuCl₂ [11] (in the middle part only) - could be interpreted as corrosion products.

Unlike the bronze axe, the silver diffraction lines in both parts of the silver axe have similar width as the silver powder lines.

DISCUSSION AND CONCLUSIONS
The above described neutron diffraction data are similar to data of the internal structure of an as-cast non-homogeneous bronze without additional thermal and/or mechanical treatment [4]. Annealing and cold work of the blade area after casting, even if it was conducted originally could not be seen in this case due to heavy corrosion and a need of a more detailed neutron diffraction study with better space resolution (about 1 mm). The absence of any sign of treatment after casting in the silver axe could indicate the use of this object as a ceremonial object rather then a 'real' weapon.

It is well known that as a result of the big difference in the melting temperatures of the Cu and Sn metals the solidification process starts from growth of grains (dendrites) containing solution of pure Sn content and finishes by solidification of Sn rich solutions (between the dendrites). This phenomenon is called "dendrite liquation" [9]. For bronze casting this liquation gives the minimal Sn content close to the surface and the maximal Sn content in the bulk. From our ND results the bronze axe contains two solutions (12 and 5% Sn). From our XRF data the surface layers contain 5.5% Sn. It means that the Sn rich part is placed indeed in the bulk in accordance with expected situation for bronze cast objects.

The solubility limit of Cu in Ag is not more than several percents. Thus, the Cu content, measured by the XRF (about 23% Cu), suggests the existence of two phases based on the Ag and Cu metals. This is in qualitative agreement with our ND data, where accurate quantitative interpretation is difficult due to the influence of corrosion products and strong neutron attenuation by silver.
The data measured in this work show no difference between the blade and middle part of both silver and bronze axes, albeit a hint that such a difference does exist in the bronze axe (please see cell parameters in Table 1). A more spatially resolute ND measurement is required to more accurately determine whether a difference does exist or not. This will enable us to shed more light on the means of manufacturing the studied axes. However, it is clear from our present relatively low resolution ND data that this method is able to easily observe the original artifact’s composition regardless of the existence of corrosion and without the need to cut or drill the object.

REFERENCES
[8] See [3], Gruppe 3, Band 6, Seite 1.

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Beam Commissioning of Phase I of the SARAF Accelerator

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INTRODUCTION
The Soreq Applied Research Accelerator Facility (SARAF) linac's [1] injector consists of a 20 keV/u protons and deuterons ECR ion source (EIS), a 5 mA low energy beam transport (LEBT) [2] and a 1.5 MeV/u, 4 mA, 176 MHz, 4-rod RFQ [3] (figure 1). The RFQ is followed by a short medium energy beam transport (MEBT) and a dedicated diagnostic plate (D-plate) for beam measurements. After beam commissioning of the RFQ the prototype superconducting module (PSM), housing six 176 MHz, s.c. half-wave resonators, will be installed between the MEBT and the D-plate finalizing phase I of SARAF.

This paper describes the commissioning results of the ion source with $H^+$, $H_2^+$ and $D^+$ particle beams and of the RFQ with $H^+$ beam.

Figure 1: Proton/Deuteron linac installed at SARAF. The components are (from right to left): ECR ion source (covered by HV/RF cage), LEBT, RFQ and D-plate. The PSM is placed off the beam line waiting to be moved to its final position downstream of the RFQ.

BEAM MEASUREMENTS AND RESULTS OF THE ION SOURCE
The particle beam of the ion source was commissioned and measured with the beam diagnostics built in the LEBT. The huge range of beam current was realized by varying gas flow, magnetron power, solenoid field at the plasma chamber and variable aperture in the LEBT. The results of the transversal emittance measurements (slit-wire scans) are summarized in table 1. Figure 2 shows the improving effect of the aperture on the emittance.
<table>
<thead>
<tr>
<th>particle beam current</th>
<th>rms emittance (100%, normalized)</th>
</tr>
</thead>
<tbody>
<tr>
<td>specification: 0.04 to 5 mA</td>
<td>specification $X/Y$: 0.2 / 0.2 $\pi$ mm mrad</td>
</tr>
<tr>
<td>protons [$\pi$ mm mrad]</td>
<td>deuterons [$\pi$ mm mrad]</td>
</tr>
<tr>
<td>5.0 mA</td>
<td>0.2 / 0.17</td>
</tr>
<tr>
<td>2.0 mA</td>
<td>0.13 / 0.13</td>
</tr>
<tr>
<td>0.04 mA</td>
<td>0.18 / 0.19</td>
</tr>
</tbody>
</table>

Table 1: Emittance measurement results of the ion source during site acceptance tests

Figure 2: Effect of the variable aperture. The low current tail seen in the phase-space contour plot (up) is cut away (bottom) by the aperture installed in the LEBT, improving the emittance.

**BEAM MEASUREMENTS AND RESULTS OF THE RFQ**

Since the beam power behind the RFQ can be high (6 kW for 1.5 MeV and 4 mA CW proton beam) the commissioning was done in pulsed mode at low duty cycles not to destroy beam intercepting diagnostics [4]. The generation of short beam pulses is shown in figure 3. The diagnostic signals were analyzed 10 µs before the end of the pulse when the beam current reached equilibrium. The transmission through the RFQ was determined by current measurements in front of the RFQ with a Faraday Cup and after the RFQ with a Faraday Cup and a current transformer (Bergoz MPCT). The energy was determined by time-of-flight measurements analyzing the sum signals of two beam position monitors in the MEBT and the signals of two phase probes in the D-plate. The results of these measurements for various RFQ forward power values, shown in figure 4, indicate that the optimum power level is 62 kW.
Figure 3: Pulsed beam current mode for RFQ commissioning. Short pulses behind the RFQ are realized by combining 2 ms pulses of the ion source with high duty factor pulses of RF power for the RFQ during a short overlapping time. The measurement trigger is set close to the end of the current pulse.

Figure 4: Proton beam transmission and energy measurements as a function of RFQ input power
The longitudinal emittance was determined by bunch length measurements with two Fast Faraday Cups (bandwidth 6 GHz) at two different distances behind the RFQ. The results shown in figure 5 also point to an optimum RFQ forward power of 62 kW. The analyses of beam measurements were followed by multi particle simulations performed by Soreq [5].

![Figure 5: Measurements of bunch length and RFQ longitudinal emittance as a function of RFQ input power](image.png)

**NEXT STEPS**
A misalignment of the RFQ electrodes was found by measurements with a laser tracker. This can explain the poor transmission of 70% and the beam steering out of the RFQ. The commissioning of the RFQ will be continued after re-alignment with proton beam for transmission optimization and transversal emittance measurements. After RFQ conditioning to the power level required for deuterion operation it will be commissioned with H2+ as a deuterion substitute.

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Locating, Identifying and Quantifying Radioactive Sources in an Inaccessible Area

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INTRODUCTION
In April 2007 an international In-Situ Intercomparison Scenario (ISIS2007) exercise was organized in cooperation between the Austrian Research Centers ARC and the IAEA and was conducted at Wiener Neustadt (Austria). A team from the Radiation Safety Division at Soreq NRC took part in the exercise. The exercise consisted of 10 tasks oriented to exercise identification and quantification of radioactive contamination due to reactor release, radiological terror or other accidents with radioactivity, which result in radiological debris, hidden sources, and unknown sources in containers, either shielded or unshielded cases. Several tasks were designed to test special skills, like the ability of the team to identify and quantify a buried source.
In this work we present Task No. 2 called by the Organizers "Localization – Drive by" and estimated by them to be "the most demanding task of the whole exercise"[1]. We also present the proposed solution of our team, which is based on the combination of signal intensities and geometrical properties of the source-detector configuration.

EXPERIMENTAL
A field of debris, from the era of the two World Wars at the abandoned ruins of the industrial installations of Tritolwerk (a TNT production plant), was defined by a perimeter as an inaccessible area. Its dimensions were about 300 X 100 m². Within these boundaries sources were hidden and the teams were requested to locate, identify and quantify them.
The measurement methods include a first drive-by screening from a vehicle, locate the high dose-rate points and conduct gamma spectrometry measurements at these selected sites along the peripheral one-way road. The time limit for the task was 2 hours.
The instrumentation used by us was the highly sensitive dose-rate meter IdentiFINDER (Target, Germany) for the detection and preliminary localization stage and a portable p-type coaxial HPGe detector of 65% relative efficiency and 2.05 keV energy resolution (Canberra, USA) for the spectrometric stage. Geographical coordinates of measuring points were obtained by GPS readings.
The unknowns of our problem, on top of identification, are the location and the activity of the source.
The mathematical approach to locate a source is based on finding by mathematical tools the geometrical loci that the source can be at. An intersection point of the loci is the source location and its activity can be calculated by the distance (squared) between the source and detector.
Let \( x_i, y_i \) be the coordinates of the detector in measurement \( i \), where the count-rate of the gamma line was \( C_i \). Let also \( \varepsilon \) be the efficiency of the detection system to the gamma line for the source at 1 meter distance. \( A \) is the unknown activity at the source location \( x, y \), and let \( \rho \) be the branching ratio of the specific gamma line.
The count rate at a point \( i \) thus obeys the equation:

\[ C_i = \frac{A_p e}{(x - x_i)^2 + (y - y_i)^2}, \text{ for } i = 1, 2, 3, \ldots, n \]

For each pair of points \( i \) and \( j \) we solve the equations for \( C_i \) and \( C_j \) and get an equation of a circle, on whose points the source can be located (the geometrical locus of the source).

The center of the circle (as an example of solving for points 1 and 2) is at:

\[ x_0 = \frac{c_1 x_1 - c_2 x_2}{c_1 - c_2}, \quad y_0 = \frac{c_1 y_1 - c_2 y_2}{c_1 - c_2} \]

And the radius of the circle \( R \) is given by

\[ R^2 = \frac{c_2 (x_2^2 + y_2^2) - c_1 (x_1^2 + y_1^2)}{c_1 - c_2} + (x_0^2 + y_0^2) \]

One gets another circle from solving for another pair of measuring points, 1 and 3 or 2 and 3. The intersection of two circles defines two points at which the source can be located. In order to find the intersection of two circles, let us denote the centers of the two circles by \( x_a \) or \( y_a \), \( y_b \), and the corresponding radii \( R_a \) or \( R_b \). The intersection is solved as the points that are fulfilling the equations of both circles. Their points of intersection are given by [21]:

\[ x = \frac{x_b + x_a}{2} + \frac{(x_b - x_a)(R_a^2 - R_b^2)}{2d^2} + \frac{y_b - y_a}{2d^2} \sqrt{((R_a + R_b)^2 - d^2)(d^2 - (R_b - R_a)^2)} \]

\[ y = \frac{y_b + y_a}{2} + \frac{(y_b - y_a)(R_a^2 - R_b^2)}{2d^2} + \frac{x_b - x_a}{2d^2} \sqrt{((R_a + R_b)^2 - d^2)(d^2 - (R_b - R_a)^2)} \]

Where \( d \) is the distance between the centers of the two circles:

\[ d = \sqrt{(x_b - x_a)^2 + (y_b - y_a)^2} \]

One of the two points will lay out the perimeter of the field, thus leading to one single possible location.
RESULTS
Four high-dose-rate points, at which the rate was about 0.2 μSv/h, while the background was 0.06-0.1 μSv/h, were selected. One of the spectra is presented in Figure 1 where $^{60}$Co, $^{137}$Cs and natural radio nuclides are clearly observed.

![Figure 1. A spectrum collected at the fence of the field](image)

The spectra were analyzed and "apparent" activities at 1 meter were calculated at each measuring point for each radionuclide. The $^{60}$Co was identified as an unknown source while the $^{137}$Cs was misinterpreted as environmental $^{137}$Cs due to the fact that the count-rates were not distinct from the background in this vicinity. The ISIS team also placed in the field a $^{241}$Am source, which was not identified by us, most probably because of the relatively high background continuum of the detector at this low energy (59.6 keV). The spectrum acquired at a measurement point east of the hill (see the north-eastern corner of the field in Figure 2) did not show $^{60}$Co peaks because of the attenuation by the hill (the source was west of the hill).

The location of the $^{60}$Co source was calculated according to the above method and the 3 circles were found to intersect at 2 points. One of these 2 points was out the perimeter of the field, thus leading to one single solution (location) at E16°16.500, N47°51.263, in good agreement with the real position of the source as seen in Figure 2.

After defining the location, the activity of the source was directly extracted from the count-rate equation I above. The calculated activity of the $^{60}$Co source was $400 \pm 150$ MBq which is in acceptable agreement to the $610 \pm 10\%$ MBq stated by the organizers\[11\]. The GPS inaccuracy of 4 to 5 meters, in relation to distances that are of the order of 20 meters, is probably the direct cause of an inaccuracy of 50 to 60% in the activity determination.
Figure 2. Localization of hidden sources in the field of Task 2 [1]

The red spot is our own estimate of the position of the $^{60}$Co source.

CONCLUSION
The location of the $^{60}$Co source was found to be verified but only qualitatively, since the final published report [1] included only a rough sketch and not exact coordinates which was our only way of reporting the results. Our reported activity was 35% below the 610 MBq that was claimed by the organizers. This inaccuracy is probably due to incorrect localization, but rather good for such a tough exercise.

REFERENCES

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Gamma Measurement System for Calculation of Calcium Transport in Tomato Plants

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INTRODUCTION

Xylem transport has been studied for many years because of the dominant role that it plays in the flow of water through the plant and in nutrient delivery from the root to the shoot (1). The xylem, the main pathway for Ca and Sr, is a transport conduit operating under negative pressure. The problems in xylem sap sampling arise because negative pressures are present under normal conditions of transpiration, thus incision causes entry of air rather than leakage of the xylem sap. Tracer studies are a more suitable method. The use of gamma radiating nuclides capable of penetrating the plant tissue, allows nondestructive measurements (2). For Ca transport research the 47Ca existing gamma emitting nuclide, has low applicability due to its short half-life of 4.5 days. The choice of Sr enables the use of 85Sr, a high-energy (514 keV) gamma emitter with adequate half-life of 64.8 days. A gamma measurement system has been developed in cooperation with the NRCN and the Faculty of Agriculture in the Hebrew University. The system consists of five gamma detectors. Each detector is localized in a lead collimator. Three detectors are attached along the plant stem and two detectors inspect the leaf petiole and the tomato fruit peduncle. The information processing circuit in each detector includes a microcontroller which samples and transmits the measured data from the gamma sensor, via RS485 serial communication network, to a data collection system. The developed PC software enables continuous exhibition of the measured data of each detector.

RADIOACTIVE TRACER

In order to measure the Ca transport using a radioactive tracer, the natural choice was a calcium radio-isotope. Calcium Radioisotopes are 41Ca, 45Ca, 47Ca and 49Ca. All of these isotopes emit electron or beta particles and gamma rays. Electron or beta particles are not suitable for in-vivo measurements since they are absorbed in the plant stem. 41Ca and 45Ca are long-lived isotopes with gamma energy of 3keV and 12 keV respectively. The gamma energy of these isotopes is too low for penetrating through the stem and these photons are completely absorbed before reaching the sensor. 47Ca (Ey: 1297 keV, probability: 0.75) and 49Ca (Ey> 3 MeV, probability > 0.98) emits high energy gamma rays but due to their short half-life (4.5 days and 9 minutes respectively) they are not applicable to the experiment.
Another isotope with similar chemical and biological characteristics to Calcium was therefore considered. Strontium isotopes match these characteristics. Strontium radioisotopes are $^{82}$Sr, $^{85}$Sr, $^{89}$Sr, $^{90}$Sr, $^{91}$Sr, $^{92}$Sr and $^{93}$Sr. $^{91}$Sr, $^{92}$Sr and $^{93}$Sr are short half-life, $^{82}$Sr emits only beta and low energy gamma rays ($E_\gamma < 15$ keV), $^{90}$Sr emits only beta. The only applicable strontium isotope is $^{85}$Sr ($E_\gamma$: 514 keV, probability > 0.99). For radiation protection calculation a 100 cm length line source of 200 μCi was simulated. This source simulates the gamma emission all along the stem. Exposure rates were calculated using dedicated software. The calculated exposure rates were 400 μR/h, 82 μR/h, 33 μR/h and 17 μR/h for distances of 30cm, 80cm, 130cm and 180cm respectively. The calculated fields enable a safe working environment, assuming short time equipping installation near the plant, and long time measurements for distances longer than 100cm.

**SYSTEM DESIGN**

In order to achieve high sensitivity for gamma rays measurements a Scintillator type detector has been selected. Rotem Ind. RP-11 detector (3) contains a highly efficient NaI(Tl) 2" * 2" scintillator coupled to a Photo Multiplier Tube (PMT). This detector includes a microprocessor circuit, HV power supply, Single Channel Analyzer (SCA), pulse shaper, RS-232 and RS-485 communication ports.

The entire measurement system consists of five detectors, three for monitoring the stem and two for monitoring the fruit and leaf (fig. 1). Each of the detectors was installed into a 1cm width lead collimator. Two different types of collimators were designed, a 1cm width entrance window (fig. 2) for measuring the stem and leaf petiole, and a circular 2" diameter entrance window for measuring the fruit. The detectors for measuring the stems were installed at about 50cm, 100cm and 150cm height. The stems, the leaf petiole and the fruit were attached to the collimator's entrance window.

The gamma emission has been calculated for $^{85}$Sr diluted solution with an activity of 0.125 μCi/cm³. Assuming 10% xylem/stem ratio and 1cm stem diameter, the active volume facing a 2" detector is 0.40cm³ and the maximum activity is 0.05 μCi (~1800Bq). The rate of gamma rays reaching the detector is about 400 γ/s. The RP-11 efficiency for 514 keV gamma rays is higher than 50%. For the calculated parameters, the detector counts rate increase over background is higher than 200cps. Background count rate, including gamma photons reaching the Scintillator from other parts of the plant, is about 160 cps. The expected count rate for the measured xylem window and background is higher than 360cps. In order to enable detection of lower concentrations measurements were integrated for a 10-second acquisition period.

The internal microcontroller in the RP-11 probe counts the PMT output pulses; calculates the count rate and the total counts in the selected period. Data collection software has been developed and installed in a PC computer. The computer communicates with the detectors via RS-485 serial network. The software displays the counts of each probe and its acquisition time. Measurements are stored and sorted for further data analysis.
Figure 1: Five-detector monitoring system

**SYSTEM TEST AND CONCLUSIONS**

Preliminary tests were performed on five different plants. Each plant, growing in an aerated hydroponic solution, was added with a 200 μCi $^{85}\text{Sr}$ diluted solution. The solution passage from the roots throughout the plant was monitored with the presented Gamma Measurement System. The detectors and the acquisition software functioned well in all performed tests. Related data analysis is performed by the Faculty of Agriculture in Rehovot and the biological aspects of the measurements on xylem transport will be presented separately. Obtained preliminary results demonstrated the applicability of such system for in-vivo measurements of transport processes.

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Calcium Translocation and Whole Plant Transpiration
Noninvasive Measurements using Radio-Strontium as Tracer

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INTRODUCTION
Calcium (Ca) in plants has essential roles affecting its tissue mechanical strength (related to handling, storage and shelf life) and its tolerance to biotic and abiotic stresses¹. Understanding Ca movement within the plant, its allocation to the different plant parts with time and the factors affecting it, has a high agronomic and economical value as it will allow improving Ca nutrition practices to give higher quality end product with direct economic benefit. Ca dynamics in the whole plant is considered to be coupled with water movement driven by transpiration and was shown to accumulate mainly in transpiring organs in a process affected by various environmental conditions, at both the canopy as well as at the roots level. However, controversies still arise in that relation². Furthermore, as Ca moves mainly in the xylem, a transport conduit under negative pressure, and any attempt to sample it en-route will cause cessation of flow. As a result, the use of cumbersome destructive methods, which has limited research scope due to time and space constraints, brought only circumstantial evidence leaving much of the allocation process still obscure³. Nondestructive techniques enable a more coherent view and in the last few years much attention was given to sensing combined with imaging. Neutron Computed Tomography (CT) was used to image water in flowers⁴ and Positron Emission Tomography (PET) was utilized to follow several elements in various plant species⁵,⁶. Water has been traced also using Magnetic Resonance Imaging (MRI)⁷ while functional MRI enabled looking into flow⁸ with the later used to image solute transport as well⁹. Alas, Ca nuclides are incompatible for the above mentioned techniques. Looking for a Ca analog, Strontium (Sr) was found to behave in similar ways in plants¹⁰ as well as in the more complex environment of human clinical research¹¹. Having a high energy gamma emitting nuclide (⁸⁵Sr) that can be detected outside the plant, make remote sensing feasible. In an attempt to clarify and explore the above, we developed and utilized a novel imaging technique for calcium translocation using radio-Sr analog tracer and custom made sensing devices together with transpiration indicative measurements on a whole, living plant.

MATERIALS AND METHODS
Tomato plants (Lycopersicon esculentum L., Var. 870, Hazera Genetics, Israel) were grown at a growth chamber in the faculty of Agriculture (Rehovot) under controlled climate of day/night temperatures of 28/22 °C and relative humidity of 60/90 % respectively. Each plant was grown in a 5-litter bucket containing half-strength Hoagland solution and was continuously aerated.
After three months, reaching approximately a height of 160 cm and having three fruit bearing clusters, eight plants were transferred each to a 2-litter cylinder filled with a half strength Hoagland solution and moved to the faculty radioactive allowed facility (radio-room) where the solution was aerated again.

At the radio-room the plants were subjected to temperature of 24/16 °C and relative humidity of 40/65 % during the day and night respectively. Light was supplied between 08:00 to 20:00 by two cool mercury lamps (each delivering photon flux of 400 micro-mole m\(^{-2}\) s\(^{-1}\)). Plants were arranged in four pairs. The 1\(^{st}\) plant of each pair was placed on a weighing lysimeter (12”x18” base with 1042-G transducer; Tedea-Huntleigh, IL) and its weight measured every 6 seconds, averaged and recorded every minute in a data-logger (CR10; Campbell Sci.; UT, USA) and whole plant transpiration was calculated from weight loss. The 2\(^{nd}\) plant was installed with an array of 5 gamma radiation sensors (RP-11; Rotem Ind., Israel), each with a custom-made lead shield. The sensors and shields were mounted on a moving platform targeting the following locations: 1) on the main stem (MS) below the 1\(^{st}\) fruiting truss (FS), 2) MS below 2\(^{nd}\) FT, 3) MS below 3\(^{rd}\) FT, 4) Stalk of 2\(^{nd}\) FT 5) Leaf petiole adjacent to 2\(^{nd}\) FT. The sensors were connected to a PC via a custom-made communication device (Rotem Ind., Israel) and radiation activity was measured and recorded every 3 seconds. The system is described in detail in Wengrowicz et al., 2008. After 3 days of acclimatization, a radio-Sr solution (as \(^{85}\)SrCl\(_2\); Perkin-Elmer, MA, USA) with activity of 0.25 mCi was diluted in 10 ml of distilled water containing 4 mM Sr (as Sr(NO\(_3\))\(_2\); Merck, Germany) as a carrier, and added to the plant feed solution around noon. Every few days the sensors array platform was detached from the now radioactive plant to the assigned plant in the next pair, each sensor attached to its respected location and after a day, the radio-Sr was injected and radiation measured as previously described.

**RESULTS**

Example of paired plant measurements on day of injection is presented in fig. 1. Radiation levels (in cps) of the different sensors going upstream reacted sequentially to the radioactive Sr applied to the roots (13:00). The lower-most stem sensor (SB) showed marked increase in radiation after 30 minutes of injection followed by high rate increase. The stem sensor located at the middle of the plant (SM) reacted half an hour later with lower rate increase and the top-most stem sensor (ST) reacted after additional 30 minutes (1.5 hours from start) with even lower rate of increase. About an hour after lights went off (20:00), radiation rate increase declined (yet did not ceased). Comparing stem radiation measurements to the adjacent plant transpiration (Tr) show similar pattern with slope lowered with sensor height location. Sensors located on the 2\(^{nd}\) fruit truss (FM) and adjacent leaf petiole (LM) reacted after 2 hours from start yet increase was low compared to seen on the different stem locations.
Figure 1. Absolute radiation activity readings [cps] of the sensors array on the 1st day. Legend: ST – Main Stem above 2nd Fruit Truss, SM – MS below 2nd FT, SB – MS below 1st FT, LM – Leaf petiole adjacent to 2nd fruit truss, FM – Stalk of 2nd fruit truss, Tr – total transpiration of adjacent plant.

Figure 2 presents the above measurements as % from background radiation at location. At the end of the 1st day SB show the highest relative radiation followed by SM and ST. It can clearly be seen that all stem sensors measurements correlated with transpiration with smaller magnitude the higher the sensor is along the flow path. The leaf petiole (LM) and fruit truss stalk (FM) showed similar pattern however to a much lower extent.

Figure 2. Relative radiation activity readings [%] of the sensors array on the 1st day. Legend as in fig. 1
The 2nd day is shown in fig. 3. As lights turned on, transpiration increased together with marked increase in radiation in all stems sensors. However, after midday, although transpiration rate did not decline significantly, radiation rate declined. As lights were turned off and transpiration decreased markedly, only top stem sensor showed a rate decline while the two lower stem sensors continued with the same rate of radiation increase. Leaf and fruit sensors reacted in a similar pattern to transpiration, however with lower rates, and unlike the stem sensors, a rate decline can not be seen after midday. At the end of the day fruit truss stalk radiation increased to a higher level then leaf petiole.

![Graph showing radiation activity readings](image)

Figure 3. Relative radiation activity readings [%] of the sensors array on the 2nd day. Legend as in fig. 1

Measurements made in the following days (not shown) exhibited continues increase in the top-most stem, fruit and leaf sensors. Middle stem showed an asymptotic curve pattern while lower-most stem sensor, which measured at the first few days the highest rate increase, showed later a decline to relative radiation levels lower then the above positioned sensors.

CONCLUSIONS
Using sensitive gamma sensors with custom-made shields clearly enabled following transport and partitioning patterns of root applied radio-Sr in a whole, living plant. Sr transport through the main stem and to fruit truss stalk and adjacent leaf petiole followed the daily transpiration pattern as induced by light (supported by adequate environmental conditions). An afternoon depression was shown to occur in all locations measured. Stem readings presented a loading-unloading pattern (at the lower-most stem location) possibly indicating on stem Sr exchange capacity. We intend to introduce simultaneous sap-flow (water) measurements at similar locations as Sr radiation readings to enable detailed investigation of the coupling of water and Sr, as Ca tracer, for both transport and partitioning. Root and canopy treatments will be made to test the coupling hypothesis and the factors affecting it.

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Measurement of PET Tracers Low Activity Levels During Plasma Samples Analysis

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INTRODUCTION

Plasma metabolite analysis in the PET (Positron Emission Tomography) studies requires quantification of very low levels of radioactive materials during a rapid and sequential passage of plasma samples. Quantification is necessary for deriving a kinetic model that describes the transport and metabolism of the administered radiotracer in the body. Using kinetic models, information about a particular biological process in the human body can be determined from the concentration of the PET tracer in the blood plasma and tissues, during the PET scan. To estimate the kinetic model parameters, the amount of the parent compound in plasma at a given time point is needed. The plasma contains the original drug (parent compound) and its metabolite that was produced by enzymatic activity; therefore, metabolism of the radio-labeled tracer results in radioactive metabolites that cannot be differentiated from the parent compound, because both, the metabolite and the parent compound, carry the same radioactive isotope and emit photons with the same energy (511 keV). The total plasma radioactivity and the radiotracer decay time are known, but in order to define the kinetic model, an adjustment for the actual unmetabolized (parent) radiotracer requires correction at each time point. Metabolite analysis of the radiotracer includes blood sample collection with a known time stamp, separation of the plasma by centrifugation, separation of the plasma into its compounds (parent and metabolite) by High Pressure Liquid Chromatography (HPLC), and analysis by a radiation detector (radio-HPLC) to determine the percentage of the unchanged drug (parent compound). The detector system records the activity levels in the HPLC output line for each compound during the plasma analysis procedure (the metabolites as well as the original unmetabolized radiotracer).

Our goal was to design a high sensitivity measuring system to improve the detection level of the positron-emitting isotope and enhance metabolite analysis.

METHOD

The required system must be capable, not only of distinguishing between the two compounds, but also of measuring the amount of radioactivity in each. This task becomes further complicated when the parent compound has low activity levels, compared to its metabolites, which results in a low Signal to Noise Ratios (SNR). Therefore, a detection system suitable for analysis of plasma samples must meet the following four design requirements: a) Short measuring time; b) Quantitative analysis; c) High noise level; and d) Low Signal to Noise Ratio.

The typical approach for measuring of low activity levels during plasma samples analysis is based on coincidence measurements of the two annihilation photons. Coincidence measurements improve detection by reducing the noise level in the energy window whose channels are set for detecting the 511 keV photons, and through the use of the timing coincidence electronic circuit for the signal received from the two detectors. However, in this configuration sensitivity is not optimal since some of the events caused by the 511 keV photons may only be measured by one of the two detectors and therefore, are not counted.
Our approach is based on a new radiation detector design. This detector consists of a Bismuth Germanate \( \text{Bi}_4\text{Ge}_3\text{O}_{12} \) (BGO) scintillation crystal to detect 511 keV photons and a plastic BC-408 scintillator to detect the emitted positrons, both coupled to a single PMT. Since the BGO crystal is a high density scintillation material, it increased the probability of photon attenuation and efficiency for gamma photon absorption. Furthermore, the crystal is rugged and has non-hygroscopic properties that can simplify the design studies and the set-up for testing the different detector configurations. The plastic scintillator is a low density polyvinyltoluene based material (C\(_{10}\)H\(_{11}\)). The plastic is also a rugged and non-hygroscopic material. These properties, in addition to the fact that the scintillator is a low cost material, commercially available in different thicknesses and can readily undergo mechanical processing, makes it highly suitable for use with the new detection system. The PMT output signal amplitude, after interaction with the plastic, is approximately 40% higher than that produced by interaction with the BGO.

RESULTS
The results shown in Figure 1 demonstrate a major influence on the spectrum when measuring only the gamma photons compared to measuring both, positrons and photons using an \(^{11}\text{C}\) source. Using our detector design the detection sensitivity for measuring the combined signal from both positrons and annihilation photons was compared to that obtained by measuring only individual 511 keV photons. The main observable effects on the spectrum are as follows:

1. Shift in the number of events from the photo peak region into the high energy region. These events occur because the light output is caused by a combined signal from the positrons and the 511 keV photons. These are detected as a single event with a high amount of light signal. The broad high energy spectrum is also a result of events where only the positron was measured by the detector. This effect increases the measurement sensitivity.

2. Change in the percentage of counts in the low energy region (below 300 keV). The positron energy shifts the Compton events from the 511 keV photons into the energy window in the region of interest (ROI), i.e. >300 keV. This effect increases the measurement sensitivity by approximately 40%.
Figure 1 – Measured spectra from $^{11}$C detection with and without positrons.

A – Measurement of only 511 keV photons. The gross measurement for energies above 300 keV - 395K counts, counts at Photo Peak (511 keV) - 3.6K and counts at energy of 1.5 MeV equal to background level. B – Measurement of both the 511 keV photons and positrons. The gross measurement for energy above 300 keV - 361K counts, counts at Photo Peak (511 keV) - 1.0K and 0.2K counts at energy of 1.5 MeV. The number of counts in the spectrum at the energies below 300 keV is 36% for measuring only the gamma photons compared to only 14% for measuring the gamma photons and the positrons.

The spectrum change with regard to the distribution (percentage) of the number of counts within different regions of the spectrum as contributed to the output signal by the positron energy, was investigated using four PET isotopes. Results are summarized in Table 1. The linearity of the detector readings over a wide measuring range was used to derive a factor indicative of the improvement in the detector sensitivity. The factor was determined by the curve slope and calculated as the ratio between the detector output from the new configuration for energies above 300 keV (Full Spectrum region) and those above 880 keV (High Energy region), relative to the counts measured by the configuration that measured the 511 keV gamma photons only.

Table 1 – Fraction of counts in different energy regions detected for all four main PET isotopes

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Detection Configuration</th>
<th>Fraction of counts detected in regions above 300 keV</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Photo-Peak Region 300-600 keV</td>
</tr>
<tr>
<td>$^{511}$ keV Background Source</td>
<td>Gamma Only</td>
<td>0.92</td>
</tr>
<tr>
<td></td>
<td>PB</td>
<td>0.92</td>
</tr>
<tr>
<td>$^{F-18}$ Injection</td>
<td>PB</td>
<td>0.42</td>
</tr>
<tr>
<td>$^{C-11}$ Injection</td>
<td>PB</td>
<td>0.25</td>
</tr>
<tr>
<td>$^{N-13}$ Injection</td>
<td>PB</td>
<td>0.21</td>
</tr>
<tr>
<td>$^{O-15}$ Injection</td>
<td>PB</td>
<td>0.14</td>
</tr>
</tbody>
</table>

**Conclusion**

Increase in the detection sensitivity and improvement of the Minimum Detection Activity (MDA) were achieved by designing a new radiation detector composed of a plastic scintillator and a BGO crystal. The new detector is highly sensitive for measuring both, the decaying positrons and the annihilated 511 keV photons, producing a combined output signal. The resulting combined light output shifted the position of the summed light pulse to higher energies in a spectrum calibrated with 511 keV photons. By accepting only the events with energy above the photo peak of single photons, we were able to achieve much higher efficiencies than with the current radio-HPLC detectors, with minimal inclusion of was improved by approximately 780%, and the MDA by ~520%. By improving the measurements sensitivity an opportunity for identifying additional parameters that could even further improve the overall validity and reliability for kinetic modeling, became available.

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INTRODUCTION
Assessment of internal exposure to uranium is generally carried out by analysis of urine samples, and correlating the result according to the ICRP bio-kinetic model. Hair samples can also be used for a sensitive bio monitoring of environmental or occupational exposure. There are several advantages to analyze hair samples rather than urine. First, hair is an effective bio-concentrator for uranium and is a major excretion pathway. Second, hair samples are stable and the uranium content does not change over time. Third, hair does not require special storage and handling. And finally, hair has the unique ability to reflect the total body intake during an extended period and it is possible to trace changes over time. These features were recently demonstrated in a comprehensive joint study by groups from Finland and Israel, where hair samples were rinsed, digested and the uranium level determined by inductively coupled plasma – mass spectrometry (ICPMS).

RESULTS
Two different studies of the uranium content in single hair starnds will be described below. In one case, the subject moved from Israel, where the uranium level in drinking water was about 2000 ng/L to spend a year in Germany where the uranium content was 30 ng/L. In the other study, several hair samples were collected from workers who may have been occupationally exposed to uranium compounds.

In the first study, the uranium content was determined by a double-focusing sector field ICP-MS (ICP-SFMS, ELEMENT, Thermo Electron GmbH, Bremen, Germany) coupled with commercial laser ablation system LSX-200 (CETAC LSX-200, CETAC Technologies Inc., Omaha, NE, USA) at the Central Division of Analytical Chemistry Research Centre Juelich, Germany. In the second study, a Thermo Elemental Plasma Quad 3 ICPMS with a UV Microprobe I ablation unit, at the Environmental Analytical Laboratory, Rotem Park, Israel, was used.

Figure 1 shows an electron microscope image of an intact hair strand (on the right) and a hair strand after being analyzed by laser ablation (on the left). The diameter of a hair strand is typically about 80 μm, and that of the laser beam was around 20 μm, but after several laser pulses (shots) impinge on the same area the fissure along the hair strand is slightly enlarged.
Figure 1. An electron microscope image of an intact hair strand (on the right) and a hair strand after being analyzed by laser ablation (on the left).

Figure 2 shows the concentration of uranium along a single hair strand as a function of the distance from the scalp.\(^{(5)}\). The uranium concentration in the distal part of the hair strand is 220 ng/g and reflects the uranium content in Lehavim drinking water (2000 ng/L) while in the proximal part the uranium concentration in hair is 20 ng/g corresponding to the uranium content in Bonn drinking water. Considering that the rate of hair growth is about 1 cm per month, the gradual decrease in the uranium content in the hair lasted about four months, indicating the time it took for the body to reach a new equilibrium.

Figure 2: Change in uranium concentration along a single hair strand collected from a person who relocated from Lehavim to Bonn, reflecting the level of uranium in drinking water (dotted lines).
Figure 3 shows a high resolution image of the longitudinal distribution of the uranium content along a single hair strand of a person who was involved in an exposure incident \[^4\]. The average uranium concentration was 2270 ng/g, as determined from a sample of digested hair, does not reflect the incident. The true picture can only be seen with meticulous laser-ablation analysis: the normal uranium content in hair of this person is around 600 ng/g, but due to the incident the uranium concentration peaked at 12000 ng/g. The duration of the incident can be estimated from the X-axis where a length of 0.33 mm (one day's hair growth) was scanned by the laser beam in 2.2 seconds, so that the effect of the exposure incident on the body lasted about 3 days, until the concentration in hair returned to its normal level.

Figure 3. The longitudinal concentration of uranium along a single hair strand of a person who was involved in an exposure incident.
CONCLUSIONS
Laser-ablation-ICPMS is a powerful tool for assessment of internal exposure to uranium. As very detailed information may be obtained by this technique it can serve as a means for detection of exposure incidents long after they have occurred and the duration of the effect may also be determined. Hair samples can be easily acquired and stored for long periods. Analysis of digested hair sample reflects the integrated body content of uranium, while analysis of a single hair strand can indicate whether an exposure incident occurred and its severity.

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Counting Delayed Neutrons in Uranium Analysis-
Statistical Interpretation of Experimental Results

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INTRODUCTION
Instrumental neutron activation analysis of uranium and other fissionable isotopes by counting
delayed neutrons (DN) has been extensively used since the year 1960 at Soreq NRC(1). In the
previous 23rd INS Conference, we have presented the renovation of the counting assembly,
including the integrated electronics and the software developed for manipulation and processing of
data(2). In that presentation, the counts registered from a natural uranium sample were analyzed by
fitting a decay function of two exponentials with half-lives of approximately 25 s and less than 1 s.
In this paper we present a different statistical approach to interpret experimental data by using
well-known half-lives of the DN groups.

THEORETICAL
Keepin et al.(3) performed very precise measurements of DN in several fission systems and were
able to deconvolute the decay curves by a numerical method into groups of different half-lives $T_i$
and amplitudes $\beta_i$. The deconvolution function of the time $t$ was

$$N(t) = \sum_{i=1}^{n} \beta_i \exp\left(-t \frac{\ln(2)}{T_i}\right)$$

Decay constants of the groups are given by $\lambda_i = \frac{\ln(2)}{T_i}$. An optimal fitting was obtained for $n = 6$ groups and the analysis of the thermal neutron fission of $^{235}\text{U}$ yielded the parameters shown in
Table 1.

<table>
<thead>
<tr>
<th>Group</th>
<th>$T_i$ (s)</th>
<th>$\lambda_i$ (s$^{-1}$)</th>
<th>$\beta_i$</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>55.72</td>
<td>0.01244</td>
<td>0.033</td>
</tr>
<tr>
<td>2</td>
<td>22.72</td>
<td>0.03051</td>
<td>0.219</td>
</tr>
<tr>
<td>3</td>
<td>6.22</td>
<td>0.11144</td>
<td>0.196</td>
</tr>
<tr>
<td>4</td>
<td>2.30</td>
<td>0.30137</td>
<td>0.395</td>
</tr>
<tr>
<td>5</td>
<td>0.61</td>
<td>1.13631</td>
<td>0.115</td>
</tr>
<tr>
<td>6</td>
<td>0.23</td>
<td>3.01368</td>
<td>0.042</td>
</tr>
<tr>
<td>Sum</td>
<td></td>
<td></td>
<td>1.000</td>
</tr>
</tbody>
</table>

Identification of DN precursors among fission products is not within the scope of this work. For
example, $^{87}\text{Br}$ (half-life 55.65 s) is a single precursor in the 1st group.
Analysis of uranium at Soreq NRC is done by a 60 s irradiation, 20 s delay and 60 s counting.
Hence, the number of counts accumulated in the counting interval $[0, t]$ is given by

$$I(t) = \sum_{i=1}^{6} N_i \sigma_i \phi \beta_i \varepsilon_i \left(1 - e^{-60 \lambda_i} \right) e^{-20 \lambda_i} \left(1 - e^{-\lambda_i t} \right) / \lambda_i$$
where \( N_f \) is the number of \(^{235}\text{U}\) atoms in the sample, \( \sigma_f \) is the cross section of fission of \(^{235}\text{U}\) by thermal neutrons, \( \phi \) is the thermal neutron flux, and \( \varepsilon_i \) is the counting efficiency of DN in group \( i \), averaged on the energy spectrum in the group.

The energy spectrum of DN is rather constant in the groups and therefore, counting efficiency is constant and is group independent (\( \varepsilon_i = \varepsilon \)). Hence, the factor \( N_f \sigma_f \phi \varepsilon \) is group independent and we can assume it is equal to 1. The difference \( I(t) - I(t - 1) \) for \( 1 \leq t \leq 60 \) is equal to the number of counts registered in the 1-second interval \([t-1; t]\), i.e. the count-rate (CR, in cps) at the time \( t \).

The calculated CR of DN emitted in the thermal neutron fission of \(^{235}\text{U}\) as a function of the time \( t \) elapsed from the beginning of the measurement is shown in Figure 1.

![Figure 1. Calculated count-rate of DN](image)

**EXPERIMENTAL, RESULTS AND DISCUSSION**

The renovated counting assembly can give the number of counts registered in each second. A standard containing 58.6 \( \mu \)g of natural uranium gave the measured CR shown in Figure 2. A background of 0.51 cps was subtracted. The statistics of the number of counts in Figure 2 is of the order 100\(\pm\)10 cps, i.e. poorer than the very precise curves of Keepin et al. \(^3\), and therefore, does not allow a deconvolution into 6 groups. Instead, we will perform another fit of the calculated curve, by finding a factor which multiplies the calculated CR at each time \( t \) and equates it to the measured CR. The weighted average of the 60 ratios of CRs is 17.82 \(\pm\) 0.30. The fitted curve fits well the measured, as can be clearly seen in Figure 2.

The normalization factor \( k \) can be derived also by minimization of chi-square,

\[
\chi^2 = \sum_{i=1}^{60} \omega_i (m_i - kc_i)^2
\]

where \( m_i \) and \( c_i \) are the measured and calculated CR and \( \omega_i = m_i \) is the weight. The value \( k = 18.01 \) thus found by \( \chi^2_{\text{min}} \) agrees well with the weighted average 17.82 \(\pm\) 0.30 of CR ratios.

The dispersion of measured points relative to the fitted line behaves inversely with the CR.
Figure 2. Measured (diamonds) and fitted (line) count-rates of a 58.6 $\mu$g $\text{nat}^{{\text{U}}}$ standard

We can analyze the goodness of fitting by looking at the standard deviations $\Delta_i$, given by

$$\Delta_i = \frac{m_i - k c_i}{m_i^{1/2}}$$

In this Equation, the difference between measured and fitted CR is expressed in units of the standard uncertainty of the measured CR. Figure 3 shows the findings.

Figure 3. Standard deviations of the 58.6 $\mu$g $\text{nat}^{{\text{U}}}$ sample
It is easily seen in Figure 3 that deviations $\Delta$ are centered around 0 within the interval [-3;3], namely the difference between measured and fitted CR is smaller than 3 times the standard uncertainty of the measured. A very similar behavior as depicted in Figure 3, was found with smaller samples, of the order 10 $\mu$g $^{nat}$U. With respect to a confidence level, this is a satisfactory situation.

In counting DN we can discover a transitory event of the CR as a result of a background sudden rise, e.g. due to a rapid opening of a neutron beam tube in the reactor. This phenomenon can be detected by testing point-by-point the decay curve of the CR. A significant deviation ($|\Delta| \geq 5$) at a point, much greater than expected by pure statistics ($|\Delta| \leq 3$), indicates that an excursion in the CR had occurred. We did an experimental simulation by inserting a small ($10^4$ n/s) Pu-Be neutron source into the sample cavity of the assembly during a regular DN counting. Figure 4 shows the observed outcome of inserting the neutron source 3 times.

![Figure 4. Excursions in the DN counting of a 230 $\mu$g $^{nat}$U sample](image)

Work to detect such excursions in real-time, and automatically repair the decay curve by the software, is in progress. Our basic idea is to identify outlying points in the decay curve, using the statistical criterion ($|\Delta| \geq 5$), then remove and replace them by correctly computed values. Algorithms for testing, rejecting and computing points will be installed in the renovated software.

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Neutron Flux Measurements of a D-T Neutron Generator

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INTRODUCTION

A new D-T sealed-tube neutron generator model ING-07T (VNIIA, Russia) was installed at Soreq NRC. A deuterium (D) beam (a minimum 15 mm diameter) irradiates a target containing tritium (T) and via the $\text{D} + \text{T} \rightarrow n + ^4\text{He}$ reaction generates neutrons at $E_n \approx 14.2$ MeV. The neutron generator ion source is of a penning ion source type and allows continuous beam operation. The neutron production can be regulated by the acceleration voltage and ion source parameters. The specified neutron emission rate is $1 \times 10^9$ n/sec. Neutrons are emitted almost isotropically in space. The advantage of the sealed-tube is that it eliminates the use of a vacuum pump. The neutron generator is contained in a 44 cm long aluminum cylinder with a diameter of 19 cm. The generator is connected by a 50 m long cable to a control unit (laptop) [1].

A known method [2] for calibration of a 14 MeV neutron flux is performed by copper activation, using the fast neutron reaction $^{63}\text{Cu}(n,2n)^{62}\text{Cu}$, having a neutron energy threshold of $E_n = 10.8$ MeV. $^{62}\text{Cu}$ decays by $\beta^+$ emission ($T_{1/2} = 9.67$ min; $P_{\beta^+} = 97.43\%$). Photons are produced from positron-electron annihilation (511 keV) at 194.86% of the disintegrations [3].

In this paper we present a series of measurements performed in order to calibrate the neutron emission rate of the D-T generator and investigate the profile distribution of the flux.

MATERIALS AND METHODS

The neutron flux hitting the copper target is calculated by $\phi = \frac{Q}{4\pi x^2}$ where $Q$ is the neutron emission rate (n/sec) calculated from:

$$Q = \frac{4\pi \lambda x^2 MCe^{\lambda t_d}}{N_d s P_{\gamma} m a (1 - e^{-\lambda t)})}$$

where $\lambda$ is the $^{62}\text{Cu}$ decay constant ($1.195 \times 10^{-3}$ sec$^{-1}$), $x$ (cm) is the distance between the neutron source (tritium target) of the generator and the copper target, $M$ is the Cu atomic weight (63.546 amu) [4], $C$ is the peak area (counts) at 511 keV, $t_d$ is the decay duration (s) between end of irradiation to start of measurement, $N_d$ is Avogadro constant, $s$ (counts/γ) is the HPGe (High Purity Germanium) detection efficiency (at 511 keV), $P_{\gamma}$ is the gamma abundance (194.86%), $m$ is the target mass (g), $a$ is the isotopic abundance of $^{63}\text{Cu}$ (69.17%) [4], $\sigma$ is the reaction cross section (4.7x10^{-25} cm$^2$) [5], $t_{irr}$ is the irradiation duration (s) and $RT$ is the real-time measurement (s). The emission rates were corrected for self-absorption in the copper targets (5.7% in a 1.5 mm thick copper disk).
The activity of $^{62}\text{Cu}$ was measured using a calibrated gamma spectrometry system based on a HPGe detector, shielded by 5 cm Hg and 10 cm Pb, with a relative efficiency of 60% (Ortec, USA) and a resolution of 1.90 keV both at 1332.5 keV.

The emission rate of the generator was measured by a set of irradiations of Cu targets (copper disks of 9 to 19.4 mm diameter and 0.1 to 1.5 mm thick) with varying HV (High Voltage) and IPenn (Penning ion source current) parameters as seen in Figure 1. The emission angular distribution was assayed by a number of Cu targets, placed at different distances and angles from the neutron source.

Figure 1. Left: The neutron generator inside a borated polyethylene housing; neutrons are emitted from the upper end side. Right (bottom): Four copper disks mounted at different angels and distances. Right (up): The neutron generator situated in a concrete room (1 m thick walls).

RESULTS AND DISCUSSION

The measured emission rates at varying HV and IPenn values are shown in Table 1.

In column 6 of Table 1, there are comments concerning an off-axis position to measure the angular distribution of the neutrons. Emission rates in Table 1 were calculated assuming a point source of neutrons and a point target of copper. At short distances this assumption is not valid and the emission rates calculated there deviate from the true values. This was the case with samples #1 and #2. Hence, the overall uncertainties at close targets are larger than the quoted values in column 5. The results of samples #8 and #12 are showing the true emission rate, i.e. $1.73 \times 10^9$ n/s at HV = 170 kV and IPenn = 140 (rel. units), as recommended by the supplier. At these settings, the emission rate is high and the generator life-time is not impaired.

Table 1 shows that the emission rate is strongly dependent on the ion source parameter IPenn, as can be clearly seen in Figure 2 that depicts results of the first group of measurements (HV = 150 kV).
Table 1. Neutron emission rate at different operating settings of the generator

<table>
<thead>
<tr>
<th>Sample ID</th>
<th>HV (kV)</th>
<th>IPenn [rel. unit]*</th>
<th>Q (n/s)</th>
<th>u (Q) (%)</th>
<th>Comments</th>
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<tr>
<td>C1</td>
<td>150</td>
<td>150</td>
<td>$1.09 \times 10^9$</td>
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<tr>
<td>C3</td>
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<td>150</td>
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<td>C4</td>
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<td>$6.60 \times 10^6$</td>
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<td>$7.59 \times 10^8$</td>
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<tr>
<td>C5-4</td>
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<td>#12</td>
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<td>2.3</td>
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<td>140</td>
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<td>2.0</td>
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</table>

*1 rel. unit ≈ 2.75 μA (average current between ion source electrodes)

The results of samples C5-1 to C5-4 (see Figure 1) are subject to large uncertainties (see Table 1) and therefore, they do not yield precise information on the angular distribution. In the 2nd group of measurements (170 kV), there were 4 samples in a configuration similar to Figure 1. The count-rates were corrected for decay (end of irradiation to beginning of measurement) and for sample mass. Figure 3 displays the corrected count-rate to the power $-\frac{1}{2}$ versus the distance. The straight line fitted to the experimental results shows that:

(a) the system behaves according to $1/x^2$
(b) the zero intercept shows that the source is at $x = 0$
(c) the assumption of point-source, point-target is valid

The outcome of this analysis is that neutrons are emitted isotropically at a flat angular profile.

A calculation of the relationship between the $^{62}$Cu activity and the distance between the neutron source (a tritium target of 60 mm diameter) and the copper disk, shows a dependence of $1/x^{1.8}$, which is very similar to $1/x^2$ as shown above.
CONCLUSIONS
A compact 14 MeV neutron generator is operational at Soreq NRC. The emission rate was measured by the Cu activation method and was found to be $1.73 \times 10^9$ n/s at HV = 170 kV and IPenn = 140 (rel. units). The emission of neutrons is isotropic. Potential applications of the generator, e.g., neutron radiography, neutron activation analysis, production of specific nuclides ($^6$He and $^8$Li [6]) are under consideration.

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1. ING-07T Neutron Generator, Operating manual, All-Russian Research Institute of Automatics, VNIIA, Russia.

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Absolute Calibration of a Cf252 Neutron Source Using the MnSO4 Bath Method

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Radiation Safety Division, Soreq NRC, Yavne, Israel

INTRODUCTION
Recently, a new $^{252}$Cf neutron source was installed at the Soreq NRC neutron calibration facility. Absolute calibration of the neutron emission rate of the source was carried out using the manganese sulfate bath method which is a well-known method for absolute determination of radionuclide neutron sources \(^{(1-4)}\). In this method the neutron source is placed at the center of a large spherical or cylindrical bath filled with a concentrated aqueous solution of MnSO$_4$. Fast neutrons emitted isotropically from the source will slow down in the hydrogenous medium and will be captured by the components of the aqueous solution, mainly by $^{55}$Mn atoms to produce radioactive $^{56}$Mn ($T_{1/2} = 2.5789$ h). Thus, from the measurement of $^{56}$Mn, combined with the neutron capture cross section and atoms concentrations, it is possible to determine the neutron emission rate ($B$) of the $^{252}$Cf neutron source:

$$B = \frac{1}{f} \times \frac{1}{F} \times R$$ \hspace{1cm} (1)

where $f$ is the fraction of neutrons captured by $^{55}$Mn, $R$ is the production rate of $^{56}$Mn by the neutron capture reaction, and $F$ is a coefficient factor which contains e.g., the fraction of neutrons that escape from the bath and the fraction of neutrons self-absorbed in the source.

$f$ Can be expressed as:

$$f = \frac{N_{Mn} \cdot \sigma_{Mn}}{N_{Mn} \cdot \sigma_{Mn} + N_{H} \cdot \sigma_{H} + N_{S} \cdot \sigma_{S} + N_{O} \cdot \sigma_{O}}$$ \hspace{1cm} (2)

where $N_{Mn}, N_{H}, N_{S}$ and $N_{O}$ are the number of nuclei of manganese, hydrogen, sulfur and oxygen in the solution, respectively and $\sigma_{Mn}, \sigma_{H}, \sigma_{S}$ and $\sigma_{O}$ are their thermal neutron capture cross sections, 13.3, 0.33, 0.53 and 0.00019 barn, respectively.

EXPERIMENTAL
A cylindrical bath with an inner diameter of 100 cm and height of 120 cm was installed at the Soreq neutron calibration facility. The schematics of the bath and system components are shown in Figure 1. The 785 liter bath was filled with an aqueous solution of highly pure manganese sulfate monohydrate. The mass concentration of MnSO$_4$ in the solution was 0.173% and the solution density was 1.002 g/cm$^3$. The aqueous solution was stirred for 30 minutes before the beginning of the experiment. The rather low concentration of MnSO$_4$ was prepared because of the high emission rate of the $^{252}$Cf neutron source and the predicted activity of $^{56}$Mn to be sampled and measured after the irradiation phase (see below).
The $^{252}$Cf neutron source was transferred from its storage cask, using a pulley system, towards a guide funnel to the geometrical center of the bath, at the bottom of a stainless steel sleeve. After an overnight irradiation (17.3 hours), the source was removed and the solution stirred for about 30 min to achieve homogeneity of the $^{56}$Mn atoms in the aqueous solution. Several 200 g samples of the solution were taken and immediately measured by high resolution gamma spectrometry. The gamma spectrometry system consists of a shielded High Purity Germanium (HPGe) detector (Ortec, USA) with a relative efficiency of 60% and an energy resolution of 1.9 keV, both at 1332.5 keV. The data was analyzed using Genie 2000 v 3.1 (Canberra, USA). $^{56}$Mn was identified and quantified by its gamma rays at energies of 846.8, 1810.8, and 2113.1 keV, of abundances 98.9, 27.2, and 14.3%, respectively.

**RESULTS**

Input parameters used in computing $^{252}$Cf neutron emission rate were:

- Fraction of neutron leakage (escape) from the Mn bath taken from literature is 1\%\(^{(2,4)}\). To verify this value we preformed Monte Carlo simulations (MCNP code version 4C) for a $^{252}$Cf source which was simulated as a point source and the energy spectrum of its spontaneous fission neutrons was approximated by the expression\(^{(5)}\):

\[
\chi(E) = E^{1/2} \cdot \exp^{-0.776E}
\]

- Fraction of neutrons self-absorbed in the source was taken from literature to be 0.1\%\(^{(4)}\).
- The ratio of hydrogen, oxygen and sulfur number of nuclei to manganese nuclei was calculated by the solution concentration and found to be $9.677 \times 10^3$, $4.842 \times 10^3$ and 1, respectively.
The measured $^{56}$Mn activity concentration was $3341.6 \pm 1.02\%$ Bq/liter. The calculated emission rate of the $^{252}$Cf source was $6.40 \times 10^8$ n/sec $\pm 5\%$ on 1/11/2007. For computing the decay of $^{252}$Cf we use the half-life 2.645 y.

CONCLUSIONS
An absolute calibration of the $^{252}$Cf neutron emission rate was carried out at Soreq NRC neutron calibration facility using the manganese sulfate bath method. The emission rate was found to be $6.40 \times 10^8$ n/sec $\pm 5\%$ within supplier specifications. This emission rate corresponds to a $^{252}$Cf mass of 278.3 $\mu$g $^{(5)}$.

ACKNOWLEDGEMENT
The authors would like to thank Mr. Alon Tal from the Electro Optics Division at Soreq NRC for the mechanical design, assembly of the bath system and enthusiastic support.

REFERENCES

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The Haling Power Depletion For Nuclear Core Analysis Including Genetic Algorithm Optimization Methods

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INTRODUCTION
The Haling Power Depletion (HPD)\(^1\) is an iteration method that begins with an initial guess of the power distribution at the beginning-of-cycle (BOC). It then depletes the core in one step and compares the BOC power distribution with the end of cycle (EOC) calculated power distribution. A recursion expression modifies the guess of the power distribution at BOC to be closer to the EOC power distribution. The iteration continues until the BOC and EOC power distributions are the same. The principle states that “for any given set of end-of-cycle conditions, the maximum normalized power (NP) is maintained at the minimum value because the power shape does not change during the operation cycle”. The NP is the ratio of the fuel assembly (FA) power to the average FA power in the core. The HPD calculation is performed only at EOC and is presently the fastest method for determining the end-of-cycle characteristics. There are two Golden Rules for the HPD applications; 1) Calculation with the HPD must use its EOC results when comparing with the accurate burnable poison (BP) loaded core calculations and 2) No component or other structures, e.g. control rods, can be moved in or out of the core during the HPD calculation. The HPD has been used to greatly accelerate the loading pattern (LP) optimization calculations for pressurized water reactors (PWR). This is especially true when optimizing with Genetic Algorithms (GA)\(^{2,3,4}\). The HPD deals only with the arrangement of FAs or Loading Pattern design (LP) leaving the placement of BPs for a subsequent step.

A long cooperative program for Penn State was begun with the GPU nuclear engineering staff in designing the reload patterns for the TMI-1 core (B&W design) starting in the early 1970’s. Techniques were then developed to improve the priority scheme\(^4\) using the HPD method to optimize loading the FAs into the core, which was followed by placing the BPs into the core in a manner that resulted in attaining many successful reloads\(^4\). During the early 1990’s, GPU decided to go to a two year cycle. PSU helped the GPU nuclear engineering staff and B&W develop the core design for the first two year cycle. The priority scheme using the HPD method created the 2 year cycle LP and then the BPs were successfully designed to prevent any safety constraint from being violated during the depletion calculation\(^4\). The same procedure was used to successfully reload the Beaver Valley reactors (Westinghouse design).

In 2001, the SKODA JS group headed by J. Svarny began a cooperative work with Penn State to obtain an optimization technique for loading the VVER-1000 core.
The first procedure was to prove that the HPD method followed by the BP placement method was valid for the VVER-1000 core. Once this was achieved we developed an initial priority scheme and developed a few excellent reload designs for the second and third cycles. It is our understanding that they continue to successfully use this method (4).

However, Sun, et. al. (5) examined if the HPD was true or false by comparing power distributions during the cycle. Control elements were used during the cycle and determine the HPD was false. Their results are expected because their study violated both HPD Golden Rules. The HPD is valid only at EOC and control elements cannot be involved during the calculated HPD cycle (2 Golden Rules). Penn State studies showed that for the TMI-1, the Beaver Valley, and the VVER-1000 reactors (3) 1) The burnup distribution and FA power distribution obtained with the Regular Depletion (RD) method at EOC state are close to the HPD burnup and FA power distributions. (See Figure 1 and 2), 2) The hottest FA in the core for the HPD is essentially the same as for the actual core depletion with BPs, and 3) When a good BP design is placed in the core, the HPD peak NP is approximately 3 to 5% lower than the peak NP during the actual depletion cycle with the BPs. Thus, it was necessary to design the maximum HPD NP 3 to 5% lower than the allowed maximum NP.

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<td>6.40%</td>
<td>2.56%</td>
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Fig. 1. A Comparison of the HPD Method NPs and the Peak NPs of the RD Method
The HPD has been effectively applied as a filter in the GARCO genetic algorithm (GA) code developed at Penn State. The GARCO (Genetic Algorithm Reactor Core Optimization)\(^{(2,3)}\) is an innovative GA code that optimizes the reload of PWR's. This code is applicable for all types of PWR cores having different geometries and designs with an unlimited number of FA types in the inventory. GARCO has three modes: the user optimizes the core configuration (LP pattern) without BPs in the first mode; the second mode is the optimization of BP placement in an optimum LP chosen from mode 1, and mode 3 optimizes the LP and BP placements simultaneously. Different selection, crossover, and mutation GA operators are employed in GARCO. Mode 1 incorporates a new Age process and the priority system (worth technique) as probability tables into GARCO to accelerate an improved final solution. Figure 3 shows mode 1 results as the worth method and the age method are added for the VVER-1000. Mode 3 applies the simultaneous optimization in GARCO using methods similar to those in modes 1 and 2. However, the calculations are not sequential and simulate better the non-linear nature of reload design process. The goal in mode 3 is the same as in modes 1 and 2, namely, to find the BP placements in a selected LP to obtain the longest cycle length while satisfying all safety constraints. The accurate depletion calculation of the LP with BPs is done with the coupled lattice and reactor physics CASMO-4/SIMULATE3 package. However, the fact that these codes validate the safety of the core with the added BP placement design also validates the use of the HPD method. The calculations are applied to the TMI-1 core as an example PWR providing concrete results.

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![Location Number](29)  
RD Burnup (EOC)  
HPD Burnup  
Difference

Fig. 2. A Comparison of EOC Burnup Distributions of the HPD and RD Methods
RESULTS
The LP and BP optimizations for the TMI-1 reactor were tested successfully sequentially with mode 1 and mode 2 in GARCO and for VVER-1000 reactor in mode 1. It was proved that the HPD method is very important in quickly selecting the suitable reference core for separate BP optimization calculations and to create the initial population for the LP and BP mode 3 simultaneous calculation.

![Graph showing the best fitness results for the VVER-1000](image)

Figure 3  Comparison of the Best Fitness Results for the VVER-1000

LP and BP optimization. Such data will be presented at the meeting. LPs having maximum HPD NP of 1.38 or an NP standard deviation less than 0.305 were utilized for BP placement. The reference BP configurations for the TMI-1 FAs incorporated in this study are taken from the cross section libraries produced at Penn State\(^6\). Both the Age method and the worth method (use of probability tables) greatly improved the LP solution in Mode 1 and Mode 3 producing the highest fitness or optimum LP.

The mode1 - mode2 sequence best final design which, satisfied all safety constraints, had the highest EOC boron concentration of 139.0 ppm. The results from mode 3 slightly improved the TMI-1 results having an EOC boron concentration of 144.0 ppm. For both cases, mode2 and mode 3, 300 generations are created each taking approximately 46.4 hours. However, the best LP for the second case shown in Figure 8 is found after 26.9 hours.
CONCLUSIONS
The HPD method is valid for analyzing LPs for PWRs provided the two Golden Rules are followed wherein only the cycle length and the maximum NP are determined. It is essential that filters, basic reactor physics, e.g., the worth technique, and other fast and shortcut methods be incorporated into GA’s to enable the solution to be determined within reasonable computer times.

REFERENCES

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The 24th Conference of the Nuclear Societies in Israel
A Safe Solution to World Energy Supply - the Very High Temperature PB Reactor

David Saphier

DSNP, SNRC, Rehovot, Israel

INTRODUCTION

For the energy hungry world there is a solution which has the potential to resolve most of the present energy needs, with almost zero pollution and high thermal efficiency. The Very High Temperature Reactor (VHTR) can produce Hydrogen for automotive needs to replace the polluting gas and oil; it can produce electricity at very high efficiency with almost no pollution, and provide clean process heat for the industry and the energy needed for desalination plants to provide fresh water.

In the present study it is shown that choosing the Pebble Bed (PB) concept for the VHTR is not only a very effective way to supply all the energy needs, it is also one of the safest nuclear reactor concepts\(^1\). Depending on the fuel cycle chosen, it is possible to reduce significantly the TRU waste\(^2\) normally produced in light water reactors and thus further reduce the environmental concerns of long living FP.

A conceptual 600MWt High Temperature Pebble Bed\(^3\) reactor is proposed, and its safety characteristics are analyzed by simulating various hypothetical accidents, using the DSNP\(^4\) simulation system.

VHTR GEN-IV REACTORS

Various Generation IV reactor concepts are being studied world-wide with the objective to develop inherently safe, pollution free, superior economics, sustainable and proliferation resistance energy supply for the 21\(^{st}\) century and beyond. The countries participating in the VHTR development project include: US, UK, Japan, Russia, China, France, South-Africa, Germany and Switzerland.

Superior economics can be obtained by using very high temperatures; hence the various VHTR concepts have coolant exit temperatures at or above 1000°C. The various designs include the Fast Breeder Reactor (FBR) either liquid metal or gas cooled, the Lead-cooled Fast Reactor (LFR), the Molten Salt Reactor (MSR), and the gas cooled graphite moderated reactor using either prismatic fuel, or spherical fuel elements – the pebble bed reactor. The reactors using Helium as coolant can reach very high temperatures, particularly in combination with graphite. Lead and Molten salt reactors have boiling points above 1400°C which enables them also work with coolant temperatures above 1000°C. The safety merits of PB-VHTR concept is the subject of the present study. One of the major advantages of the PB-VHTR is that its construction can be based on proven technology. Several countries have\(^5\) or are operating pebble-bed reactors with exit temperatures of up to 950°C. Fig1 demonstrates the superior thermal efficiency of the HTRGR compared to the presently operating light water reactors. The superior safety characteristics of the PB-VHTR are assured by three major factors. Large thermal inertia—very slow transients, single phase coolant – no phase change during power excursions, large negative feedback-mitigating any accident and limiting the temperature increase.
CONCEPTUAL DESIGN OF THE PEBBLE BED HIGH TEMPERATURE REACTOR

A conceptual design of the Pebble-bed High-temperature Gas-cooled power plant is shown in figure 2. The PB core heats up the He gas to 1000°C or above. The He pressure is 60 bars. The gas flows inside a concentric hot duct (preheating the incoming cold He gas) through a series of heat exchangers, in each giving up part of the heat. Optionally, the hot He is transported directly to the production plants.

Fig. 2: Schematic description of a high-temperature pebble-bed power plant.
The energy between 960°C and 540°C is used for hydrogen production in one of the chemical processes being developed for this purpose. The energy between 540 and 350°C is used to drive the turbine-generator to produce electricity. This part of the energy can be driven either through a steam generator producing high pressure steam to drive a conventional steam turbine, or using the direct cycle to drive a gas turbine. Finally the energy between 350 to 250°C is used via appropriate heat exchanges as industrial process heat or to drive a desalination plant.

The fuel used in the pebble bed core is made up of 6cm graphite spheres in which the TRISO fuel particles are embedded as shown in figure 3. The TRISO fuel particle includes UO₂ fuel kernels surrounded by four layers of protective coatings, including two pyrolytic carbon layers and a silicon carbide layer, to improve the fission product retention capability of the fuel even at very high temperatures. The fuel enrichment is about 10%, but depends on the particular fuel cycle chosen. The particles are dispersed in a graphite matrix sphere of 5cm, which is then surrounded by a denser graphite shell of 0.5cm. The TRISO particles can achieve very high burn-ups, and experimental irradiations have shown that up to 747 Gwd/t, no detrimental effects were observed.

THE COMPUTATIONAL MODEL

The DSNP⁴ simulation package was used to study the proposed PB VHTR response to various accident conditions. A schematic description of the core and its various enclosures are shown in figure 4. Each of the shown elements is represented by a two dimensional cylindrical DSNP module surrounded by a one-dimensional cylindrical axial flow path.

![Fig. 3: A spherical graphite fuel element for the pebble bed reactor.](image)

![Fig. 4: Schematic representation of the High Temperature Pebble-bed Reactor.](image)
The core is represented by a cylindrical element with axial and radial subdivisions, and each cylindrical segment contains a spherical fuel element, the coolant flow is from top to bottom. The radial reflector which is cooled by incoming cold He also isolates the steel pressure vessel from the hot He. From the outside the pressure vessel is cooled by natural air flow. The reactor is enclosed in a concrete silo which is water cooled on its inner side by a special liner. The silo is cooled by the ambient air.
In addition to the reactor cylindrical elements and the flow hydraulics, the various heat exchangers, the pipes and valves, the steam generator, turbine and other component models were included in this simulation. The core neutronics was represented by the kinetic equations and the various control and safety features were also included.

**RESULTS OF THE SIMULATIONS**
The hypothetical accidents investigated in the present study include loss of flow as a result of a breach in the main coolant duct, which also includes depressurization of the core and a loss of heat sink by failure of one of the thermal loads, the turbine, the H2 production plant or the plant using the process heat.
A small part of the results are reproduced in figures 5 to 8. Figures 5 and 6 show the loss of coolant accident due to a break in the cold duct. As the safety system detects a change in pressure the reactor is shut down. Figure 5 shows the He transient temperatures in the core hottest region for the 6 axial segments, while Fig. 6 shows the fuel sphere center temperature for the same region. The maximum coolant temperature is 1360°C, while the fuel temperatures are a few degrees higher. Figures 7 and 8 show the same accident but includes the malfunction of the safety system, i.e. no scram. As can be observed, in all cases the power is reduced, either by the safety system or by the negative feedback, and no damage to the fuel will occur. The decay heat is dissipated by radiation and by natural convection.

CONCLUSIONS
From the simulations of various hypothetical accidents of the PB-VHTR it is obvious that this reactor is safer than any presently known reactor concept. It can generate fuel, electricity and process heat cover all energy needs by a single system. Further design studies are needed to resolve economic issues and technological concerns of the very high temperatures.

REFERENCES

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Verification of Decay Heat Calculation Capabilities of BGCore System

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An accurate prediction of total decay heat and its time dependence is essential for determining the heat removal requirements after reactor shutdown, during spent fuel transportation and storage, and in the reactor safety analyses for such accidents as Loss of Coolant (LOCA).

The amount of the decay heat generated by UO2 fuel after normal or emergency shutdown in the existing Light Water Reactors can be predicted with high degree of accuracy on the basis of computer simulations and extensive experimental data\(^1\). However, the amount of decay heat in future generation of nuclear reactors or in existing reactors operating with innovative fuel cycles has much greater uncertainty for the following reasons:

- Alternative to UO2 fuel compositions would have different fission product (FP) yields because they originate from different fissionable nuclides.
- Fission product yields are different in fast and thermal neutron spectra.
- Existing computer codes, capable of predicting the decay power, have at times outdated database for the FP decay constants, FP yields, and recoverable decay Q-values.
- The choice of nuclides included in the existing computer codes is based either on the nuclides' neutronic importance during irradiation or their out-of-core characteristics (radio-toxicity, decay power etc.) but never both.
- No experimental data exists to confirm the computer simulation predictions.

BGCore reactor analysis system, recently developed at Ben-Gurion University for calculating in-core fuel composition and spent fuel emissions following discharge, couples the Monte Carlo neutronic solver (MCNP\(^2\)) with SARAF burnup and decay module. The BGCore approach for generation of one group cross-sections\(^3\) takes advantage of the fact that dividing the neutron flux tally into multiple energy bins has practically no effect on the MCNP execution time. According to this approach, only multi-group neutron spectrum is calculated by MCNP, while reaction rates are not tallied but calculated in a separate subroutine using pre-generated multi-group cross-section set and the fine group neutron spectrum obtained from MCNP. This procedure reduces dramatically the computation time of MCNP. The reduction in execution time by factor of 3 to 10 between the standard and multi-group approaches was observed depending on the nature and complexity of the problem. Additional advantage of the method is the fact that the one group cross-sections are calculated for all available isotopes and, in principle, for all available reaction types without any increase in computation time. Currently, about 1700 isotopes are tracked in the SARAF code. The following guidelines were used in choosing these nuclides:

- All the nuclides that have evaluated cross-sections in JEFF3.1 file with there respective decay chains.
- All the nuclides that have radio-toxicity (Sv/Bq) coefficients in the most recent IAEA Basic Safety Standards issue (BSS-115) with their respective decay chains.
- All nuclides that can be generated by fission (have fission yield data) with their respective decay chains.
The fact that all of the isotopes, and not just the most neutronically important, are tracked throughout all depletion steps, allows calculations of post-irradiation fuel characteristics prediction such as, activity, radiotoxicity, and decay heat.

In this work, we tested decay heat calculation capabilities of BGCore system. We calculated the decay power produced by 4.2% enriched UO2 fuel in a typical PWR core using the BGCore computer code. The obtained results were compared (Fig. 1) with those of the widely-used ORIGEN-2(4) depletion code as well as with the most recent edition of ANS Standard UO2 fuel decay power curve(1). Figure 1 shows very good agreement between both codes and ANS standard curve. The ANS standard curve is based on conservative calculations and experimental data. Therefore, the ANS standard data slightly overestimates the decay heat during first few hours. However, the difference between ANS standard data and that predicted by the BGCore does not exceed 5%.

The obtained results indicate that the BGCore system can be used with confidence for prediction of decay heat of innovative reactor systems.

![Figure 1. Decay heat comparison](image-url)

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The authors acknowledge the help of Dr. G. Raitses of Brookhaven National Laboratory for performing the ORIGEN calculations.

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Seed-Blanket Fuel Concept for Pebble Bed Reactors

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INTRODUCTION
This work explores the basic feasibility of the Seed-Blanket Unit (SBU) fuel concept for Pebble-Bed gas cooled reactors. The primary goal is to reduce Pu production and to change the Pu isotopic composition in the discharged fuel, thus improving proliferation resistance to the fuel cycle. Additional goal is to improve the fuel utilization and, by that, reduce the Uranium ore requirements. The main idea of SBU concept is to use Th as primary fertile material instead of $^{238}\text{U}$. The main advantage of using thorium is that $^{233}\text{U}$, generated as a result of neutron captures in Th, is a superior fissile material. Additionally, using Th as a primary fertile component reduces dramatically the amount of generated Pu and minor actinides improving the fuel proliferation resistance and waste characteristics. These goals can be achieved if Uranium (seed) and Th (blanket) parts of the fuel are spatially separated allowing independent fuel management and design optimization of each part.

It was shown that applied to PWRs, SBU concept reduces the rate of plutonium production by a factor of 3 to 5 and waste volume by up to 50%. Hence, it reduces considerably the proliferation potential of the fuel cycle and its environmental consequences.

However, the feasibility of SBU fuel in PWRs is debatable primarily due to the extended burn-up of the seed fuel and the core residence time of the Blanket fuel well beyond current LWR fuel experience. All these problems can be eliminated if the SBU concept is applied to the gas-cooled Pebble Bed Modular Reactors (PBMR) currently under development in South Africa. The advanced TRISO coated particles fuel provides extremely high radiation, chemical and mechanical stability, so that the extended burn-up and core residence time are no longer an obstacle for the PBMRs.

Therefore, PBMR 400MW reactor core was used as a reference reactor design in this study. The analysis was primarily focused on the optimization of fuel management scheme, core geometry and fuel composition in order to achieve maximum uranium savings and reduce Pu production rate. The parameters considered were thorium residence time, radial positions of pebbles and various relative volume fractions of seed and blanket regions.

METHODOLOGY
Most of the reactor physics calculations, presented in this paper were carried out with the VSOP-94(2) computer code system, which was developed especially for simulations of Gas Cooled Reactor (GCR). VSOP is a system of well proven computer codes linked together for a full simulation of gas cooled reactors: cross-sections processing, neutron spectrum calculations, diffusion calculations, burn-up, fuel management, cost analysis and steady-state or transient thermal hydraulics analysis. These calculations were adapted for treating the special features of the Pebble-Bed gas cooled reactors.
A model of PBMR core was also created for the MCNP\(^{(3)}\) computer code in order to verify the results calculated with VSOP. A good agreement was obtained between the two codes suggesting that VSOP produces results with the accuracy sufficient for the purposes of this study.

The main parameters investigated were Natural Uranium (NU) utilization, SWU utilization, as well as production rate and quality of Pu in the discharged fuel as a measure of proliferation resistance of the fuel cycle.

The SBU fuel concept was applied to the design of the reference PBMR-400\(^{(4)}\) core using two spatially separated fuel types. The seed (uranium only) is enriched to 20%. The blanket is thorium with a small amount of uranium to maintain the power of the blanket pebbles and isotopically dilute the generated U235 below non proliferation enrichment limit of 12%. The seed pebbles discharged from the core still have relatively high enrichment of \(\sim 8\%\). Therefore, the discharged seed pebbles are re-inserted into an All-U core to produce additional power, thus improving NU and SWU utilization.

The optimized SBU core performance parameters were compared to those of the reference case.

**RESULTS**

Thorium residence time is defined by the number of passes that a pebble makes through the core. In order to improve the NU three models were carried out - 6, 12, 18 and 24 passes. As can be observed from Figure 1, the NU utilization peaks if Th pebbles pass through the core about 18 times.

![Figure 1: Fuel utilization vs. number of blanket passes through the core](image)

In the study aiming at optimization of the radial locations of the Seed and Blanket regions, five different channel configurations were compared. For all cases, the volume fraction of seed and blanket pebbles in the core was the same (50% Seed - 50% Blanket). The uranium utilization is improved if the power produced by the Blanket region is maximized. In this case, the power peaking in the core is also reduced, which is highly desirable from the thermal-hydraulic design and safety perspective. Table 1 summarizes the results obtained for the different seed and blanket channels locations in the core.
The performed analysis suggests that the most favorable configuration is where the blanket is adjacent to both inner and outer reflectors (Case 4 in Table 1). This configuration maximizes the number of thermal neutrons that are absorbed in the blanket region because of the additional moderation in the graphite reflectors.

Figure 2 shows the axial power density profile in the blanket region. The results, shown in Figure 2, support the fact that the most effective power production is obtained in the configuration, where thorium is adjacent to the reflectors.

<table>
<thead>
<tr>
<th>Case#</th>
<th>Configuration</th>
<th># of U passes</th>
<th># of Th passes</th>
<th>Power share U/Th</th>
<th>Power peak</th>
<th>K-inf at discharge</th>
<th>Enrichment of seed at discharge</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>BSBS</td>
<td>4</td>
<td>12</td>
<td>70/30</td>
<td>4.04</td>
<td>1.03</td>
<td>0.95</td>
</tr>
<tr>
<td>2</td>
<td>SSBB</td>
<td>4</td>
<td>12</td>
<td>79/21</td>
<td>4.83</td>
<td>0.98</td>
<td>0.86</td>
</tr>
<tr>
<td>3</td>
<td>SSBBBB</td>
<td>4</td>
<td>12</td>
<td>71/29</td>
<td>4.66</td>
<td>1.00</td>
<td>0.87</td>
</tr>
<tr>
<td>4</td>
<td>BSSB</td>
<td>6</td>
<td>18</td>
<td>67/33</td>
<td>3.28</td>
<td>1.08</td>
<td>0.90</td>
</tr>
<tr>
<td>5</td>
<td>BSBSB</td>
<td>6</td>
<td>18</td>
<td>69/31</td>
<td>3.76</td>
<td>1.03</td>
<td>0.96</td>
</tr>
</tbody>
</table>

In the additional optimization stage, two cases were investigated: SBU core with volume fraction 60% blanket and 40% seed, and 70% blanket and 30% seed, instead of equal division as in the previous cases. The amount of fissile material in the core was unchanged. The SBU core used in both cases was divided into four channels: two for seed and two for blanket. The results are summarized in Tables 2 and 3. The results show that there is no particular improvement of fuel utilization. Increase of the volume of the blanket in the core, requires faster seed pebbles feed rate, and thus, higher natural uranium requirements. Plutonium production rate increases, but not significantly. In summary, the increase of the number of blanket pebbles in the core was found to be not an efficient strategy for improvement of NU utilization. The 50% Seed – 50% Blanket division appears to be more favorable.
SUMMARY AND CONCLUSIONS
This work focused on application of the Seed-Blanket fuel concept to the fuel cycle of PBMR reactor, which is currently under construction in South Africa. By substituting some of the U238 in the core with Thorium, the Seed-Blanket concept has a potential to add some attractive features to PBMR such as greatly reduced Pu production and savings of natural uranium. In contrast to LWRs, PBMR-type reactors offer more flexibility with respect to the Seed-Blanket core design optimization because of the less restrictive fuel residence time, burnup, as well as reshuffling scheme and reshuffling frequency. A comprehensive analysis of Seed-Blanket PBMR core was performed aiming at the reduction of natural uranium requirements and Pu production rate. The considered optimization parameters were Th blanket residence time, number of fuel passes through the core, location and dimensions of the Seed and Blanket regions, and volumetric fractions of Th, enriched Uranium and Graphite in the core. The most attractive combination of the optimized parameters results in a reduction of Pu generation rate by almost a factor of two and natural uranium savings of about 10%.

### Table 2: Equilibrium cycle

<table>
<thead>
<tr>
<th>Case</th>
<th>Reference</th>
<th>SBU 60/40</th>
<th>Secondary core 60/40</th>
<th>SBU 70/30</th>
<th>Secondary core 70/30</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Seed</td>
<td>Bl.</td>
<td>Fresh</td>
<td>Burnt</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Power Peak, MAX/AV</td>
<td>3.22</td>
<td>3.71</td>
<td>3.04</td>
<td>4.92</td>
<td>2.78</td>
</tr>
<tr>
<td>Number of passes per pebble</td>
<td>6</td>
<td>6</td>
<td>18</td>
<td>9</td>
<td>6</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pebbles injection rate, #/day:</td>
<td>485</td>
<td>243</td>
<td>121</td>
<td>243</td>
<td>243</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Discharge burnup, MWD/kg</td>
<td>91</td>
<td>121</td>
<td>102</td>
<td>166</td>
<td>58</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pebble Residence Time, days</td>
<td>921</td>
<td>744</td>
<td>2232</td>
<td>1116</td>
<td>744</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>U²³⁵ annual feed, kgU²³⁵/year</td>
<td>157</td>
<td>177</td>
<td>6.7</td>
<td>106</td>
<td>6</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Seed enrichment at discharge</td>
<td>2.11</td>
<td>8.10</td>
<td>-</td>
<td>3.83</td>
<td>3.38</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Seed k-inf at discharge</td>
<td>0.91</td>
<td>1.05</td>
<td>0.92</td>
<td>1.03</td>
<td>0.87</td>
</tr>
<tr>
<td></td>
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<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

### Table 3: Equilibrium cycle data per core

<table>
<thead>
<tr>
<th>Case</th>
<th>Reference</th>
<th>SBU 50/50</th>
<th>SBU 60/40</th>
<th>SBU 70/30</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel Utilization, MWDₜₚ/kgNT</td>
<td>3.87</td>
<td>4.24</td>
<td>4.15</td>
<td>3.74</td>
</tr>
<tr>
<td>SWU Utilization, MWdₜₚ/kg-SWU</td>
<td>5.29</td>
<td>5.32</td>
<td>5.20</td>
<td>4.68</td>
</tr>
<tr>
<td>Fissile U²³⁵ annual feed, kgU²³⁵/year</td>
<td>314</td>
<td>284</td>
<td>290</td>
<td>322</td>
</tr>
<tr>
<td>Total Pu, kg/GWYe</td>
<td>131</td>
<td>72.5</td>
<td>79</td>
<td>82</td>
</tr>
<tr>
<td>U²³³, kg/GWYe</td>
<td>-</td>
<td>17.5</td>
<td>19</td>
<td>18</td>
</tr>
</tbody>
</table>
REFERENCES

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High Conversion Thorium Fuel Cycle for PWRs

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INTRODUCTION

Commercial production of nuclear power was pioneered in 1957 at Shippingport, Pa. by a prototype Pressurized Water Reactor (PWR). Before the decommissioning of Shippingport, the last core was converted into a Light Water Breeder Reactor (LWBR). The main goal of the LWBR program was to prove that a sustainable closed fuel cycle based on Thorium ($^{232}$Th) and reprocessed uranium ($^{233}$U) feed is possible without the need to develop an advanced and costly fast reactor technology. Successful operation and results of numerous experiments that followed the Shippingport core confirmed that the reactor achieved Breeding Ratio (BR) higher than unity. Since then, the worldwide interest in closed fuel cycles has declined, mainly because of abundant Uranium reserves and fear of nuclear proliferation.

Theoretically, the only practical combination that is capable of self-sustainable breeding in the thermal spectrum is Th-$^{233}$U. The interest in Th as a possible nuclear fuel remained high due to its favorable properties, which would allow existing reactors to substantially reduce their production of Plutonium and their volume of spent fuel. Furthermore, Thorium, being at least three times more abundant than Uranium and used in reactors with high conversion ratio, would greatly extend the available energy resources and reduce the dependence on fossil fuels.

Recently, the development of sustainable energy has become one of the top priorities of the industrialized world. Moreover, some of the countries with vast Th reserves, but no reactors, are debating initiation of a nuclear power program. These developments suggest that a Th-based fuel cycle should be reconsidered with a shift in the design objectives, where savings in natural resources becomes important, in addition to non-proliferation and waste issues.

The objective of this work is to examine the possibility of using the Th closed fuel cycle in existing reactors, and to approach the performance of the Shippingport LWBR (BR close to 1) through moderate changes in the design of current generation of PWRs.

Clearly, the use of movable fuel for reactivity control, as employed in LWBR, is not an option nowadays. However, high conversion ratio can be achieved through the use of advanced materials, heterogeneous reactor core structure, optimization of fuel composition and other innovative features. Any of these modifications should not compromise the reactor safety under normal conditions.

This study outlines some effective strategies to achieve (or to approach as closely as possible) a sustainable fuel cycle, in which the only feed is Th.
WORK DESCRIPTION

All neutronic calculations in this study were performed with fuel assembly transport and burnup BOXER\(^{(1)}\) code, which is a modular code for two-dimensional calculation of LWR fuel assemblies.

**Reference fuel assembly**

First, we performed calculations of homogeneously mixed ThO\(_2\)-U\(_{233}\) fuel assuming typical PWR fuel assembly geometry and standard PWR operating conditions. The fuel cycle length was evaluated by applying the Linear Reactivity Model (LRM)\(^{(2)}\) to the results of fuel cell burnup calculations. U\(_{233}\) content was adjusted to achieve fuel cycle length of 18 months considering a 3-batch reloading scheme and 3% leakage reactivity, and was found to be equal to 3.8\(^{\%}\)/o. The conversion ratio (CR) was calculated by using the following relation:

\[
CR = \frac{RR_{c}(^{232}Th) - RR_{c}(^{233}Pa) + RR_{c}(^{234}U)}{RR_{a}(^{233}U) + RR_{a}(^{235}U)}
\]

Where RR\(_c\) is the macroscopic capture reaction rate and RR\(_a\) is the macroscopic absorption reaction rate. The life average conversion ratio for the homogeneous ThO\(_2\)-U\(_{233}\) fuel assembly was found to be equal to 0.7.

**Seed-blanket heterogeneous fuel assembly**

In order to increase the conversion ratio, it is necessary to increase capture reaction rates in the fertile material (Th). This goal can be achieved by utilizing of a heterogeneous “seed and blanket” fuel assembly geometry, in which the assembly is divided into a U\(_{233}\) rich “seed” region, and mostly a Th\(_{232}\) “blanket” region. In such a configuration, the supercritical seed works as a neutron supplier, while the subcritical blanket acts as U\(_{233}\) breeder. Only modest efforts were made to optimize the dimensions of the seed and blanket unit cell geometries. The nearly optimal dimensions of the seed and blanket regions were assumed to be in order of one neutron migration length for each region.

We performed a series of ThO\(_2\)-U\(_{233}\) unit cell calculations, in order to determine neutron migration lengths. The calculations were performed with the U\(_{233}\) content varying from 3.5\(^{\%}\)/o to 20\(^{\%}\)/o, and the hydrogen to heavy metal ratio (H/HM) varying from 2.5 to 7.7 (Fig. 1). Figure 1 shows that the migration length exhibits strong dependence on H/HM ratio, while varying only slightly with U\(_{233}\) content. For typical PWR conditions, the migration length in both seed and blanket regions is about 8.0 cm.

The migration length calculations suggested that the seed-blanket assembly configuration should be 12×12 fuel pins with standard PWR unit cell dimensions. The layout of the assembly is shown in Fig. 2. For this assembly, we performed a set of burnup calculations with blanket U\(_{233}\) content varying from 0.0\(^{\%}\)/o to 3.8\(^{\%}\)/o, while keeping the whole assembly U\(_{233}\) content constant and equal to that of the reference homogeneous ThO\(_2\)-U\(_{233}\) case. The results of the calculations, presented in Figs. 3 and 4, show that the CR decreases as the blanket U\(_{233}\) content increases. However, reduction in U\(_{233}\) content in the blanket, reduces k-inf of the assembly and increases the power peaking. It should be noted that only cases that are supercritical at BOC are of interest. For these cases the maximum achievable lifetime average CR is about 0.8. However, if a multibatch reloading scheme is employed, this restriction is less relevant.
Figure 1. Migration length vs. $^{233}$U content for different H/HM ratios

Figure 2. South-East quarter of 12×12 seed-blanket fuel assembly

Figure 3. Assembly k-inf vs. irradiation time for different seed $^{233}$U contents
Figure 4. CR vs. burnup for different seed $^{233}$U contents

In order to further improve the CR, we performed additional modification of the seed-blanket assembly by increasing the radius of the blanket fuel pellet from 0.4095 cm to 0.55 cm. The larger fuel pin dimensions reduce H/HM ratio, harden blanket neutron spectrum and, thus, increase resonance absorption in thorium. We repeated the burnup calculations for modified seed-blanket assembly while preserving $^{233}$U to $^{232}$Th mass ratio as in the reference case (Figs. 5 and 6). The results of the calculations exhibit the same trend of reduction in CR as the $^{233}$U content in the blanket increases. The maximum attainable whole life CR for modified seed-blanket assembly is slightly above 0.9.

CONCLUSIONS

In this work, we investigated a number of strategies that may allow achieving a breeding ratio close to unity in PWRs. A substantial improvement in the conversion ratio can be achieved by changing the assembly configuration and fuel pin geometry. The best conversion ratio obtained in this work was about 0.91. It was achieved by means of using large pin dimensions and low $^{233}$U content in the Th blanket.

At this point, we did not consider thermal design and safety related issues. The major concern is the concentration of fissile material, primarily in the seed region, which causes a large power peak, challenging feasibility of the concept from a thermal-hydraulic point of view.

This study can only be regarded as preliminary. In the future, the seed-blanket assembly geometry with Th-U233 fuel will be optimized with respect to obtaining the highest possible conversion ratio, while maintaining the thermal margins comparable to those of existing reactors.
Figure 5. Assembly k-inf vs. irradiation time for different seed $^{233}$U content. Modified SB assembly

Figure 6. CR vs. burnup for different seed $^{233}$U content. Modified SB assembly

REFERENCES

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Breeding of $^{242m}$Am in a PWR

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INTRODUCTION

$^{242m}$Am is an actinide with very high fission cross section and a large number of neutrons released per fission. Also it has relatively long half life of 141 year – large enough for practical use. Therefore, it can be used for various applications such as: space propulsion, medical application and energy production \(^{(1-4)}\). The most significant drawback of $^{242m}$Am is that it is very difficult to produce in appreciable quantities. Only small amount of $^{242m}$Am (0.1\% from all Am) is present in the spent fuel. In contrast to $^{242m}$Am, significant quantities of $^{241}$Am are available in the spent fuel of power reactors as a product of $^{241}$Pu beta decay with half life of 14 year. Therefore, $^{241}$Am can be easily separated chemically from the spent nuclear fuel. Then, $^{242m}$Am can be produced by irradiation of $^{241}$Am through neutron capture reaction with energy dependent branching ratio to a metastable $^{242m}$Am state.

$$
^{241\text{Am}}_{95} (n,y) \rightarrow ^{242m\text{Am}}_{95}, \quad b \quad ^{242\text{Am}}_{95} (T_{1/2} = 141y) \quad 1-b \quad ^{242g\text{Am}}_{95} (T_{1/2} = 16h)
$$

It was recently suggested to breed $^{242m}$Am in the fast reactors because of its more favorable $^{241}$Am to $^{242m}$Am branching ratio and $\sigma_{cap}^{241\text{Am}}/\sigma_{abs}^{242m\text{Am}}$, $\sigma_{cap}^{243\text{Am}}/\sigma_{abs}^{242m\text{Am}}$ cross section ratios in the fast neutron energy spectrum \(^{(5,6)}\). The $\sigma_{cap}^{241\text{Am}}/\sigma_{abs}^{242m\text{Am}}$, $\sigma_{cap}^{243\text{Am}}/\sigma_{abs}^{242m\text{Am}}$ ratios are responsible for the mass and enrichment of bred $^{242m}$Am. These ratios should be as high as possible for the efficient breeding of $^{242m}$Am.

In thermal reactors, equilibrium concentration of $^{242m}$Am is relatively low because of its high thermal absorption cross section. However, epithermal $^{241}$Am capture and $^{242m}$Am absorption cross sections are much more favorable for $^{242m}$Am breeding (Fig.1). In our study, we explore the possibility of producing $^{242m}$Am in the existing Pressurized Water Reactors (PWR) by shielding the Am breeding targets with strong thermal absorbers. The use of the thermal absorber makes it possible to take advantage of strong resonance absorption in $^{241}$Am while preventing $^{242m}$Am destruction through thermal neutron absorption.
In our study, standard Wet Annular Burnable Absorber (WABA) geometry was used for Am breeding targets. The $^{241}$Am targets were assumed to be inserted into the guide tube positions of the fuel assembly. $^{241}$Am was assumed to be in a metallic form coated with thermal absorber.

The main design objective was to breed maximum amount of $^{242m}$Am with minimum interference with the reactor performance. Therefore, the following assumptions were made:
- $^{241}$Am targets are placed in all 24 guide tubes of fresh fuel assemblies;
- The targets can be irradiated only in discrete time increments equal to the length of the fuel cycle;
- After each cycle, the thermal absorber around breeding targets is renewed and the targets are relocated into fresh fuel assemblies;

![Graph](image.png)

**Figure 1:** Energy dependence of the $\sigma_{\text{cap}}^{241\text{m}Am}/\sigma_{\text{obs}}^{242mAm}$ ratio.

The following thermal absorbers were considered: natural Cd, Gd, B and Sm. It was found that the use of the Gd yields the best performance with respect to $^{242m}$Am breeding. However, the using of Gd due to his effect on reactivity required increasing of the fuel enrichment to 5%. In such a case there is no interference with the reactor performance. Non-linear reactivity model was used for the fuel cycle length calculations $^{(7,8)}$. 
RESULTS
In the course of the study, we optimized the thickness of thermal absorber and $^{241}$Am target in order to obtain the maximum mass and enrichment of $^{242m}$Am with reasonable fuel cycle length. The optimal parameters that yield the most favorable performance are summarized in Table 1:

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{241}$Am target inner radius, cm</td>
<td>0.343</td>
</tr>
<tr>
<td>$^{241}$Am target outer radius, cm</td>
<td>0.353</td>
</tr>
<tr>
<td>Thermal absorber</td>
<td>Gd$_2$O$_3$</td>
</tr>
<tr>
<td>Thermal absorber density, gr/cm</td>
<td>0.45</td>
</tr>
<tr>
<td>Fuel cycle length, Full power days</td>
<td>440</td>
</tr>
<tr>
<td>Irradiation time, # of batches</td>
<td>3</td>
</tr>
</tbody>
</table>

These parameters were used in the subsequent $^{242m}$Am enrichment and mass calculations. It was found that the optimal target irradiation period lies between second and third fuel reloads. Therefore, three cycle irradiation (1320 EFPD) was used for further analysis. Figures 2 and 3 show the bred $^{242m}$Am mass and enrichment as a function of fuel burnup.

![Figure 2: $^{242m}$Am amount vs. time.](image1)

![Figure 3: $^{242m}$Am enrichment vs. time.](image2)

As we can see from the Figures, at end of third cycle, 85 gram of $^{242m}$Am enriched up to 7% can be obtained from each fuel assembly. About 1.55 kg of $^{242m}$Am can be bred per year in a typical 1000 MWe PWR core if 1/3 of the fuel assemblies contain $^{241}$Am targets.
CONCLUSIONS
The main goal of this study is to explore the possibility of breeding $^{242m}$Am in the existing PWR-type reactors. In contrast to other studies related to $^{242m}$Am production, the main advantage of this option is that PWRs are readily available and do not require any particular modifications in order to accommodate $^{242m}$Am breeding targets.

The main design objective of the study was to minimize the influence of the $^{242m}$Am breeding on the reactor operational parameters. A number of different thermal absorber materials with different densities and dimensions were considered to shield the breeding targets from thermal neutrons and maximize the enrichment of irradiated target.

It was shown that significant quantities (1.55 kg/GWe-Y) and relatively high enrichment (up to 7%) of $^{242m}$Am can be obtained in PWR reactors without significant interference with the reactor performance. The enrichment of the driver (UO$_2$) fuel will have to be increased from the typical 4.2% to 5% in order to compensate for the reduction in fuel cycle length as a result of the breeding targets presence in the core.

REFERENCES

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The Use of 241Am to Enhance Proliferation Resistance of LWR Fuel Cycle

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INTRODUCTION

The issue of nuclear weapons proliferation poses one of the major problems for the use of nuclear energy. Increasing the spent fuel 238Pu to total Pu ratio is the most effective way to reduce or prevent the risk of proliferation from the civilian fuel cycle. Recent studies showed that Pu mixture with more than 6% 238Pu content can be considered as proliferation resistant (1,2).

In this work, we suggest using 241Am as a proliferation-resistant burnable poison in light-water reactors. We suggest that a fuel mixture of UO2 and 241AmO2 can not only prevent proliferation risk, but also reduce the burnable poison requirements. The proliferation resistance of the conventional fuel mixed with 241Am is enhanced by increasing the spent fuel 238Pu/Pu ratio. Burnable poison savings are obtained by taking advantage of favorable characteristics of 241Am transmutation chain, which allows substitution of some of the conventional burnable poison, Wet Annular Burnable Absorber (WABA), rods with the 241Am added to the fuel mixture.

In this paper, we investigate the 241Am loading requirements necessary to obtain proliferation resistant Pu isotopic composition in the spent fuel and burnable poison requirements for the core reactivity control. Three parameters of interest that require optimization as a result of introduction of 241Am into the fuel are the 238Pu fraction in the Pu vector at fuel discharge, the amount of excess reactivity being suppressed at the Beginning of Cycle (BOC) and the fuel cycle length penalty as a result of residual burnable poison and 241Am.

The isotope 238Pu is obtained from 241Am by the neutron capture reaction and subsequent decay of the reaction products:

\[
241\text{Am} + n \rightarrow^{\theta} 242\text{Am} \rightarrow^{\beta^-} 242\text{Cm} \rightarrow^{\alpha} 238\text{Pu}
\]

Where \(\theta\) is the branching ratio to 242Am ground state (3).

The 241Am one group thermal spectrum averaged absorption cross-section is about 800 barns (4); therefore, it can be effectively burned in thermal reactors. In addition, the neutron capture cross sections of the reaction products are much smaller than the capture cross section of 241Am. Furthermore, 242mAm bred from 241Am has outstanding fissile properties, which reduces the residual burnable poison reactivity penalty. These facts suggest that 241Am can be effectively used as a burnable poison.

METHODOLOGY

In our analysis, a pressurized water reactor (PWR) with a 17x17 rods assembly and 3% neutron leakage rate was considered. This reactor uses a 4.5% enriched uranium fuel and it operates with 3-batch fuel management. It also uses 24 BP WABA rods (BPR) per assembly in the first fuel cycle (5).

The BOXER computer code (6) was used for the fuel-assembly multiplication factor and burnup calculations. A nonlinear reactivity model was used for the fuel-cycle length and core multiplication factor calculations (7,8).
In the current study, we investigate the use of $^{241}$Am as proliferation resistant burnable poison in two steps: first, we found the minimal amount of the admixed $^{241}$Am required to obtain $^{238}$Pu/Pu ratio greater than $6\%$. In this step, different amounts of $^{241}$Am were added to the fuel. In the second step, we found the number of WABA rods that need to be used in order to obtain the same core criticality as for fuel without admixed $^{241}$Am. In this step, the core criticality and cycle length were calculated.

We assumed that the spent fuel was cooled for 5 years after discharge. In this time, most of the $^{242}$Cm that present in the discharged fuel decays to $^{238}$Pu. Therefore, the total amount of $^{242}$Cm in the discharge fuel was added to $^{238}$Pu.

**RESULTS**

The performed analysis showed that admixture of about $0.15\%$ $^{241}$Am can both, increase the $^{238}$Pu/Pu ratio to more than $6\%$ and keep the fuel cycle length without significant reduction (Figs 1,2).

![Figure 1: Assembly multiplication factor.](image)

We can see from Figure 1 that the multiplication factor behavior with burnup is similar for conventional fuel and for the fuel with admixed $^{241}$Am. This means that the fuel cycle length is also about the same.
Figure 2 shows that the spent fuel $^{238}$Pu/Pu ratio is always above 6% if $^{241}$Am is mixed with the fuel. For about 50 kilograms of $^{241}$Am added per batch (about 0.8 kg per fuel assembly) to the 4.5% UO$_2$, we can obtain about 7% $^{238}$Pu content in the Pu. It was found that $^{241}$Am admixture to the fuel causes almost no penalty to the fuel cycle length (Table 1). However, increasing the ratio of $^{238}$Pu/Pu by increasing the initial amount of $^{241}$Am added to the fuel will result in a small reduction in the fuel-cycle length. Also, it was concluded that the use of $^{241}$Am for nonproliferation purposes reduces the burnable poison requirements by a factor of two, 12 WABA instead of 24 WABA. It was shown that criticality of the core with admixture of $^{241}$Am and 12 WABA rods is similar to that of the core with 24 WABA rods and without admixture of $^{241}$Am (Figure 3).

Table 1: fuel cycle length:

<table>
<thead>
<tr>
<th>Case</th>
<th>Fuel cycle length (EFPD)</th>
</tr>
</thead>
<tbody>
<tr>
<td>4.5% UO2</td>
<td>477</td>
</tr>
<tr>
<td>4.5% UO2 with 24 WABA</td>
<td>457</td>
</tr>
<tr>
<td>4.5% UO2 with 0.15% of $^{241}$Am admixture</td>
<td>464</td>
</tr>
<tr>
<td>4.5% UO2 with 0.15% of $^{241}$Am admixture and 12 WABA</td>
<td>452</td>
</tr>
</tbody>
</table>
It was also found that the soluble boron worth is comparable for both cases, which indicates that the core critical boron concentration will also be similar. The observed $^{241}$Am transmutation efficiency are fairly high. For an initial load of 50 kg of $^{241}$Am per batch, at the EOL, we obtain only 3.3 kg, that is about a factor of 15.2 reduction in the amount of $^{241}$Am. The total amount of Am (accounting for all Am isotopes at EOL) is reduced by about a factor of 5. Therefore, the use of $^{241}$Am as a proliferation-resistant burnable poison also offers an attractive and effective option for $^{241}$Am transmutation.

CONCLUSIONS
The main conclusions from this study can be summarized as follows:
1. Admixture of small amounts of $^{241}$Am to the conventional PWR UO$_2$ fuel can increase the $^{238}$Pu/Pu ratio in the spent fuel to about 7%, which would assure proliferation-resistance without significant penalty to the fuel cycle length.
2. The $^{241}$Am introduced into the fuel can substantially reduce the burnable poison requirements by a factor of 2.
3. The $^{238}$Pu/Pu ratio can be increased beyond 10%, although this will result in larger cycle length penalty.
4. High transmutation efficiency of $^{241}$Am can be achieved.
REFERENCES


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Some Implications of Branching Ratio of the 241Am(n,γ) Reaction
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INTRODUCTION
The growing interest in innovative reactors and advanced fuel cycle designs requires more accurate prediction of various trans-uranic actinide concentrations during irradiation or following discharge because of their effect on reactivity or spent fuel emissions such as gamma and neutron activity and decay heat.

Many of the important, in this respect, actinides originate from 241Am(n,γ) reaction (Fig. 1), which leads to either ground or metastable state of 242Am. The branching ratio for this reaction depends on the incident neutron energy and has very large uncertainty in the current evaluated nuclear data files (Fig. 2).

241Am is obtained by the β− decay of 241Pu with a half-life of 14 years, and as a result found in conventional Uranium spent fuel and MOX fuel. Significant amount of generated 241Am does not undergo fission but converted into 242Am in a ground or meta-stable state. 242gAm decay eventually results in accumulation of Cm isotopes, which are particularly important in the assessment of spent fuel characteristics. 242mAm, on the other hand, has relatively long half-life of about 141 years and exceptionally good fissile properties. Therefore, even small amounts of 242mAm in the fuel may significantly affect the core reactivity.

As evident from Figure 1, the amount of 242mAm and 242gAm obtained from the 241Am(n,γ) reaction is highly dependant on the reaction branching ratio b. Therefore, the branching ratio values are very important for the accurate calculation of fuel composition and criticality.

Several studies have been conducted (1,2) on the measurement and theoretical calculation of the 242mAm branching ratio - b. It was found that the branching ratio is extremely sensitive to the energy of the captured neutrons. However, the branching ratio data still has large discrepancy in different evaluated data files as illustrated in Figure 2.
In most of the neutronic modeling codes that provide burnup calculations, only one value for the $^{242}_{\text{Am}}$Am branching ratio is typically used. Therefore, it is very important to account for the energy dependence of the branching ratio, in order to have a single, averaged value, so that amount of $^{242}_{\text{Am}}$Am will be calculated correctly. In this study, we investigated the effect of the branching ratio ($b$) on fuel composition and criticality. In the first part of the study, we have calculated the $^{242}_{\text{Am}}$ spectrum-weighted average branching ratio for a wide range of representative fuel and reactor types using the three considered databases: ENDF, JEFF, and JENDL.

In the second part, we have investigated the effect of using the calculated average spectrum weighted branching ratio from different databases on fuel composition and criticality of the selected representative fuel and reactor types.

**METHODOLOGY**

In order to evaluate the effect of the branching ratio energy dependence and the effect of discrepancies in the branching ratio values as they appear in different data files, we considered some innovative as well as conventional fuel and reactor designs. Table 1 summarizes cases, for which spectrum averaged branching ratio was calculated.

<table>
<thead>
<tr>
<th>Case #</th>
<th>Case description</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Typical PWR UO$_2$ fuel, 4.5% enrichment, at 55 MWd/kg</td>
</tr>
<tr>
<td>2</td>
<td>U-Pu MOX fuel$^{(3)}$</td>
</tr>
<tr>
<td>3</td>
<td>$^{241}$Am in a fast neutron spectrum</td>
</tr>
</tbody>
</table>

The averaging of the branching ratio is obtained by conserving the reaction rate leading to $^{242}_{\text{Am}}$ Am production Eqs. (1):

$$
\bar{b} = \frac{N_{241} \cdot \int b(E) \cdot \phi(E) \cdot \sigma_c(E) dE}{N_{241} \cdot \int \phi(E) \cdot \sigma_c(E) dE} \quad \phi(E) \cdot \sigma_c(E) = RR_c(E)$$

$$
\bar{b} = \frac{\int b(E) \cdot RR_c(E) dE}{\int RR_c(E) dE}
$$

where: $N_{242}_{\text{Am}}$ and $N_{241}$ - are the atom densities of the $^{241}_{\text{Am}}$ Am and $^{242}_{\text{Am}}$ Am respectively,

$\phi(E)$ - is the energy dependant neutron flux,

$\sigma_c(E)$ - is the neutron capture cross-section of $^{241}_{\text{Am}}$ Am.

$RR_c(E)$ - is the capture reaction rate in $^{241}_{\text{Am}}$. 

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The 24th Conference of the Nuclear Societies in Israel
The reaction rates in thermal light water reactor fuels were calculated by using two-dimensional transport and depletion computer code - BOXER (4), which was developed for the analysis of LWRs. For the fast neutron spectrum reactors, the reaction rate was calculated using MCNP4c (5). Fuel depletion calculations in fast reactor cases were performed using BGCore computer code (6). BGCore is a software package for comprehensive computer simulation of nuclear reactor systems and their fuel cycles. It couples the Monte Carlo particles transport code MCNP4c with fuel depletion and decay routine.

The branching-ratio calculations for thermal reactors were calculated at typical PWR operating conditions, on a single assembly with the reflective boundary conditions basis. A fast flux spectrum was obtained for a MONJU reactor(7) single fuel assembly with reflective boundary conditions.

RESULTS
The main finding of this study is that the branching ratio varies significantly from one case to another depending on fuel composition, as well as on the reactor type. Moreover, the effect of the database used for calculation of branching ratio is also substantial.

Our results are presented in Table 1.

<table>
<thead>
<tr>
<th>Database</th>
<th>Country</th>
<th>Case 1</th>
<th>Case 2</th>
<th>Case 3</th>
</tr>
</thead>
<tbody>
<tr>
<td>JEFF 3.1</td>
<td>Europe</td>
<td>0.130</td>
<td>0.137</td>
<td>0.154</td>
</tr>
<tr>
<td>JENDL 3.3</td>
<td>Japan</td>
<td>0.137</td>
<td>0.183</td>
<td>0.304</td>
</tr>
<tr>
<td>ENDF /B-VI.8</td>
<td>U.S.A.</td>
<td>0.120</td>
<td>0.133</td>
<td>0.219</td>
</tr>
</tbody>
</table>

In this part of the study, a series of burnup calculations were performed to evaluate the effect of using spectrum average branching ratio values calculated previously using different databases on reactivity and fuel composition. The calculations were performed for all cases listed in Table 2.

There are number of cases, in which the branching-ratio value becomes important. Such cases are related to the development and use of advanced fuels, $^{242m}$Am production and minor actinide recycling (11). In our study, a number of parameters were selected to assess the effect of branching ratios calculated on the basis of different evaluated data files. These parameters are the neutron multiplication factor (k-inf) and concentrations of $^{243}$Am, $^{242}$Cm-$^{248}$Cm and $^{238}$Pu isotopes as a function of burnup. The infinite medium neutron multiplication factor of every fuel type ultimately defines the core reactivity control requirements, fuel cycle length and the core power distribution. The selected isotope concentrations have an important effect on the spent fuel radiotoxicity, decay heat, and proliferation resistance characteristics.

It was found that the difference in $k$ and $^{238}$Pu, $^{243}$Am, Cm isotopes densities for ENDF /B-VI.8 and JENDL 3.3 branching-ratios in conventional 55 MWd/kg UO$_2$ fuel were small.

In the MOX fuel case, on the other hand, the $^{241}$Am concentration is much higher than in UO$_2$ fuel. Therefore, the importance of branching ratio value is higher as well. Furthermore, the concentration of $^{241}$Am in the MOX fuel depends on the time period passed since the separation of Pu from the spent UO$_2$ fuel. In some cases, MOX fuel can contain up to 5.5% of $^{241}$Am (7). As a result, we have analyzed MOX fuel that has Pu isotopic vector corresponding to that of the spent UO$_2$ fuel with 55 MWd/kg discharge burnup, 7 years cooling before separation and different time periods between separation and the beginning of irradiation as MOX. The time periods chosen were 2, 4, 10 and 14 years.
Figure 3: Differences in multiplication factor for MOX fuel: ENDF /B-VI.8 vs JENDL 3.

Figure 4: $^{239}$Pu amount differences for ENDF /B-VI.8 and JENDL 3.3 branching-ratios.

Figures 3 and 4 summarize the results of the analysis for MOX fuel. As the decay time between the Pu separation and MOX fuel irradiation increases, so does the concentration of $^{241}$Am in the MOX fuel. As a result, the largest differences in k-inf and the studied nuclide concentrations are observed for the “oldest” Pu case. The difference in k-inf calculated with ENDF-VI and JENDL3.3 branching ratio data exceeds 200 pcm in the 14 years old Pu MOX case, the difference in $^{239}$Pu concentration could be as high as 3%. The difference in $^{243}$Am concentration may reach 1%. The Cm isotopes concentrations sensitivity to the value of the branching rations is relatively high but the absolute amount of Cm in the fuel is still fairly small.

The large amounts of $^{241}$Am are present in cases that related to minor actinide recycling. Additional case analyzed in this study was the fast reactor loaded with TRU fuel and operating in close to equilibrium fuel cycle with near unity conversion ratio. Relatively large loading of $^{241}$Am, in this case, is a consequence of TRU recycling including all minor actinides. In the analyzed case, the TRU composition originated from a typical LWR spent fuel.

The results of the analysis are presented in Figures 5 and 6. The differences may reach up to 5% in $^{238}$Pu density, over 1.5% in $^{243}$Am, and over 6% in Cm isotopes densities for the JENDL3.3 versus JEFF3.1 data. The discrepancies are a factor of two smaller if ENDF-VI branching ratio is used as compared with JEFF3.1 data. The difference in the fast reactor neutron multiplication factor also cannot be regarded as negligible. The discrepancies in k-inf between the used JENDL and JEFF branching ratio data are as high as 200 pcm, while agreement between the ENDF and JEFF is much better. In the latter case, the maximum difference in k-inf is only about 80 pcm.

Figure 5: Differences in multiplication factor for TRU-MOX fuel relative to JEFF 3.1 branching ratio.

Figure 6: Total Cm amount differences for TRU-MOX fuel relative to JEFF 3.1 branching ratio.
CONCLUSIONS
In this work, we have investigated the effect of uncertainty in $^{241}$Am (n,γ) reaction branching ratio on the accuracy of the neutronic modeling of various reactors. Three major neutron interaction evaluated data files were considered: JEFF3.1, ENDF-VI.8, and JENDL3.3. The $^{241}$Am (n,γ) reaction branching ratio was calculated by spectrum weighting of the evaluated data for a wide range of reactor systems – from conventional LWRs with standard UO$_2$ fuel to advance fast spectrum reactors with heavy loading of minor actinides. The following conclusions have been obtained based on the results of the performed analysis.

1. The $^{241}$Am (n,γ) reaction branching ratio is energy dependant and, thus, it is determined by the fuel material composition and geometry. In the conventional reactor design codes however, a single value of the branching ratio is typically used. The results of the performed analysis show significant variation of the branching ratio depending on the fuel and reactor type and suggest that the spectrum average branching ratio should be used, especially in fast spectrum reactors where the branching ratio energy dependence is particularly strong.

2. As a result of uncertainty in the branching ratio, the prediction accuracy of some important nuclide densities as well as neutron multiplication factor can be strongly affected. The performed modeling of various reactor and fuel designs using different evaluated data files revealed that:
   - The use of different branching ratio values has practically no effect on criticality and isotopic composition of a conventional UO$_2$ fuel in PWR environment.
   - Much more pronounced effect was observed for the MOX fuel in PWRs; namely, up to 0.3% difference in criticality, up to 4% difference in $^{239}$Pu concentration, and 1% and 6% difference in $^{243}$Am and Cm isotopes concentrations respectively.
   - In the next generation fast reactors with sustainable TRU recycling, the use of different branching ratio values also have very important effect on the recycled actinides isotopic composition and the fuel criticality. The observed differences in isotopic densities are up to 5%, 6.5% and 1.5% for $^{238}$Pu, $^{242}$Cm-$^{248}$Cm and $^{243}$Am respectively if different data sources were used for calculating the spectrum weighted branching ratio. These differences in the fuel composition result in up to the 200 pcm difference in the fuel criticality.

3. Since next generation fast reactors with TRU recycling become increasingly more important subject of research, we recommend giving high priority to experimental measurements of the $^{241}$Am (n,γ) reaction branching ratio in order to reduce its uncertainly in evaluated data files and improve the accuracy of advanced reactor designs.

REFERENCES:

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The EGS5 Electron Transport in Magnetic Fields

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INTRODUCTION

The recent developments in the EGS5 Monte Carlo code system include an advanced electron transport scheme, which overcome previous known discrepancies corresponding condensed history electron step-size in media [1]. The new scheme is based on a set of hinge points for energy loss and another set of hinge points for multiple scattering. During the last two decades, the magnetic field implementation in EGS4 was carried out using the EMF macro code. The routines in EMF were based on the equation developed by Bielajew, published in Jenkins 1988 for relativistic motion in vacuum [2]. After a series of tests, it was found that the equation of motion was mistakenly derived, and the magnetic term must be inspected.

METHOD

The equation of motion for electron in electrical and magnetic field is given by:

\[
\frac{d\vec{P}}{dt} = e \left( \vec{E} + \vec{\nu} \times \vec{B} \right)
\]

\( P \) is the momentum of electron. For a relativistic momentum, the following relations are known:

\[
\beta = \frac{\vec{v}}{c}, \quad \gamma = \frac{1}{\sqrt{1 - \beta^2}}, \quad \gamma = \frac{E}{m_0 c^2}
\]

In form of path-length derivative, Eq. 1 becomes:

\[
\frac{d(\gamma \vec{p})}{ds} = \frac{e}{m_0 c^2 \beta} \left( \vec{E} + e \left[ \vec{\beta} \times \vec{B} \right] \right)
\]

But,

\[
d(\gamma \beta) = \beta d\gamma + \gamma d\beta
\]

\[
d\gamma = d \left( \left( 1 - \beta^2 \right)^{\frac{1}{2}} \right) = \beta \left( 1 - \beta^2 \right)^{\frac{3}{2}} = \beta \gamma^3
\]

Hence,

\[
\frac{d(\gamma \vec{p})}{ds} = \vec{\beta} \beta \gamma^3 \frac{d\beta}{ds} + \gamma \frac{d\beta}{ds} = \left( \beta^2 \gamma^2 + 1 \right) \gamma \frac{d\beta}{ds} = \frac{1}{1 - \beta^2} \gamma \frac{d\beta}{ds} = \gamma^3 \frac{d\beta}{ds}
\]

Eq. 2 can be written as,
\[
\frac{d\vec{\beta}}{ds} = \frac{e}{m_e c^2 \beta \gamma^3} \left( \vec{E} + c \left[ \vec{\beta} \times \vec{B} \right] \right)
\]

Eq. 3 obtained here is different from Eq. (19.3) in Bielajew's work (in Jenkins et al 1988), due to an error found in Eq. (19.3) of Bielajew's work. The origin of the error is the E-field term developed in Bielajew's previous work [3].

The electric field and the magnetic field are two non-dependent additive terms in Eq. 3, therefore we allowed to separately treat each field influence on the electron motion.

The Magnetic Field Effect is:

\[
\frac{d\vec{\beta}}{ds} = \frac{ec}{m_e c^2 \beta \gamma^3} \left[ \vec{\beta} \times \vec{B} \right]
\]

RESULTS

Implementing magnetic fields to the electron transport can be preformed by taking the directional differential obtained from Eq. 4:

\[
\beta \frac{d\vec{u}}{ds} = \frac{ec}{m_e c^2 \beta \gamma^3} \left[ \vec{u} \times \vec{B} \right]
\]

\[
\frac{d\vec{u}}{ds} = \frac{ec}{m_e c^2 \beta \gamma^3} \left[ \vec{u} \times \vec{B} \right] \rightarrow \begin{cases} 
\frac{e}{m_e c^3 \beta \gamma^3} B & \text{for } d(u \parallel \vec{B}) = 0 \\
\text{and } d(u \perp \vec{B}) = 0 & \text{for } d(u \parallel \vec{B}) = 0 \end{cases}
\]
Figure 1: 20 MeV electron beam trajectories in a 1 Tesla magnetic field inside water – simulations results output image using the CG-View program.

Due to the need of changing these equations, a new method of electron transport in magnetic fields was developed for the EGS5. This method is based on a new subroutine named MAGNET, which is not depended on any free parameters for dominance matching the magnetic force to the multiple scattering.

Figure 1 shows the results of the new method in EGS5 simulation of 20 MeV electron beam in a 1 Tesla magnetic field inside water.

CONCLUSIONS

The EGS5 code includes verified magnetic field effects on electron transport, which is accessible to the user. The new subroutine enables to define the magnetic field in any direction in respect to the electron beam direction. The documentation of magnetic fields for EGS5 is now detailed and accurate, without any unexplained parameters to be set. The multiple scattering is matched to the B-field strength via the electron transport model.

Simulations of magnetic fields in applications such as for medical physics are available to be performed by users.

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MCNP Simulations of NSD as a Basis to Optimization for Detection of Illicit Trafficking of Radioactive Material

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INTRODUCTION

The NSD is a portable Neutron Searching Detector developed at NRCN\textsuperscript{1} with a high efficiency for counting fast and thermal neutrons employing improved gamma rejection. In order to fulfill the latest standards for Detection of Illicit Trafficking of Radioactive Material\textsuperscript{2} the NSD should be optimized using MCNP neutron transport code to establish a 5 kg maximal weight detector configuration required to detect and localize neutron emitting radioactive material. The optimized NSD should detect a slight variation in the range of usual background caused mainly by illicit trafficking or inadvertent movement of radioactive material. It can be used to for pedestrian monitoring in non-canalized configurations or scan larger volume items as a vehicle or a container at check points or critical areas.

MCNP SIMULATIONS FOR NSD

The neutron detection of the NSD is based on two \(^3\text{He}\) detectors installed within a polypropylene moderator achieving low background by discrimination of gamma and electronic noise pulses. The NSD detection efficiency is mainly based on the neutrons energy. It is in direct correlation with the \(^3\text{He}\) cross section. The lower the neutron kinetic energy the higher the cross section. Therefore to increase the detector's sensitivity the NSD includes polypropylene surrounding the \(^3\text{He}\) tubes. The polypropylene effectively moderates the neutrons due to its high hydrogen concentration. The neutron detection within the \(^3\text{He}\) sensors is based on the following reaction\textsuperscript{4}:

\[ \frac{1}{0}n + \frac{3}{2}\text{He} \rightarrow \frac{1}{1}p + \frac{3}{1}\text{H} + 764 \text{ keV} \]

The current structure of the NSD has been designed in a way that the highest efficiency is achieved in the front of the NSD. As well, an environment containing plenty of hydrogen like building materials and water concentrations can function as a natural moderator to the fast neutrons and improve detection efficiency significantly.

Monte-Carlo code MCNP-4C2 (Breiesmeister 2000)\textsuperscript{3} was used to perform Neutron detection calculations utilizing the F4 average cell flux tally. This tally is used to estimate the expected flux value in a specific cell. Weighting Tally F4 by the appropriate material atomic density and absorption cross section using FM4 and F4 combination will score the number of neutrons absorbed in cell at the same flux. This code combination was used to evaluate the number of neutrons absorbed in each of the NSD's \(^3\text{He}\) tubes by placing it at a neutron flux emitted by \(^{252}\text{Cf}\) point source. \(^{252}\text{Cf}\) spontaneous fission energy spectrum was taken according to Watt fission spectrum from MCNP4C2 libraries.

This paper presents the correlation between current NSD measurements versus MCNP4C2 calculations. NSD optimization will be performed by changing the suitable parameters in the MCNP4C2 code.
RESULTS
MCNP calculations were compared to NSD for a \(^{252}\text{Cf}\) source placed at 80 cm to 150 cm distances in front of the detector. Tables 1,2 summarize NSD's measured versus calculated count rates for two set of measurements. Each count rate was obtained by using a \(^{252}\text{Cf}\) source with a known activity at the required distance in front of the detector, thus creating a specific neutron flux hitting detector's front. The match-up between MCNP calculations and NSD's average measurements is well shown in Figure 1.

Table 1. NSD's Measured vs. Calculated count -rates of a 15353 [n/s] \(^{252}\text{Cf}\) source.

<table>
<thead>
<tr>
<th>(^{252}\text{Cf}) distance from NSD [cm]</th>
<th>Neutron Flux at front of NSD [n*cm(^{-2})*s(^{-1})]</th>
<th>NSD's Measured Count- rate [cps]</th>
<th>Calculated Count-rate [n/sec]</th>
<th>Measured / Calculated Count-rate [%]</th>
</tr>
</thead>
<tbody>
<tr>
<td>80</td>
<td>1.91E-01</td>
<td>3.99 ± 3.5%</td>
<td>4.34 ± 0.6%</td>
<td>-8.1%</td>
</tr>
<tr>
<td>100</td>
<td>1.22E-01</td>
<td>2.72 ± 4.2%</td>
<td>3.04 ± 0.8%</td>
<td>-10.5%</td>
</tr>
<tr>
<td>120</td>
<td>8.48E-02</td>
<td>2.34 ± 4.6%</td>
<td>2.24 ± 0.9%</td>
<td>+4.4%</td>
</tr>
<tr>
<td>150</td>
<td>5.43E-02</td>
<td>1.39 ± 6.0%</td>
<td>1.59 ± 1.1%</td>
<td>-12.6%</td>
</tr>
</tbody>
</table>

Table 2. NSD's Measured vs. Calculated count -rates of a 13749 [n/s] \(^{252}\text{Cf}\) source.

<table>
<thead>
<tr>
<th>(^{252}\text{Cf}) distance from NSD [cm]</th>
<th>Neutron Flux at front of NSD [n*cm(^{-2})*s(^{-1})]</th>
<th>NSD's Measured Count- rate [cps]</th>
<th>Calculated Count-rate [n/sec]</th>
<th>Measured / Calculated Count-rate [%]</th>
</tr>
</thead>
<tbody>
<tr>
<td>80</td>
<td>1.71E-01</td>
<td>3.75 ± 3.0%</td>
<td>3.89 ± 0.7%</td>
<td>-3.6%</td>
</tr>
<tr>
<td>100</td>
<td>1.09E-01</td>
<td>2.66 ± 3.5%</td>
<td>2.72 ± 0.8%</td>
<td>-2.2%</td>
</tr>
<tr>
<td>120</td>
<td>7.60E-02</td>
<td>1.80 ± 4.3%</td>
<td>2.01 ± 0.9%</td>
<td>-10.4%</td>
</tr>
<tr>
<td>150</td>
<td>4.86E-02</td>
<td>1.37 ± 4.9%</td>
<td>1.42 ± 1.1%</td>
<td>-3.5%</td>
</tr>
</tbody>
</table>

![Figure 1. NSD average measured Response to \(^{252}\text{Cf}\) source and the MCNP calculations vs. distance.](image)

The NSD rejects electronic and gamma noise using a discriminator set point. The slight difference between measurements and calculations can be corrected by appropriate set up of the discriminator.
MCNP calculations were also compared to measurements of NSD angular response to a $^{252}$Cf source at radius of 1 meter. The measured angular response curve at figure2 shows higher count-rates at the left side of the detector at $210^0$-$330^0$ angles. This is due to a higher sensitivity of the left $^3$He sensor at the NSD.

Table 3. NSD Measured vs. Calculated count-rate of a rotating $15353$ [n/s] $^{252}$Cf source

<table>
<thead>
<tr>
<th>Angle between Source &amp; Detector [deg] at 1 [m] radius</th>
<th>Measured Count Rate [cps]</th>
<th>Calculated Count-rate [n/sec]</th>
<th>Measured / Calculated Count-Rate [%]</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>$2.62 \pm 4.4%$</td>
<td>$2.95 \pm 0.8%$</td>
<td>$-11.2%$</td>
</tr>
<tr>
<td>30</td>
<td>$2.65 \pm 4.4%$</td>
<td>$2.83 \pm 0.8%$</td>
<td>$-6.4%$</td>
</tr>
<tr>
<td>60</td>
<td>$2.12 \pm 4.9%$</td>
<td>$2.50 \pm 0.9%$</td>
<td>$-15.2%$</td>
</tr>
<tr>
<td>90</td>
<td>$1.92 \pm 4.2%$</td>
<td>$2.13 \pm 0.9%$</td>
<td>$-9.9%$</td>
</tr>
<tr>
<td>120</td>
<td>$1.89 \pm 4.2%$</td>
<td>$2.16 \pm 0.9%$</td>
<td>$-12.5%$</td>
</tr>
<tr>
<td>150</td>
<td>$2.07 \pm 4.0%$</td>
<td>$2.15 \pm 0.9%$</td>
<td>$-3.7%$</td>
</tr>
<tr>
<td>180</td>
<td>$2.03 \pm 4.0%$</td>
<td>$2.26 \pm 0.9%$</td>
<td>$-10.2%$</td>
</tr>
<tr>
<td>210</td>
<td>$2.19 \pm 3.4%$</td>
<td>$2.21 \pm 0.9%$</td>
<td>$-0.9%$</td>
</tr>
<tr>
<td>240</td>
<td>$2.08 \pm 3.5%$</td>
<td>$2.18 \pm 0.9%$</td>
<td>$-4.6%$</td>
</tr>
<tr>
<td>270</td>
<td>$1.98 \pm 5.0%$</td>
<td>$2.13 \pm 0.9%$</td>
<td>$-7.0%$</td>
</tr>
<tr>
<td>300</td>
<td>$2.50 \pm 3.7%$</td>
<td>$2.49 \pm 0.9%$</td>
<td>$0.4%$</td>
</tr>
<tr>
<td>330</td>
<td>$2.77 \pm 4.3%$</td>
<td>$2.83 \pm 0.8%$</td>
<td>$-2.1%$</td>
</tr>
</tbody>
</table>

**Figure 2.** NSD measured and calculated angular response, each normalized by its count-rate at $0^0$
The calculated angular response at Figure 2 shows that the highest efficiency is achieved homogeneously in the front of the NSD according to its current design. It is possible to improve NSD detection efficiency while reducing its current directional efficiency by changing the tubes position in the moderator. Figure 3 shows the NSD internal representation in MCNP4C2 simulation.

Figure 3. Central cross sectional view in NSD (from the MCNP4C2 simulation)

CONCLUSIONS
This paper presents results of MCNP4C2 calculations of current NSD configuration and their correlation to NSD measurements. The comparison of these calculations to their equivalent measurements indicates that the NSD is well simulated by MCNP code. The match of NSD response to different distances of a $^{252}$Cf source along with its angular response are well shown. This match up is obtained by Weighting Tally F4 with appropriate material atomic density and absorption cross section using FM4 and F4 combination. This code, after being successfully validated, will be used to optimize the current NSD structure according to the latest standards for Detection of Illicit Trafficking of Radioactive Material[2].

REFERENCES


Acknowledgements
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INTRODUCTION
Calculations of the dose rates due to radionuclide release into the atmosphere from a nuclear reactor under accident or normal operation conditions are of great importance, both due to the regulatory requirements and to know the environmental impact of such releases. Model prediction of environmental dose rates from an atmospheric release of radioactive materials rely on the availability of meteorological data and on information of the source term [1]. Dose rates monitors and/or gamma radiations monitors placed around nuclear installations can provide an on-line estimation of the source term. Wind speed and direction, temperature etc...can be measured from an instrumented meteorological mast located at the site. In turn, these updated parameters may be used as an input for both long and short-range atmospheric dispersion models resulting in improved dose rates assessments.

Recently, we have developed a Fast Deploy Radiation Monitoring Array (FDRMA) which is composed of several fast Deploy Radiation Monitoring Stations, and one or more central stations [2]. Each FRMS includes a radiation detector, GPS, cellular modem, control unit, solar cells (optional), and meteorological instruments (optional).

The main target of this array is to measure dose rates and meteorological parameters (wind direction, wind speed, turbulence, etc..) in order to make a rapid assessment and prognosis of possible radiological consequences after an accidental release of radionuclides. Here, we present a basic simulation mode that was developed for this system, which is based on fast (real-time) dose rates calculation approach. The reduction of the problem from 3D to 1D integral, gives us the possibility to simulate the dose/fluence rates of the primary photons during a radionuclides release accident. The suggested approach can be easily implemented as a part of predictor – corrector estimator algorithm for source term estimation.

DOSE RATES CALCULATIONS
In the general case of radionuclide distribution an accurate estimation of a dose rate requires a numerical solution of a 3D integral.

The absorbed dose rate in air (in Gy\(^{-1}\)) at a detector position \(\vec{r}_0\) is given by

\[
\mathcal{D}(\vec{r}_0) = \frac{K E_p \mu_a}{4 \pi \rho} \hat{B} \int d^3 \vec{r} \frac{\exp\left(-\mu (\vec{r} - \vec{r}_0)\right)}{(\vec{r} - \vec{r}_0)^3} \]

(1)
in which \(K\) is a numerical factor depending on the unit choice; \(E_p\) is the photon energy; \(\mu\) and \(\mu_a\) are the linear attenuation and energy absorption coefficients respectively, and \(\rho\) is the air density. \(\hat{B}\) is a differential operator in \(\mu\) that describes the build-up of Compton-scattered radiation. In its simplest form, \(\hat{B} = 1 - k \mu (\partial / \partial \mu)\), with \(k = (\mu - \mu_a)/\mu_a\), it corresponds to a linear build-up factor.
The concentration distribution $\chi(\vec{r})$ is assumed to be Gaussian:

$$\chi(\vec{r}) = \frac{\theta(x)\dot{Q}}{2\pi u \sigma_y(x)\sigma_z(x)} \exp\left(-\frac{y^2}{2\sigma_y^2(x)}\right) \times \left[ \exp\left(-\frac{(z-h)^2}{2\sigma_z^2(x)}\right) + \exp\left(-\frac{(z+h)^2}{2\sigma_z^2(x)}\right) \right]$$

(2)

Where $h$ is the release height, $\sigma_y(x)$ and $\sigma_z(x)$ are dispersion coefficients (evaluated at the downwind distance $x$), $\dot{Q}$ is the release rate of radionuclides (the released activity per unit time), $u$ is the mean wind speed, and $\theta(x)$ is the Heaviside unit function.

For detectors placed on the ground surface ($z_0 \approx 0$) we can use a reduced 3D integral to a 1D integral approximation well suited for numerical computation [3], with receptor coordinates given by

$$\vec{r}_0 = \begin{pmatrix} x_0 \\ y_0 \\ z_0 \end{pmatrix} = \begin{pmatrix} r_0 \cos(\theta - \theta_0) \\ r_0 \sin(\theta - \theta_0) \\ 0 \end{pmatrix}$$

(3)

where $\theta$ is the wind direction (assumed to be the direction of the plume centerline), and $(r_0, \theta_0)$ the horizontal polar coordinates of the detector measured from the release point.

The dose rate can be written in this approximation as,

$$\dot{D}(x_0, y_0) = \frac{\dot{Q}v u_0}{4\mu \sqrt{\pi} \rho} \int_0^1 dt \exp\left[-g(t, x_0)\right] \times \text{erfc}\left(-\mu x_0 \frac{1-t}{t}\right) \times$$

$$\left[ \text{erfc}\left(\frac{t}{2(1-t)}\right) + \frac{k}{\sqrt{\pi}} \left(\frac{t}{1-t}\right) \exp\left(-\left(\frac{t}{2(1-t)}\right)^2\right) \right]$$

(4)

with

$$g(t, x) = \frac{(\mu y_0)^2(1-t)^2}{t^2 + 2[\mu \sigma_y(x)]^2(1-t)^2} + \frac{(\mu h)^2(1-t)^2}{t^2 + 2[\mu \sigma_z(x)]^2(1-t)^2} +$$

$$0.5\left\{ \ln\left[t^2 + 2[\mu \sigma_y(x)]^2(1-t)^2\right] + \ln\left[t^2 + 2[\mu \sigma_z(x)]^2(1-t)^2\right]\right\}$$

(5)

We compared the plume air absorbed dose rates values resulting from this 1D solution [Eq (4)] and from the exact 3D form [Eq (1)]. Generally, relative discrepancies of the calculated results were found to be less than 6% in various meteorological conditions and source terms. These calculations were carried out for six atmospheric stability classes, downwind distances up to 5000 m, crosswind distances up to 500 m, and source heights of 0 to 110 m.

In figure 1 we can see the simulated dose rates from both calculations (Eq 1 & 4). The calculations depicted in figure 1 were done for $^{41}Ar$ release from a 30m stack in F(2) stability class.
Figure 2 shows the discrepancy in [%] between the two methods of calculations. It is very clear that in this example the results are absolutely the same for downwind distance up to 2000 m. For 5000 m there is a slight difference of ~ 6%.

Generally, we can expect some greater deviations (not presented here) as the release height approaches ground level, and in 'large' crosswind distances. But, in all cases the agreement was reasonable for practical purpose like preliminary risk assessments.

![Graph showing dose rate vs downwind distance](image-url)

**Figure 1.** Direct plume Gamma dose rates as a function of downwind distance. The calculations were done for $^{41}$Ar release from a 30m stack in F(2) stability class.
**Figure 2.** The discrepancy (in %) between the two method of calculations (Eqn 1 & 4). The results are related to the calculated points in Fig 1 (For $^{41}$Ar release from a 30m stack in F(2) stability class).

**SUMMARY**

A method for simulating direct plume gamma dose rate by solving a 1D integral has been presented. The 1D approached solution have been compared to the exact 3D calculated dose rates and the relative error between the results was found to be less than 6% in most cases. Taking into consideration the calculation efficiency of the 1D solution (less than 1sec in Pentium 4), this method can be easily a part of computational efficient algorithm suited for real time assessment of the source term in case of a nuclear accident. The calculation method, shortly presented here, can be used in data assimilation. In particular, in our future work: estimation of the source term using off-site radiation monitoring data obtained close to the release point.

**REFERENCES**


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Radioactive Contamination Field Simulation for VMS

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INTRODUCTION
A new Vehicle Monitoring System (VMS) has been lately developed by the NRCN Electronics Laboratories. The system, designed for radioactive radiation fields mapping, consists of a base station and a dozen of monitoring vehicles, connected via cellular communication network. In order to enable operators' training, the VMS was designed with a special training mode that receives radiation field data from simulation software instead of the VMS radiation monitors. This work presents the concept and basic principles which will be implemented on the simulation software.

DISCUSSION AND RESULTS
Few years ago, the NRCN Electronics Laboratories developed a simulator\(^1\) for the Airborne Radiation Monitoring System (ARMS) operators. This simulator saved flight hours and offered the trainee the feeling of operating the system in a real radiation field. Nevertheless, a reused policy could not be applied for the developed VMS simulator because of the following reasons:

- The ARMS simulation software generates a well defined 3D radiation plume before training starts. This software requires a lot of computer resources and several minutes to complete the process. One of the VMS requirements is to produce the radiation field data in real time.
- The radiation field generated by the ARMS simulator is static and is designated to simulate a plume at a predefined time. The VMS simulator requirement is to evolve the field data in real time.
- The ARMS simulator generates enormous pre-training files. The VMS communication channel has limited-band pass and cannot transmit these huge files. Instead of transmitting dynamic data files, the VMS simulator will send to the monitoring vehicles dynamic field parameters.
- The ARMS simulator uses 3D radiation field to simulate the plume at any flight altitude. The VMS simulator needs only 2D radiation field to work on the ground-level.
- The ARMS simulator concept is based on continuous generating of particles and tracing each separated particle in semi-random course (see Figure 1). The VMS simulator cannot implement this iterative concept.

Figure 1. ARMS - contamination particles plume
In order to meet the VMS simulator requirements such a good estimation of real-like field and implementation in a standard mobile computer, a simple, non-iterative mathematical function was developed. The function parameters are location, wind speed, weather stability, time, contamination eruption rate, and eruption location. The base station shall transmit these parameters to the monitoring vehicles where the required field values will be calculated in real time.

A deterministic approach, defining the contamination plume at any time and adding a random effect to the final results, shall be implemented. The adopted function is based on the Pasquill\(^{(2)}\) dispersion model. This model is based on the assumption that the dynamic distribution of the plume is perpendicular to the wind direction (plume center line) and is Gaussian. The Gaussian parameter \(\sigma_y = f(x)\) monotonically decreases with the wind course. Figure 2 shows a plume correspondence to contamination generated by a constant wind (speed and direction).

![Figure 2. Plume based on Pasquill dispersion model](image)

In order to simulate a plume generated by dynamic changes in wind direction, a transformation to the XY plane will be applied. Instead of changing the plume state function, the plane is distorted. The following case assumes that the wind direction changes linearly in time. A zero random variable \(\sigma_y(x) = 0\) contamination plume is shown on Figure 3. The figure demonstrates a 100-sec. eruption time at 1 m/s constant wind speed. These parameters generate a straight 100-meter contamination "line" plume towards the wind course (Figure 3a). Wind direction change from west at \(t=0\) to south at \(t=100\) generates a plume that looks like a circle quarter (Figure 3b). The arc length and angles are the constrains; the radius and center location \([X, Y]\) are the free parameters.

![Figure 3. Wind direction change effect](image)
Figure 4 shows a 120-degree \((1/3 \pi)\) change in wind direction.
Where:

\[ z \text{ – Angle change [rad]} \]
\[ \alpha = \pi - z \]
\[ L \text{ – Plume length [m]} \]
\[ R = L/z \]
\[ X = R \times \text{Sin}(\alpha) \]
\[ Y = R \times \text{Cos}(\alpha) \]

**Figure 4. Arc parameters according to plume length and wind direction**

For constant wind speed and effective plume width smaller than \(R\) \((R>>\sigma_y(x))\), it can be assumed that contamination intensity behaves as the Gaussian model. The intensity value is changed as a function of the distance from the plume axis. Hence, the distance between point \([x,y]\) and X axis (plume axis without wind direction change) is transformed to the same distance between \([x',y']\) and the arc (wind direction change). The Gaussian distribution contamination that was perpendicular to the X axis is transformed to be perpendicular to the arc, at any point.

**Figure 5. Gaussian distribution perpendicular to the plume center line**

Figure 5 shows how point \([x,0]\) on the X axis transforms to \([x_1,y_1]\) on the arc:

1. \[x_1 = L/z(1 - \text{Cos}(x/L \times z - z + \pi/2)) - L/z^*(1 - \text{Cos}(z - \pi/2))\]
2. \[y_1 = L/z^* \times \text{Sin}(x/L \times z - z + \pi/2) + L/z^* \times \text{Sin}(z - \pi/2)\]

The arbitrary point \([x,y]\) is transforms to \([x',y']\):

3. \[[x',y'] = [x_1 - y \times \text{Cos}(\pi/2 - z^*(1 - x/L)), y_1 + y \times \text{Sin}(\pi/2 + z^*(1 - x/L))]\]

The result of the above is an algorithm to transfer any point \([x,y]\) from a constant wind case to a new \([x',y']\) location on a changed wind case determined by the wind parameters.

\[
\begin{align*}
[x',y'] & \Rightarrow [x_1 - y \times \text{Cos}(\pi/2 - z^*(1 - x/L)), y_1 + y \times \text{Sin}(\pi/2 + z^*(1 - x/L))] \\
\end{align*}
\]
Reverse Transform

In order to apply the above transformation to the VMS simulator, an inverse algorithm is required to obtain a contamination field value at any location \([x',y']\) on the transformed plane. The field value at \([x',y']\) (see Figure 5) is calculated by finding the corresponding \([x, y]\) point and placing it in the original contamination density function (at the pre-transformed plane). The algorithm includes five steps:

- Calculating the circle parameters, \(R, X\) and \(Y\), based on the plume parameters \(L\) and \(z\)
- Finding the distance \(y\) between the circle and the field point \([x',y']\)
- Finding \(x\) – the arc length from origin to \([x1,y1]\)
- Inserting the \([x,y]\) value into the 2D distribution function to obtain the contamination field value
- Calculating the radiation field value based on the contamination density

**Step 1 - Calculating the circle parameters**

\[
R = L / z
\]

\[
\cos(\pi - z) = Y / R \Rightarrow Y = R \cdot \cos(\pi - z)
\]

\[
\sin(\pi - z) = X / R \Rightarrow X = R \cdot \sin(\pi - z)
\]

**Figure 6. Circle parameters**

**Step 2 – Calculating the distance \(y\) between the circle and point \([x',y']\)**

\[
\cos(\alpha) = \frac{X - x'}{\sqrt{(X - x')^2 + (Y - y')^2}}
\]

\[
x1 = R \cdot \cos(\alpha) + X
\]

\[
a = \frac{y' - Y}{x' - X}, \quad b = Y - a \cdot X
\]

\[
yl = a \cdot x1 + b
\]

\[
y = \sqrt{(x' - x1)^2 + (y' - y1)^2}
\]

**Figure 7. Calculating the value of \(y\)**

**Step 3 - Finding the arc length \(x\)**

The ratio between the arc length and the circle perimeter \((2\pi R)\) is identical to the ratio between the angle corresponding to the arc and \(2\pi\). It will be easier to find the dotted arc length - \(d\) (Figure 8) that complements the arc length to \(L\).

\[
\cos(\beta) = \frac{yl - Y}{\sqrt{x1 - X)^2 + (yl - Y)^2}}
\]

\[
\beta = \cos^{-1}(\cos(\beta))
\]

\[
d = \beta \cdot R
\]

\[
x = L - d
\]

**Figure 8. Calculating the \(x\) value**
Step 4 - Calculating the contamination field
Placing $\sigma_y(x)$ and $y$ on the 2D field function gives the required contamination value:

$$F = C \cdot \frac{1}{\sigma_y(x) \cdot \sqrt{2\pi}} e^{-\frac{x^2}{2\sigma_y(x)^2}}$$

Step 5 - Calculating the radiation field
A real physical radiation field is affected by the whole plume contamination. Since radiation is $1/R^2$ dependent, summing the radiation contribution from the proximity surrounding area to any point is satisfying for the simulator.

CONCLUSIONS
A simple algorithm generating a radioactive plume for training simulation is presented. The algorithm uses small amount of computer resources that can be easily implemented on a standard mobile PC or microcontroller to run in real time. The algorithm requires few parameters that can be easily spread through a limited band pass communication network. The plume dynamics is based on the Pasquill Gaussian atmospheric diffusion and can be suitable to any weather stability class. The Pasquill model was expanded in order to be implemented in case of changing of wind direction.

REFERENCES

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**GammaGen - a Computer Code for Gamma-ray Spectra Generation**

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**INTRODUCTION**

"GammaGen" is a Windows software, developed at The Nuclear Research Center Negev (NRCN) to generate synthetic gamma ray spectra obtained with a NaI(Tl) scintillation detector and a solid state Ge detector. 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, as Gaussian of each photo peak, or as sum of all Gaussians. The Compton continuum can be included as well. The spectra mixture for different nuclides can be produced for visual analysis. All spectra can be exported to formats required by commercial spectra analyzing programs.

There are several purposes for developing of the program:
- to generate synthetic spectra for predicting the detectors response in case of a nuclear reactor accident.
- to check operational limits and performances of commercial spectrum unfolding programs.
- to develop a training tool for spectrometry laboratory workers.

**THE METHOD**

The program works with two libraries of radioisotopes. One is a general library, containing all practically encountered radioisotopes. This library is used for analyzing an arbitrary radioisotope mixture. The second library contains the fission products radioisotopes with half life higher than 1 min.

*Arbitrary radio-nuclides mixture*

Any radio nuclide mixture can be chosen by manual selection from the general nuclides library list. The activity default value of each radio-nuclide is 1 pCi. The radioisotopes and their activity can be chosen at will, to represent the chosen actual radioisotopes mixture. The detector type [NaI(Tl) or Ge] and the display method (energy lines, Gaussian or full spectra) must be chosen. The results are displayed on the computer screen.

*Fission products spectra*

One of the main purposes of the program is to predict the spectra which will be obtained after a nuclear reactor accident. Such spectra are complicated and include about 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, operation time and cooling time). The gamma ray spectrum reaching a detector placed near the accident site is dependent also on the distance from the contamination site. The detector parameters must be input for calculation of the detector response. If the detector is shielded by lead, the shield thickness must be defined too. The results are displayed on the computer screen.
Program operation

First, the library must be chosen (general library or fission products library). The main menu options of the program for the fission products library are shown in Figure 1a. On the left side are given the input parameters defining the reactor operational details and the detector definition. The fission products are given on the screen with selective markings. For the nuclear accident case, 7 groups of radioisotopes were defined, based on the data in the literature\(^1\). For each group there is a different specific release probability, including the option of release to a plume. Different combinations of the 7 groups can be chosen by the boxes on the right side of the screen. The main menu options for the general radioisotopes combination is given in Figure 1b. The principle is similar to the fission products screen, except the option to determine arbitrary activity values for the chosen radioisotopes.

Screen output characteristics

For easy visual identification of the radio nuclides, a specific color can be applied to the various peaks displaying energy lines or separate Gaussians. The Compton continuum of the radio-nuclides mixture can be displayed in a chosen color as well. The area of each photo peak is computed by the radionuclide activity, intensity and abundance of each energy peak (as listed in the nuclides library), and based on the detector's efficiency curve. If energy line display is chosen, each energy line is displayed proportional to the calculated count rate height. Gaussian spectrum is displayed according to the detector resolution, and its height is calculated to give an area corresponding to the total count rate. The Compton continuum is approximated to a rectangle by calculating the Compton edge energy and normalizing the height to obtain an area corresponding to the peak to Compton ratio function. The higher end of the Compton rectangle is smoothed by a Gaussian with the appropriate resolution. Zooming functions are available for easy and detailed view of the spectra obtained. The Y axis can be logarithmic or linear with selected amplification. The channel, energy and nuclide data are displayed according to the cursor position in the spectrum.

SAMPLE RESULTS

Two examples are given for a NaI(Tl) detector, to elucidate the advantages of using the GammaGen program for analyzing complicated radioisotope mixtures. In Figure 2 the case of one main peak (of \(^{95}\)Y) overlapping with other minor peaks is given. By ignoring the contribution of the smaller peaks, which mostly remain undetected in regular analysis, an error of about 30% can be induced when evaluating the activity of \(^{95}\)Y. In Figure 3 the case of two radioisotopes with very close energies, which can be easily distinguished by the different colors, is given.

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 hundreds of photo-peaks present in the case of a nuclear reactor accident, where a risk evaluation is to be made according to the fission products analysis and release fractions. The program is still under development, and additional features are intended to be added, as combining the calculated spectra with an existing measured background and adding statistical noise fluctuations.
Figure 1. The GammaGen program main screen in:
A) "Fission-Products" mode.
B) "Gamma-Nuclides" mode.
REFERENCES

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The Benefit of Adding Detectors in the Back when Measuring Heterogeneous and Homogenous Contamination in the Lungs

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INTRODUCTION

The evaluation of the activity of a radioactive source depends strongly on the distance between the source and the detector. After breathing an aerosol particle containing radioactive material, its location in the lungs is not known, and mostly the assumption is made that the source is distributed homogenously. However, this assumption can lead to errors of more than one order of magnitude, as was found by Kramer (1-2) and Pelled (3).

The use of modern lung counters with four detectors yields information which can be used to localize a point source of uranium contamination in the lung (4-5) and determine its activity more accurately. The method (6) is based on calibration with as many points as reasonable by inserting a radioactive point source in various points in a phantom, simulating the actual bulky material. In the case of a lung counter, this is a phantom of human lungs. For each calibrating point, every detector measures the count rate of each gamma line emitted by the source. The localization of the unknown radioactive point source is done by comparing the set of count rates for the unknown with the group of sets for the calibrating standard points. The mathematical problem is to find which of the sets of the standards fits best the set of the unknown. The values of the count rates of the different detectors are defined as the components of a vector of unit length. If two vectors are identical in all components, there will be a perfect "match".

Previous studies (4-5) showed that for point sources of natural Uranium positioned at various points inside a phantom of lungs, the accuracy in locating the source and calculating the source activity is quite good, specially if the lines of 92 and 186 keV are taken together to provide an eight dimension vector (4,5). In all configurations the detectors were placed in front of the phantom (on the chest). The purpose of the present work was to evaluate the benefit of placing detectors also in the back. The improvement in accuracy when adding a virtual point representing homogeneity was checked as well.

MATERIALS AND METHODS

The measurements were performed by the NRCN Lung Counter system, which consists of four Semi-Planar HPGe detectors, manufactured by Eurisy Mesures- France. The radioactive source was a sealed cylinder with a radius of 1.5 mm and length of 5 mm, which contained natural uranium with activity of 1.85kBq. The main γ lines are 92 keV due to the ²³⁸U chain and 186keV due to ²³⁵U and ²²⁶Ra. The information from each detector was analyzed separately by a multi-channel analyzer. The quantitative information from each measurement was the count rates of the 186 keV and 92 keV peaks. The detectors were calibrated using a realistic phantom designed by Lawrence Livermore National Laboratory and manufactured by Humanoid Ltd. The phantom lungs used were made of tissue equivalent plastic material and had 56 cylindrical holes, where point sources of natural uranium could be placed. The 56 points for placing the sources at different locations are distributed all over the lungs. 28 points are located on the upper surface of the lungs and 28 on the lower surface.
In order to check the effect of extra detectors in the back, we turned the phantom down, as shown in fig.1. The counts from the back were added to the vectors of the count rates from the front side, producing a 16 components vector (eight detectors, each one measuring two gamma-lines, 92 keV and 186 keV).

![Image of detectors positioned on the back of the phantom.](image)

Figure 1. The detectors positioned on the back of the phantom.

Various combinations of the detectors in the front and in the back were considered. The detectors in the front are assigned by A,B,C and D, while the detectors in the back are denoted by BA, BB, BC and BD. Each of the 56 standard points was measured four times, each measurement for 10,800s for both measurements with the detectors in front and in the back. Two countings of each point were used for constructing the library (taking the average) and the others two countings were used as an unknowns. For each of the 112 unknowns a “predicted point” was determined by finding the optimal vector, and comparing it to the actual point location. If there was a match between them, the program declared it a “successful hit”. The library of 56 standard points was expanded to 84 points for measurements with enhanced resolution, by adding an intermediate points plane between the upper and lower lung surface.

In order to check the match between the measurements and the database matrix, a computer application based on Microsoft.NET was developed and used. After determination of the “guessed” point location, the source activity was calculated from the counts of the source at this “guessed” point. The difference between the calculated activity for each single counting and the known source activity is the error for the specific measurement.

In addition to the effect of extra detectors in the back, also the heterogeneous distribution effect was checked, simulating different volume distributions by several combinations of point source measurements.
RESULTS
Table 1 presents the percentage of "hits" and the errors in the calculated activity of a point source contamination, both for using only the count-rates at 186 keV peak (vectors of four dimensions) and the simultaneous use the 92 keV and the 186 keV peaks. As can be seen from the table, positioning one or two detectors out of four in the back improves considerably the accuracy of the localization of the radioactive point source, as the percentage of the "hits" is increased from 79.8% to 91.5-97.8% and the error in the calculated activity decreases from 10.2% to 2.1-3.6% (when using two γ lines). Analyzing the various combinations of four detectors, it can be seen that the better combinations are when using detector BC in the back. The other three detectors can be positioned anywhere in the front. The use of more than four detectors increases the percentage of the "hits" by another 1-2% due to the larger total count-rates. The activity error is reduced too by the increase in the total number of the detectors. However, the benefit per cost of adding more detectors has little advantage for the percentage of hits, mainly in the case of two γ lines. Even in the case of one γ line the advantage of more than four detectors is of a marginal nature, as the average error in the observed activity decreases from 3.2% to 2.4% for five detectors and to about 1.6% for six detectors.
Table 1. The average percentage of "hits" and the average activity error for a point source contamination. The 56 points library was used.

<table>
<thead>
<tr>
<th>Combination number</th>
<th>Number of Detectors</th>
<th>Detectors</th>
<th>92keV+186keV</th>
<th>186keV</th>
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<tr>
<td></td>
<td></td>
<td></td>
<td>Number of hits (%)</td>
<td>Error in activity (%)</td>
</tr>
<tr>
<td>1</td>
<td>4</td>
<td>A, B, C, D</td>
<td>79.8±3.0</td>
<td>10.2</td>
</tr>
<tr>
<td>2</td>
<td>4</td>
<td>B, A, B, C, D</td>
<td>77.0±4.8</td>
<td>13.1</td>
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<tr>
<td>3</td>
<td>4</td>
<td>A, B, A, B, B</td>
<td>91.8±1.2</td>
<td>3.6</td>
</tr>
<tr>
<td>4</td>
<td>4</td>
<td>A, B, C, B, D</td>
<td>95.0±1.8</td>
<td>2.3</td>
</tr>
<tr>
<td>5</td>
<td>4</td>
<td>A, B, D, B A</td>
<td>93.7±0.5</td>
<td>3.0</td>
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<tr>
<td>6</td>
<td>4</td>
<td>A, B, D, C</td>
<td>97.8±1.8</td>
<td>2.1</td>
</tr>
<tr>
<td>7</td>
<td>4</td>
<td>A, B, C, B</td>
<td>92.3±1.0</td>
<td>3.4</td>
</tr>
<tr>
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<td>4</td>
<td>A, B, C, B D</td>
<td>95.5±1.9</td>
<td>2.3</td>
</tr>
<tr>
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<td>4</td>
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<td>92.7±1.4</td>
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<tr>
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<td>95.8±1.6</td>
<td>2.1</td>
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<tr>
<td>13</td>
<td>5</td>
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<td>97.2±2.1</td>
<td>2.1</td>
</tr>
<tr>
<td>14</td>
<td>5</td>
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<td>95.7±1.6</td>
<td>2.4</td>
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<td>15</td>
<td>5</td>
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<tr>
<td>16</td>
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<tr>
<td>17</td>
<td>5</td>
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<tr>
<td>18</td>
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<td>1.5</td>
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<tr>
<td>19</td>
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<td>1.8</td>
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<td>1.9</td>
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<td>21</td>
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<td>99.5±0.4</td>
<td>1.4</td>
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<tr>
<td>22</td>
<td>7</td>
<td>A, B, C, D, B B, B C, B D</td>
<td>99.7±0.5</td>
<td>1.2</td>
</tr>
<tr>
<td>23</td>
<td>7</td>
<td>A, B, C, D, B A, B B, B B D</td>
<td>99.9±0.3</td>
<td>1.6</td>
</tr>
<tr>
<td>24</td>
<td>7</td>
<td>A, B, C, D, B A, B B, B C, B D</td>
<td>99.5±0.5</td>
<td>1.3</td>
</tr>
</tbody>
</table>
Table 2. The matched point and the activity errors for various contaminated regions and for different detectors array (detectors in red were "on"). The 84 points library was used.

| Contaminated Region | A | B | C | D | A | B | C | D | A | B | C | D | A | B | C | D | A | B | C | D | A | B | C | D |
| Both lungs          | 31| 65.1| 39| 38.5| 9| 2.2| 39| 38.4| 39| 53.6|
| Right lung          | 48| 46.1| 82| 24.4| 82| 53.3| 82| 27.0| 82| 25.7|
| Left lung           | 13| 209.8| 69| 22.3| 69| 27.4| 69| 19.3| 69| 25.6|
| Half one of the right lung | 44| 113.9| 12| 26.7| 11| 11.9| 12| 30.0| 11| 13.6|
| Half two of the right lung | 13| 466.1| 65| 10.5| 65| 14.4| 62| 2.9| 65| 13.6|
| Half one of the left lung | 86| 48.8| 25| 7.0| 84| 3.6| 85| 7.4| 25| 7.4|
| Half two of the left lung | 46| 57.9| 48| 0.5| 48| 10.9| 47| 6.7| 48| 1.0|
| Cluster of 4 to 8 point sources in the left lung | Around Point 82 | 82| 1.5| 82| 2.5| 82| 1.4| 82| 0.8| 82| 3.3|
| Cluster of 4 to 8 point sources in the right lung | Around Point 85 | 86| 36.0| 25| 8.5| 86| 34.2| 85| 3.1| 25| 8.4|
| Cluster of 4 to 8 point sources in the right lung | Around Point 76 | 46| 27.8| 16| 9.1| 16| 8.8| 16| 9.5| 16| 9.0|
| Cluster of 4 to 8 point sources in the right lung | Around Point 62 | 10| 288.9| 62| 8.6| 62| 9.5| 62| 2.8| 62| 9.0|
| Cluster of 4 to 8 point sources in the right lung | Around Point 72 | 72| 1.7| 72| 0.6| 72| 3.6| 72| 5.1| 72| 2.8|
| Cluster of 4 to 8 point sources in the right lung | Around Point 63 | 65| 9.6| 65| 6.5| 65| 7.6| 65| 9.6| 65| 7.2|
| upper surface       | 39| 52.7| 39| 35.8| 9| 0.2| 39| 35.2| 39| 53.2|
| lower surface       | 31| 69.7| 39| 28.3| 39| 12.9| 39| 27.6| 39| 39.4|
| Intermediate surface | 31| 63.8| 39| 48.4| 9| 7.3| 39| 47.1| 39| 63.7|
| Including point 100  | 100| 0.1| 100| 0.1| 100| 0.2| 100| 0.1| 100| 0.1|
| upper surface       | 100| 1.7| 100| 2.0| 100| 2.0| 100| 2.3| 100| 2.0|
| lower surface       | 100| 5.4| 100| 5.2| 100| 2.9| 100| 4.0| 100| 6.3|
| Intermediate surface | 100| 3.7| 100| 7.2| 100| 5.0| 100| 6.4| 100| 6.6|
Table 2 shows the point location, which was calculated and matched by the software, for the various contaminated regions and the error in the activity calculation for different detectors array. In the case of homogenous contamination in both lungs, the errors of the activity evaluations are in the range of 65.1% to 2.2% in the case of an array of 4 detectors in the front and 4 detectors in the front and in the back respectively. When the volume of the contamination is reduced and becomes close to a point source geometry, the errors in the activity calculation are also reduced. For half lung we got a significant improvement, especially if we add another detector in the back. The errors are in range 3.6% to 14.4%, instead the range 48.8% to 466% when using an array of 4 detectors in the front. For a cluster of 4 to 8 points, the errors are in range 0.8% to 9.6%, when using detectors also in the back, instead of a range of 1.5% to 288.9% when using an array of 4 detectors only in the front.

In order to improve the accuracy of the activity determination of this method, in cases of homogenous contamination, an additional point was added to the library representing the case of homogenous distribution (we called this configuration point 100). The vector assigned to this point was the superposition of the response of the detectors to all the 84 points of the extended library. Table 2 shows also the points locations which were calculated and matched by the software for the various contaminated regions and the error of the activity calculation for different detectors array when using a library which contained also point 100.

The significant improvement in the error can be observed when using point 100.

CONCLUSIONS

For point and volume source distributions in the lungs, when using an array of 4 detectors, it will be advantageous if one of the detectors will be positioned in the back. Further improvement can be achieved by adding a virtual point to the library representing the homogenous case.

REFERENCES


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Dual Whole Body Counter: What Average to Use

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Introduction

Whole body counting (WBC) is external measurement of the $\gamma$ from radionuclides which are absorbed in the body tissues and have large biological half-life. Usually the counts due to contaminants are very small, the larger peak is due to the natural $^{40}\text{K}$, and in many cases two detectors, on the opposite sides of the human, are used in order to increase the count rate this is called dual WBC. When the radionuclides are dispersed homogenously in the body both detectors will give similar results and the sum of the counts is taken as the total counts from which the quantitative contamination is calculated. However, if the distribution in the body is heterogeneous and most of it is concentrated in one side of the body, the response of the two detectors will be different. Some people try to solve this problem by positioning the detectors far from the measured human. However, the larger distance caused reduction of the count rate, limiting the distance, and more than it, while the distancing of the detectors might solve the different angles from which the two detectors see the contaminant it do not solve the effect of different absorptions. In this case taking the sum of the counts of the two detectors is equivalent to calculating the activity from the arithmetic mean, which means assuming homogenous distribution. In the case of only absorption without geometric effects, as is the case in $\gamma$ cameras, it is common to calculate the geometric mean rather the arithmetic mean, since in the case of only parallel photon beams the geometric mean is invariant to the position of the source in the line between the two detectors$^{(1)}$. In the case of only geometric effect it was found that while geometric mean is better than the arithmetic one, the best is given by the square of the harmonic mean of the square roots of the counts$^{(2)}$. The question is what average should be used in the case of dual WBC where both geometric and absorption effects are involved. In this work all these averages were calculated for experimental results and for results from Monte-Carlo simulations of heterogeneously contaminated phantom.

Experimental setup

A rectangular 0.5 cm thick Perspex aquarium with dimensions of 46×20×20 cm$^3$ was used (46 cm simulates the width of a grown-up waist). The aquarium was filled with 3 cm Perspex slabs (used to simulate human tissue). At each end of the aquarium a 7.5×7.5 cm$^2$ NaI(Tl) detector was positioned, as shown in figure 1. The two detectors were operated simultaneously, each connected to a separate multi-channel analyzer.

Figure 1: The dual detectors measuring system.
The measurements were performed with the detectors positioned each at distances of 0, 5, 10, 15, 20 and 25 cm from the end of the box (shown as “d” in figure 1). d+h₀ in figure 1 is the distance between the aquarium and virtual point detector. Point sources of $^{60}$Co and $^{137}$Cs with γ photons of energy of 1173.2, 1332.5 and 661.6 keV were placed between the slabs at distances of 0, 3.75, 7.5, 11.25, 15, 18.75 and 22.5 cm from the center of the aquarium (shown as distance x in figure 1) for each distance of detector - box. The sources were placed on the center line between the two detectors. The entire system was protected with lead bricks to reduce the background. Only the photons which reached the detector without interaction with the phantom were used for calculations of the various means (only full energy peaks were measured). The areas under the peaks were computed by the trapezoid method used by MAESTRO software of ORTEC. In addition an Monte-Carlo simulation performed using the MCNP4C2 code system. The MCNP geometry model is based on experimental design. The geometry and materials of the detectors were based on typical commercial scintillation detector ($3\times3$). The full-energy peaks counts where collected for the same source positions as in the experimental setup. All the simulations were performed using $10^8$ histories for each run. The source energies used were defined as 1173.2, 1332.5 and 661.6 keV with a probability of 1/3 each. The detector count rate was simulated by using the MCNP Pulse height tallies (f8) for the cells that contain the NaI crystal. These tallies provide the energy distribution of pulses created in a detector. The energy bins were set between 0.03 and 1.5 MeV divided into 146 equal bins, hence the bin width is of 10 keV.

Results

There are two goals to this work. The first one is to find which kind of a mean (average) is the best one to use for counts collected by the two detectors, in a situation of heterogeneous distribution of the radionuclides with Dual WBC. The best mean is the one with the least variation of its value with different positions of the point source in the box. Three common means were calculated using the count rates of the two detectors $(N_1$ and $N_2$: the geometric mean $(MG)$, the arithmetic mean $(MA)$ and the harmonic mean $(MH)$ according to the following equations:

$$M_A = \frac{N_1 + N_2}{2}, \quad M_G = \sqrt{N_1 \cdot N_2}, \quad M_H = \frac{1}{2} \left( \frac{1}{N_1} + \frac{1}{N_2} \right)^{-1} = \frac{2N_1 \cdot N_2}{N_1 + N_2} = \frac{M_A^2}{M_A}$$

In addition two other means were calculated: the square of the harmonic mean of the square roots of the counts $(MSH)$, which was found to be invariant of the position of the contaminant in the case of low absorption, and the Alfassi–Presler (MAP) mean which is the square root of the product of the geometric and harmonic means, which was found to be almost invariant to the position of the contaminant in the case of low absorption. (Presler, Pelled et al. 2002):

$$M_{SSH} = \frac{4N_1 N_2}{(\sqrt{N_1} + \sqrt{N_2})^2}, \quad M_{MAP} = \sqrt{M_G \cdot N_H}.$$ 

The second goal is to find an optimal distance between the patient and the detectors. A large distance leads to lower count rates and hence to larger statistical error. Consequently, it is the best to use the smallest distance which will still lead to a mean which will be invariant to the position of the contamination.
These means were calculated for each of source positions when the two detectors where placed at various distances from the end of the aquarium. For a mean to be efficient it must deviated as little as possible from this value at the other positions of the source. The averages (arithmetic means) of the various means for the different positions of the source from the experimental setup and the Monte-Carlo simulations are given in Table 1. It can be seen from Table 1 that the best mean, from the five means described earlier, for all distances is the geometric mean, leading to error of at most 15% at distances above 10 cm. However, at small distances of detector-sample the error of the geometric mean is still quite large although much smaller than for the other means. The conclusion from these results is that the optimal condition is measurement at detector-sample distance of 10-15 cm calculating the geometric mean. The advantage of the geometric mean is explained theoretically using the concept of replacing the actual voluminous detectors by virtual point detectors (1,7). Using the concept that the a NaI(Tl) detector can be treated as a virtual point detector (7), the count-rates from a point source positioned inside the bulky sample at a distance x from one detector cap (figure 1), can be expressed by the equations (detector 1 is the closer one to the point source:

\[ C_i(x) = \frac{C(0) \cdot (a + d + h_0^2)}{(a - x + d + h_0^2)} \cdot e^{-\mu x} \quad ; \quad C_2(x) = \frac{C(0) \cdot (a + d + h_0^2)}{(a + x + d + h_0^2)} \cdot e^{-\mu x} \]  

(1)

Where \( C(0) \) is the count rate when the point source is positioned in the center of the cap. It is assumed that both detectors are the same and when the point source is in the center of the box; both detectors have the same count rate. \( h_0 \) is the distance from the detector cap to the virtual point detector and \( d \) is the distance between the box face and the detector caps (figure 1).
The geometric mean is given then:

\[ M_G = \left( C_1(x) \cdot C_2(x) \right) = \frac{C(0)}{x^2} \left[ 1 - \frac{1}{(a + d + h_0)^2} \right] \tag{2} \]

\[ M_G = \frac{C(0)}{C(0)} \left[ 1 - \frac{1}{(a + d + h_0)^2} \right] \tag{3} \]

The deviation of the normalized MG from unity is given by the second term in the denominator of equation (3). For large values of d, compared to the possible values of x, this is close to zero and the dependence on x is small. Thus for example for \( d = 15 \) cm, an average \( x, x = a/2 = 11.25 \) cm, \( a = 22.5 \) cm and \( h_0 = 3.75 \) cm (half of the detector thickness\(^2\)) will lead to the second term in the denominator being equal to \( (11.25/41.25)^2 = 0.0744 \), which means \( M_G/C(0) = 1.08 \) in very good agreement with the average values in table 1, 1.11 (662 keV), 1.08 (1173 keV), and 1.08 (1332 keV) in the experimental setup and 1.15 (662 keV), 1.13 (1173 keV) and 1.17 (1332 keV) in the MCNP simulations. For small d the denominator can be quite different from unity and \( M_G \) depends on x. For \( d = 0 \) cm, an average \( x, x = a/2 = 11.25 \) cm, will lead to the second term in the denominator being equal to \( (11.25/26.25)^2 = 0.184 \), which means \( M_G/C(0) = 1.23 \) in very good agreement with the values observed at \( d = 0 \) cm and \( x = \pm 11.25 \) cm from the MCNP simulations, 1.22 and 1.24 (662 keV), 1.21 and 1.22 (1173 keV) and 1.22 and 1.23 (1332keV). This fits with good agreement with the experimental setup at the same position (\( d = 0 \) cm and \( x = \pm 11.25 \) cm) 1.09 (662 keV), 1.18 (1173 keV) and 1.25 (1332keV). The standard deviations (in percents) of the means in Table 1 are given in Table 2. As can be seen from experimental setup there is a drop in the standard deviations down until 15 – 20 cm and form then a platter from then on. At lower distances there are larger errors due to the distance dependence, which is not exponential. A similar behavior can be absorbed in the MCNP simulations but not as extreme. As can be concluded from table 2 at distances above 15 cm the results of the geometric mean are quite accurate.
Table 2: The percentage of the standard deviations of the averages shown in table 1

<table>
<thead>
<tr>
<th>Distance</th>
<th>Experimental Setup</th>
<th>Monte Carlo Simulations</th>
</tr>
</thead>
<tbody>
<tr>
<td>Detector</td>
<td>$M_G$</td>
<td>$M_A$</td>
</tr>
<tr>
<td>0 cm</td>
<td>40.44</td>
<td>165.66</td>
</tr>
<tr>
<td>5 cm</td>
<td>34.02</td>
<td>138.59</td>
</tr>
<tr>
<td>10 cm</td>
<td>16.85</td>
<td>118.6</td>
</tr>
<tr>
<td>15 cm</td>
<td>11.93</td>
<td>108.96</td>
</tr>
<tr>
<td>20 cm</td>
<td>11.41</td>
<td>101.4</td>
</tr>
<tr>
<td>25 cm</td>
<td>13.93</td>
<td>95.61</td>
</tr>
<tr>
<td>1173.2</td>
<td>41.26</td>
<td>154.75</td>
</tr>
<tr>
<td>1332.5</td>
<td>33.5</td>
<td>124.4</td>
</tr>
<tr>
<td>10 cm</td>
<td>17.18</td>
<td>104.4</td>
</tr>
<tr>
<td>15 cm</td>
<td>15.23</td>
<td>94.05</td>
</tr>
<tr>
<td>20 cm</td>
<td>9.96</td>
<td>86.33</td>
</tr>
<tr>
<td>25 cm</td>
<td>5.19</td>
<td>77.74</td>
</tr>
</tbody>
</table>

Reference:


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Methods for Determining the Activity Concentration Calibration Factor for Ventilation Duct in Cyclotron Sites

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INTRODUCTION

Cyclotrons are commonly used for production of radioactive isotopes for nuclear medicine purposes. The nuclear process requires installation of a ventilation stack. According to the regulations in many countries, the released activity should be monitored and controlled in order to limit air pollution. This requires a stuck radiation detector and determining a converting factor that will translate the radiation detector reading into activity concentration units. Calibrating the conversion factor is done mainly by releasing a known amount of activity.

Having a preliminary estimation of the conversion factor, during the site construction stage, is an important option for the duct configuration design, in order to achieve the required detection sensitivity.

An algorithm for estimating the stack concentration calibration factor in positron emitting isotopes producing sites was developed. The algorithm which is described in this article is based on three independent methods that consist of MCNP simulations, analytical calculations and experimental setup with controlled static calibration releases of gaseous $^{18}$F which was produced especially for this purpose by the cyclotron site in Soreq NRC Israel. The specified methods applied on few stack sections with different shapes and sizes.

METHODS

Experimental procedures: An $^{18}$F gaseous source, with measured activity, is injected into a stack equipped with a fan on one side. After a few minutes the fan disperses the source homogeneously inside the stack. The stack section has a hole in the middle where the 2"x2" Nal(Tl) detector with its housing is inserted. Several differently shaped stacks were tested during the experiment with round, square and rectangular cross sections (Table 1). All the stack sections were 2m long.

There are two involved Nal(Tl) scintillation detectors in the calibration procedure; the duct detector, which is used for the activity measurement in the duct and the Dose Calibrator Simulator (DCS) which measures the injected activity. In order to find the calibration factor, on each injection level the following step should be done: measure background radiation value on DCS. Fill the injector with the radioactive gas. Measure the activity of the injector using Dose Calibrator. Measure the activity of the injector using DCS. Measure the activity of the duct before injection. Inject the gas into the duct and wait for stabilization and record the detector count rate inside the duct. Measure the injector using Dose Calibrator and DCS after injection for estimating the residual activity. Data sampling is done by using a microcontroller with a dedicated PC-based program which samples the detector pulses on predefined time, calculates and displays (Figure 1) the average count rate readings. Collected data is used to calculate the calibration factor.
Software simulation: The computer simulations presented in this report were obtained by using the general-purpose Monte Carlo N-Particle Transport code MCNP.\(^{(1)}\)

The geometry and material properties of the stack, the detector's housing and the scintillation detector itself have been fully and accurately modeled with the MCNP4c2 version (with the exception of the fan which has a minor significance due to its material properties and due to the gaseous source specificity).

For more precise results the source definitions were made in two steps. In the first step a positron source was simulated, and the final particles' distribution statistic was obtained, i.e. the annihilation location in the stack. In the second step a gamma source was simulated according to the results of the first step.

Analytical Calculations: The calculations were performed over stack sections with circular cross section geometry with a range of diameters from 30cm up to 100cm and 2m length. Three pertinent locations inside the stack were considered to contribute to the detector count rate: the air, the stack's boundaries and the detector's housing. Two different approaches had been applied and compared: a) integration of the total activity concentration inside the duct caused by homogeneously dispersed monoenergetic gamma source (511keV). b) Integration of the total activity concentration distributed as was obtained in the first step of the MCNP simulation i.e. using the annihilation location distribution inside the stack section. The following implemented formulas represent the calculated contribution to the detector count rate from the air cavity in the stack (1), stack's boundaries (2) and the detector surface (3):

\[
A(R) = \text{Const} \cdot \int \frac{2 \cdot \pi \cdot r}{(r^2 + x^2)} \cdot 4 \cdot \pi \, dr \, dx
\]

\[
B(R) = \text{Const} \cdot \int \frac{1}{(r^2 + x^2)} \cdot 4 \cdot \pi \, dx
\]

\[
C(R) = \text{Const}
\]

Where 'Const' consists of the detector surface area, efficiency, activity concentration distribution and geometrical elements; \(X\) and \(R\) are the stack length and cross section radii correspondingly.
RESULTS
The results that were obtained with the MCNP simulation and those that were received with the experimental setup are presented in Table 1 in the form of calibration conversion factors.
Table 1. Conversion factors for various stacks obtained with MCNP and experiments.

<table>
<thead>
<tr>
<th>Stack size &amp; shape</th>
<th>Factor (nCi/m³ per cps)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>MCNP results</td>
</tr>
<tr>
<td>36x36 cm</td>
<td>104.0</td>
</tr>
<tr>
<td>41x41 cm</td>
<td>88.6</td>
</tr>
<tr>
<td>46x46 cm</td>
<td>79.0</td>
</tr>
<tr>
<td>Ø30 cm</td>
<td>141.3</td>
</tr>
<tr>
<td>Ø40 cm</td>
<td>101.0</td>
</tr>
<tr>
<td>Ø50 cm</td>
<td>82.1</td>
</tr>
<tr>
<td>50x70 cm</td>
<td>61.1</td>
</tr>
</tbody>
</table>

The mathematical models that were obtained are presented in formulae (4), (5) and (6). A very good correlation could be seen between all three models.

(4) \[ F=35.22 \times S^{-0.52} \]

(5) \[ F=27.36 \times S^{-0.59} \]

(6) \[ a) \quad F=17.84 \times S^{-0.45} \]
\[ b) \quad F=26.6 \times S^{-0.81} \]

Where S corresponds to the stack’s cross section area and F is the concentration calibration factor, having the units of m² and nCi/m³ per cps correspondingly.
Formula (4) corresponds to a model obtained from the MCNP simulation. Formula (5) is obtained from the power regression approximation of the results from the experimental setup. Formulas (6) are from the analytical calculations: a) is the model without consideration of the positron annihilation process and distribution of positrons in stack – homogeneously spread gamma source with 511keV energy. b) is the model with using the positron annihilation location distribution as obtained from the MCNP.

All three models are displayed in Figure 2.

![Figure 2. Conversion Calibration Factor obtained from different models for various cross-section areas.](image)

**CONCLUSIONS**

The MCNP simulation has been performed on few square and round duct cross sections with different dimensions and the results were compared to the actual calibration releases. Results show a very high correlation between the MCNP simulation and the actual releases. Due to the high correlation between the simulation and the calibration releases, the simulation may now be used to estimate the design ventilation system sensitivity in order to improve the Minimum Detectable Activity and to be compatible with the restrict regulation (1) of detecting concentration level as low as 3nCi/m³.

A local calibration factor to adjust the software conversion factor to the self-characteristics of the specified detector is required. This adjustment is done by locating a calibrated radiation source in a pre-determined location and comparing between the expected reading and the detector actual reading.

This ability of determining the conversion factor minimizes the amount of accumulated activity releases during the calibration process.

The desired procedure to be performed in the future is estimating the local calibration factor for the cyclotron site in Soreq NRC Israel by using the developed algorithm and comparing it to the concentration calibration factor which will be obtained from dynamic controlled emission of ¹⁸F. So it could be used for reporting to the nuclear safety authorities and controlling the annual amount of released activity.
REFERENCES

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Development of a Radon Transport Through Soil Detection System

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INTRODUCTION
The soil is known as a major source through which radon gas is emitted to the atmosphere [1]. Radon gas transport through soil and its exhalation to air has been a major concern due to its implication on health: radon presents a significant cause of lung cancer [2]. Yet, this phenomenon has a positive aspect as well; radon can be used as a natural tracer to study underground gas concentrations and water transport [3]. In order to measure the radon transport parameters (diffusion, adsorption, convection, etc.) through different kinds of soils, an experimental Radon Detection System was developed. The goal of the experimental system is to produce an alpha particle counter which fulfills large detection area and high efficiency requirements. In this system commercial silicon solarcells were tested and found to be highly efficient as well as an inexpensive solution. Using a large amount of solarcells provides a large detection area, a fact that improves the system's efficiency as can be seen on the results section.

THE DETECTION SYSTEM DESIGN
The experimental system is based on mechanical columns, an alpha detection array, a gamma probe, charge amplifiers, a dedicated counting board and a standard PC. Three mechanical columns of 0.5m, 1m and 1.5m heights were designed in order to represent different soil depths. Each column is based on a 200 mm diameter, standard sewage PVC pipe, in which different kinds of soil can be filled for the experiments. The bottom of each column includes a valve entrance for allowing/stopping the Radon gas flow, a packing house for the radioactive source that produces the Radon gas, and a high density aluminum net (160 mesh, 0.1mm thick) mounted on a 4mm thick perforated stainless steel base (Figure 1). On the one hand, the high density aluminum net prevents sand from being contaminated by the radioactive source and on the other hand the perforated stainless steel base allows gas transport through it. The stainless steel base also provides a strong base that prevents mechanical fail due to the weight of the soil used in the experiment.
During each experiment, the detection head is mounted on the top of the selected column, inside a PVC sheath (Figure 2a), whose diameter fits the pipe's inner diameter. The detection head is used for mounting 20 solarcells PCBs (figure 2b) and a gamma detector. The charge amplifiers, which are connected to the solarcells PCBs via IP68 sealed d-type connectors (Figure 2b, 2c), are mounted and fastened to the detection head (Figure 2c). Radon gas leakage is prevented due to the use of o-rings and gaskets between the parts, providing a sealed system.
The mechanical stability of each column is achieved by using an iron stand (Figure 3). Due to the heavy weight of the highest column (1.5m) when filled with sand, it is planned to be attached to a wall by a specific designed stand, which enables the column rotation in order to empty the sand.

The most stable radon isotope is 222Rn which has a half-life time of 3.823 days. The decay of 222Rn is followed by emission of alpha particles (5.489 MeV with 0.9992 probability & 4.986 MeV with 0.00078 probability) and gamma radiation (512 keV with 0.00076 probability). In that case the 222Rn flux intensity is low. Therefore, alpha particles and gamma radiation must be measured, using the proper detectors.

**Gamma Counting:** Gamma measurements are carried out by the RP11 gamma radiation probe (manufactured by Rotem Industries), a 2"×2" NaI(Tl) scintillator.

**Alpha Counting:** The alpha detection array consists of 100 silicon solarcells of 1cm2 each. Every five solarcells are connected in parallel and installed on a separated PCB strip (see Figure 2b). A dedicated four-channel charge amplifier module (Figure 2c) was developed in order to amplify the signals received from the solarcell strips. Each channel includes a charge amplifier and a level discriminator that generates TTL pulses. Each module is connected to 20 solarcells.

A dedicated counting board, mounted on an external box, is used to sum up the five charge amplifier modules output and convert it into a single TTL pulse.

**Data logging:** Data logging is carried out with a standard PC. A serial communication line connects the PC to the RP11 gamma probe and to the microprocessor module that counts alpha particles.

**CONCLUSIONS**

An experimental system for detection of Radon transport through soil was developed. This system has a large detection area and it is designed to be sealed.

An experiment for approving the system's efficiency is planned to be done soon using a radon source.

**REFERENCES**


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INTRODUCTION
The safety of workers at nuclear plants is provided by many ways. One of them is by using a radiation monitoring system. The monitoring system continuously samples the output of numerous radiation detectors and compares the values to predefined thresholds. One of the tasks of the monitoring system is to activate a local alarm system whenever the radiation level exceeds the predefined threshold for a specific location.

When activated, the local alarm system initiates a siren sound and a blinking light to alert the operator in the area. The operator can silent the siren and stop the blinking of the light by pushing an 'acknowledgment button' placed on the front of the local alarm system. A steady light will still be on, until the radiation levels will drop beneath threshold and an operator will clear the alarm.

The local alarm system that is currently in use has a 'blind spot'. When the radiation level exceeds the first threshold and an operator pushes the 'acknowledgment button' the system remains in idle state. The siren is off, the warning light is steady and it waits for the drop of the radiation level. An increase of the radiation level above second threshold will not change the state of the system and the workers will not be alerted.

In order to overcome the 'blind spot' and to enhance the safety of the workers, an improved methodology for a novel local alarm system was developed.

Enhancing the safety is provided also by hardware modification. A small alpha numeric LCD display was added in the front panel of the system. The current level of the $\gamma$ radiation field is displayed to provide a continuous local reading of the radiation level.
SYSTEM DESCRIPTION
The novel local alarm system, shown in figure 1, includes a warning light, a buzzer, a small alpha numeric LCD display, which displays the current level of the γ radiation field, a push button to clear the alarm, another push button to check the system and a switch button for acknowledgement.

![Image of the novel local alarm system]

Figure 1. The novel local alarm system

METHODOLOGY DESCRIPTION
The new methodology of the system, as described by the following flowchart (figure 2), is based on the ability of the buzzer to produce two different siren sounds. A different sound is assigned for each threshold, making it easy to distinguish between the sirens caused by crossing the first threshold or the second one. The major improvement is made by adding another state to the system: instead of staying in idle state after an acknowledgement to the first threshold passing, the system is continuously sampling the radiation level and if it rises above the second threshold, the buzzer is activated producing a siren sound.
CONCLUSIONS
An improved methodology for a local alarm system, as presented in this paper, overcomes the 'blind spot' of the current system. Another enhancement of the system is achieved by using an LCD display that provides the workers a continuous reading of the local radiation level.

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Evaluation of Radiation Monitoring System

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INTRODUCTION
As a part of the continuous improvement policy of the NRCN, a novel radiation monitoring system is being developed (1). The challenge in developing such a system is to assure that the system provides safety improvement. In order to achieve this goal, a set of parameters was defined to estimate the safety performance of the current system. The values of these parameters will be used as a reference for a comparison between the current system and the developed one.

This paper describes the method for evaluating the defined parameters: stability and accuracy of the ionization current and the response time of the system.

SYSTEM DESCRIPTION
The major blocks of the current system are: an ion chamber, an electrometer and a radiation monitoring system as depicted in figure 1. The ion chamber converts γ radiation to currents in the range $10^{-7} \div 10^{-13}$ A. The radiation monitoring system converts this current to a negative voltage in the range of $-10 \div 0$ V respectively.

![Figure 1: Block Diagram of Current System](image-url)
WORK DESCRIPTION
The measurement of the mentioned parameters is conducted by using a highly precise current generator for simulating the current of the ion chamber and a data acquisition system using a 16 bit ADC connected to the output. Figure 2 presents the measurement system.

![Figure 2: Block Diagram of the Measurement System](image)

In order to improve the reliability of our test we used different sets of preamplifiers and amplifiers calibrating every set before testing. The use of a highly precise current source as a simulator to the ionization chamber provided the ability to control the input current, which represents the radiation field. The results were recorded on a data acquisition system using a 16 bit ADC ensuring that a sufficient data was obtained for a reliable statistical analysis. All the added cables were shielded to avoid electrical interference and the acquisitions rate was kept as low as 10Hz.

TEST RESULTS
The stability and accuracy were obtained by injecting currents in discrete values covering all the input range. The response time was evaluated by increasing and decreasing the current by an order of magnitude in a rapid change.

Figure 3 presents the system behavior to a change from $10^{-12}$A to $10^{-11}$A in the ionization current. The system time response is 3.5 sec up to 1% accuracy. The framed segment in Figure 3 is the stable value zone which was used to evaluate the stability and accuracy of the measurements.

![Figure 3: The system response time](image)
Stability Measurement
The output voltage values of the injected ionization discrete current (in the range $10^{-7} \div 10^{13}$A) were converted back to current according to equation (1).

$$I = e^{\frac{-V}{0.7238241}} \frac{1}{10^{13}}$$

For every discrete value, sufficient data was acquired to calculate the mean and standard deviation, STD, which represents the system stability. Figure 4 presents the STD of the converted currents vs. the input currents for each experiment.

Figure 4: STD of "converted Ionization current"

Accuracy Measurement
The accuracy was calculated by measuring the output voltage for the applied discrete current values in the range of $10^{-7} \div 10^{13}$A. The expected correlation between the input current and the output voltage is described in equation (2).

$$V_{calc} = -0.7238241 \cdot \ln(10^{13} \cdot I)$$

$I$ - Input current, $V_{calc}$ - Output expected value (Volts)

The error in the measured output voltage is defined in equation (3).

$$Error_v = \frac{V_{calc} - V_{meas}}{V_{fullscale}}$$

$V_{meas}$ - Measured output value (Volts), $V_{fullscale}$ - maximum output range (10 Volts)
Figure 5 presents the error calculated for each current.

**Figure 5: Error in current conversion**

**Response Time Measurement**

The response time was evaluated by increasing and decreasing the current by an order of magnitude in a rapid change as described in figure 3.

Figure 6 presents the system response time for each ionization current step.

**Figure 6: Response Time to a step up ionization current**

Figure 6 shows that the response time decreases exponentially while the input current increases.
CONCLUSIONS
A concrete method for evaluating the performance of a radiation monitoring system was presented. Using this method will ensure that the safety performances of the developed radiation monitoring system is adequate.

REFERENCES


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Aerial Radiation Detection Vehicle Manned and Unmanned Concepts

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We are developing an Unmanned Aerial Radiation Detection Vehicle that will give new abilities to the Manned Aerial Radiation Detection Vehicle, Air-Ram. A comparison between the two systems will be given, and a report to our first Unmanned Aerial Radiation Detection Vehicle flight.

Air-Ram

The Air-Ram system, figure 1, has been developed to measure and display online radiation level measurements taken above the radiation area with a chopper. The detected radiation levels are presented on a topographical map with the flight path colored with the radiation intensities. The air crew and controllers on the ground are updated every two seconds. It enables first responders to complete and real time picture of a radiological event which is essential in order to be able to activate and direct ground operations if necessary.

The system measures radiation levels and produces a spectrum graph used to identify the isotopes.

Applications

- Plume identification.
- Risk Assessment following radioactive release into the atmosphere.
- Large area surveys to identify radiation contamination boundaries.
- Routine surveys of large nuclear designated areas.

The AIR-RAM contains two 2”2” NaI(Tl) scintillation detectors and a Geiger detector. The two scintillation detectors are positioned one above the other with a 3 cm lead plate separating between them. Using these detectors, the determination of the plume height, axis and trajectories are achieved. In addition to the scintillation detectors, a Geiger detector is included for attaining the dose rates to which the aircraft crew has been exposed.

Advanced alarming mechanisms provide the crew with crucial information regarding their task. For each scenario, whether it is a plume or a surface-monitoring task, the operator is guided by the computer system in every step.
Figure 1: Air-Ram 2000 system

RaD-UAV
“Caspar” – A Radiation Detection Unmanned Aerial Vehicle (RaD-UAV).

We based our system on the known "Caspar" Unmanned Aerial Vehicle, developed by "TOP i Vision", figure 2.
The motivation for Unmanned Aerial Vehicle is to lower the risk for the air crew and to shorten the time response to activate the system.

Figure 2: “Caspar” – A Radiation Detection Unmanned Aerial Vehicle (RaD-UAV)
The tough limitation is the low payload we can carry on our Unmanned Aerial Vehicle.

We used 22x28 mm2 CsI(Tl) detector, based on the PDS model, figure 3. The detector readings are transmitted online to a ground base station, together with a GPS location signal.

![Image of radiation detector](image_url)

**Figure 3:** The radiation detector PDS100 N
We can see in figure 4 the two sources of 0.5 and 5 Ci detected on the topographical map, it is integral of measurements at heights 100-200m. (second test flight on 2/2007)

**Figure 4:** Large area surveys to identify the two radiations source with Radiation Detection Unmanned Aerial Vehicle (RaD-UAV).

The intensity of the two radiations sources with Radiation Detection Unmanned Aerial Vehicle (RaD-UAV), figure 5 for 0.5 and 5 Ci.

**Figure 5:** The intensity of the two radiations source with Radiation Detection Unmanned Aerial Vehicle (RaD-UAV).
Base on the field test results, a detection limit (of 3 sigma above background) was calculated for $^{192}\text{Ir}$, $^{137}\text{Cs}$ and $^{60}\text{Co}$ radiation sources, figure 6.

**Figure 6:** The detection limit (of 3 sigma above background) was calculated for $^{192}\text{Ir}$, $^{137}\text{Cs}$ and $^{60}\text{Co}$ radiation sources.

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A Novel Radiation Monitoring System

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INTRODUCTION
An ultra reliable radiation monitoring system is essential in order to provide safety to the workers in nuclear plants. The system should be able to detect minor fluctuations of the radiation field and activate an alarm when the radiation field exceeds predefined thresholds. The system implementation requires qualification for nuclear applications such as IEEE 323\(^{(1)}\) and IEC 61503\(^{(2)}\). In this paper such system is presented, it is a novel modular system that provides reliable monitoring of \(\gamma\) radiation field. Although such systems were developed in the past \(^{(3)(4)(5)}\), the presented system is based on modern electronics and 'went the extra mile' to provide a very high reliability achieved by using sophisticated methods of BIT (Built In Test) and feedback loops.

SYSTEM DESCRIPTION
The major blocks of the novel monitoring system are depicted in figure 1:

![Figure 1. Block Diagram of the monitoring system.](image-url)
The major system units are:

- **Ion Chamber** – detector that converts γ radiation field to electrical current.
- **Electrometer** - provides voltage to the ion chamber and converts electrical current from the ion chamber to a more robust, 4-20mA current. The electrometer is able to measure input currents from $10^{-13}$ to $10^{-7}$A.
- **Process Unit** – receives 4-20mA from the electrometer, analyses radiation levels and transfers the data to the communication unit via RS-232, to the local alarm panel via RS-485 and to the isolated analog output via an analog voltage. If radiation levels exceed predefined thresholds, several measures are taken; an alarm is activated in the local alarm panel, the relays change state and the communication unit is being notified.
- **Communication Unit** – communicates with the process unit, provides the information to the Terminals and handles the HMI that includes alpha numerical screen, push buttons and indication LEDs.
- **Master PC** – a computer with dedicated software that receives the γ radiation levels from the communication unit and displays continuous graphical presentation of the radiation levels. The user can also control system parameters such as thresholds, unit's addresses, H.V. levels, etc. via the PC.
- **Local Alarm Panel** - contains a small alpha numeric LCD display, which displays the current level of the γ radiation field, a buzzer, a lamp and push buttons to clear the alarm. Whenever the γ radiation field passes a pre defined thresholds the lamp and buzzer are activated.
- **Relays** – changes state when the γ radiation field passes a threshold, each relay is activated at a different threshold. The relays provide an input of dry contacts for programmable controllers.
- **Isolated Analog Output** – a DC voltage between 0 and 10V, proportional to the radiation level. The voltage is isolated to prevent any interference with the equipment sampling the voltage. A feedback circuit is used to ensure that the analog voltage is accurate and correct.
- **HMI** - includes alpha numerical screen, push buttons and indication LEDs that allows the user to watch parameters and γ radiation levels locally.

The system has a lot of feedback loops that allows detection of failure in the system. All the power supplies are sampled, the H.V. is monitored continuously, the analog voltage output is sampled and compared to the transmitted value and the communication channels are used like a watchdog to ensure the aliveness of the system.

The system is also equipped with a sophisticated BIT; the system is capable to inject current to the electrometer and to check all the circuits by measuring the response to the injected current.
CONCLUSIONS
As indicated in the block diagram, the monitoring system has a lot of feedback methods and redundant sub-systems that increases the reliability enormously, when combined with a comprehensive BIT procedures the monitoring system becomes an ultra reliable monitoring system.

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Extensive Environmental Radiation Monitoring Portable Stations

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INTRODUCTION

A novel approach (1) for Environmental-Radiation-Monitoring-Portable-Stations is being presented. Each station contains a proven RAM R-200 (2) Survey Meter, a cellular modem, GPS and an optional ultrasonic wind-sensor. The system consists of control centers and stations array. Each station can function either as mobile or as fixed, once it is deployed. Flexibility is a prominent feature of the system since it is required to consent with the need to fit into any state of premise with minimal installation time. The portability is achieved by a convenient suitcase-like packaging (see Figure 1), enabling rapid response when needed - up to 5 minutes for an untrained operator.

Figure 1. A portable station
THE SYSTEM FEATURES

The main and topmost feature of the system is its ability to monitor continuous gamma radiation fields using remote stations. In the event of emergency, transportable stations are deployed according to the wind direction while a control center contacts all stations, retrieves radiation and meteorological readings and stores the analyzed data in a database. The control center consists of a local graphical-user-interface (GUI), presenting the stations and their related data on a digital map.

THE STATION MAIN FEATURES

Each station (Figure 2) is featured by:

- Radioactivity detection:
  - Wide measurement range: 0.1 μSv/h to 1 Sv/h
  - Accuracy: 95% of readings are in the range of 15% accuracy
  - Energetic response: ±25% at 16 keV to 13 MeV energy range
  - High sensitivity: 1000 cpm/(μSv/h) → (0.1 μSv/h ÷ 200 μSv/h)
    10 cpm/(μSv/h) → (200 μSv/h ÷ 1 Sv/h)

- Case features:
  - Light color reduces heating by solar radiation
  - The station can be elevated up to 1 meter above ground-level using a tripod, thus enabling higher positioning of the radiation detector and reducing heat transport from the hot soil. The elevated station can be stabilized with dedicated wedges
  - External magnetic antenna allows better reception/transmission with lower energy consumption, as well as compact storage ability
  - Meets IP65 standard by using a sealed case, o-rings and gaskets
  - Can be easily cleaned in case of radiological contamination
  - Operating temperatures -20°C to 80°C

- GPS for automatic position acquisition. Allowing tracking of portable stations

- Internal cellular modem - the current modem supports iDEN™ IP Network, but it can be easily replaced with any other modem supporting RS232 interface. The system utilizes a Protocols-Adapter which dials and creates Point-to-Point-Protocol (PPP) connection and User-Datagram-Protocol (UDP) packets to be sent via the modem

- Optional connection to a robust ultrasonic wind sensor that provides wind speed and direction

- A specialized version of RAM R-200(2) Survey Meter, consisting of 2 internal Geiger Mueller tubes, enabling detection of gamma fields in the range of 10μR/h – 100R/h. The RAM R-200 can be interfaced with a wide range of external detectors such as RG12 (β and γ contamination probe) or RP11 (scintillation detector for low γ fields measurement)
Non-volatile flash memory for saving parameters and for future analysis of stored data records
- Internal clock and calendar
- LCD for local radiation field display
- Sealed LED’s for simple local alerts and alarms
- General purpose digital and analog inputs, relay outputs for future use (for instance, meteorological instruments or external remote pumps activation)

Each station supports the following communication interfaces:
- Internal communication channels to:
  - GPS
  - Internal Detector (RAM R-200 internal Geiger Mueller tubes)
  - Cellular modem (through the protocol adapter)
- External communication interfaces:
  - Direct link to a PC for local maintenance
  - External modem
  - External detector (through the internal RAM R-200)
  - Wind sensor (velocity and direction)

POWER SUPPLY
The stations automatically switch between internal batteries and external DC power supply, depending upon voltage level. The internal batteries array is composed of three 3.6V Li-Ion non-rechargeable batteries/cells, with a capacity of 19Ah providing long shelf life. Intermediate rechargeable 3.6V Hybrid-Layer-Capacitor (HLC) batteries provide current pulses while the modem is in the transmission mode. This internal batteries array enables systems’ autonomy for more than 24 hours at full operational mode, meaning data transmission rate of a full message every 2 minutes.

The external power supply should be any source providing 11-24 V_{DC}, such as solar-cells, AC adapter (220 V_{AC} \rightarrow 12 V_{DC}), etc.
Figure 2. Block Diagram of a station.

SUMMARY

Designed for environmental monitoring in routine and emergencies, the proposed system provides an excellent solution by utilizing a flexible and easily deployed array of stations, independent of surface roughness or weather changes. Still, in view of the increasing threat of radiological terror, the system seems to provide an adequate answer as well. The stations flexibility allows a fast emergency-enforcement of an existing network, by immediate deployment of additional stations according to need. Currently, the system is in its final development phase, and is expected to be operational by the mid of 2008.

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**Environmental Radiation Monitoring System Control Center**

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**Introduction**

An Environmental Radiation Monitoring System has been developed and is being tested in the NRCN to be an auxiliary tool in case of a nuclear accident. The system collects real-time environmental data such as: gamma radiation, wind speed and direction, and temperature. The system also stores malfunctions and status of the data loggers. This is done by the main server, or the secondary server which is the backup. The collected data from the stations is stored in a local database for future processing. Users have a Graphical User Interface (GUI) as front-end display that presents the collected data on a geographic map (GEO). Radiation values are colored according to predefined threshold levels. The GUI enables users to view various types of graphs and reports over different cross sections. In addition, parameters like radiation thresholds and sampling time, that impact the station operation modes are defined via GUI and uploaded to the stations.

**System block diagram**

The Environmental Monitoring System consists of a series of measurement stations. Each station includes radiation monitor consist of a serial communication line (RS-232) for the external sensors. The measurement station data is transmitted to the main server via Virtual Private Network (VPN) Ethernet protocol. There are two options to transmit the data, via data cellular network, at an average speed of 600 Byte/Sec or via Digital Subscriber Line (DSL), at a speed of 256kBits/sec. The main server collects and stores the data on the Server Query Language Data Base (SQL-DB). An internal Ethernet link transfers the data to the GUI.
Measurement Station General Description
Each measurement station consists of: gamma detector, GPS, wind direction and speed sensor, temperature measurement unit and communication channels. The station collects the environmental data, stores it in a non-volatile memory (up to one month) and sends it according to the server request. The communication channel, which enables server connection to the stations, is implemented with mobile or point to multi-point DSL communication channels. The station receives commands (i.e. get data, get history data, reset station, etc.) from the server and responds back.

Maintenance Software Module description
The station operation mode is dictated according to the parameters saved in the non-volatile flash memory. The software module is used to modify all the station parameters: name, location, peripheral configuration, data reading and display on graph (without saving on a local data base), radiation units (Sv or Rem) setting, etc. In addition, most of the parameters can be remotely modified over the communication channels, such as: radiation threshold, Build In Test (BIT) commands, alarm dose rates and accumulated dose thresholds setting. The maintenance software works locally using RS-232 serial line.

Data Collector Module Description
The data collector module operates on the main or secondary server. Its main task is to periodically request environmental data from the stations and to save the data for future processing. Data is collected by cellular communication or DSL channels. The collector module polls the station and retrieves data through the User Datagram Protocol (UDP). In addition, it can send other commands such as threshold changes. The collected data is saved on SQL DB and in the cache memory for long and short data length request. The data collector module main functions are:

- Establishing connections with the stations
- Polling and Collecting radiation and environmental data from the stations
- Obtaining stations' status by retrieving BIT results
- Managing SQL database
- Sending alarms to GUI
- Sending short time history to the GUI
- Retrieving history to the GUI (using SQL query)
- Sending reset commands to the stations
- Synchronizing stations time
- Synchronizing secondary server data
- SQL backup

Collecting Data from the Stations
The system is designed so that over 50 stations can be polled in one server. The maximum number of polled stations will be derived from the communication channels bandwidth limitation. The collector module sends a request to the station every one minute. The poll interval is equally distributed among the stations (for example, for 30 stations the collector will poll each station every 2 seconds). The collector is NOT waiting for response from the station; however, it waits for reply from all the stations (synchronous vs. asynchronous method). This a-synchronic method offers the advantage of increasing the number of polled stations despite of the 3-second delay in the cellular communication when sending the first byte and any station can send bursting messages in case of threshold exceeding or increase its data sampling rate.
A timer for each station is started when a data request is sent and resets when the station replies. If data does not arrive after a predefined time the collector labels this station as unreachable. In this case, after reestablishing the connection, the missing history data, if exists, will be requested from the station and will be saved on the server database.

**Data Base**

The DB is implemented using SQL on the main and secondary servers. The whole collected data is saved on the database. This data includes the following tables:

- Stations details
- Radiation and environmental readings from stations sorted by location
- Malfunction log associated to the station
- Events log (on, off, establish connection, disconnection, etc.)
- Threshold changes log (who changed and when)
- List of users and privileges

**Transmit\Receive Rate Calculations**

Cellular transmit\receive bandwidth is around 600 Bytes/Sec or 36000 Bytes/Min. In case that 50 stations are polled every minute, the received bandwidth will be 3500 Bytes/Min, which is one magnitude order smaller than the maximum bandwidth.

The fast Ethernet, from the GUI to the server, is 100MBpS which is much larger than the actual transmit/receive bandwidth. In case of pulling 120kByte graph data per station every 5 min from a server, the data amount will be two magnitudes order less than the maximal bandwidth which is 100MBpS. This calculation demonstrates that there is no bandwidth limitation concerning the stations and GUIs maximum capacity.

**Graphic User Interface (GUI)**

![GUI Station 1](image)
The GUI software is the user's front-end. This module communicates with the collector main server or secondary one, retrieves data and displays it. The collected data can be classified by groups on a map or on a real-time graph. The UDP is the communication channel over fast Ethernet line. It is possible to communicate with the server via DSL channels but performance will be around 1% relative to the fast Ethernet. However, since the amount of requested data is relatively small, the GUI performance is acceptable.

The GUI software performs the following tasks:

- GEO representation of data received from the stations. The software main menu is a map with easy access buttons
- Standard GEO operations such as: Zoom In, Zoom Out, Pan and Rotate
- Data is updated in real time, including point color, according to predefined radiation thresholds
- Vocal alarms in case of malfunction or threshold level exceeding, even when the station is not displayed on map
- Allows the user to see the collected data on grouped stations graphs
- Cross-section reports: station maintenance and malfunction, and radiation according to location for a period of time
- Export data to EXCEL or text files
- Privileged operations such as station threshold changes

The Simulator Program

This presented program simulates the stations' dose rate readings as function of time and location. These dose rate values are used for practicing the GUI operation. The communication interface between the GUI and the simulator is the same as the interface between the GUI and the data collector. All the dose rate values are calculated according to a predefined script that simulates a radioactive plume movement. The simulator also sends malfunctions to the stations randomly.

The simulator main properties are: simulation of a dose rate field as function of time and location according to predefined wind speed, direction and stability condition, and simulation of station malfunctions.

Summary

In this work, we presented a full control center structure and software features of the Environmental Radiation Monitoring System. Data from the stations is transferred via different communication channels to the server that stores and sends it to the GUI station upon request. The GUI displays the collected data on GEO map or graph. The Environmental Radiation Monitoring System can be easily expanded to support different types of stations like meteorology or any other data logger.

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New Scoring System and a Novel Method of Plan Quality Evaluation and Comparison

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OBJECTIVES: We propose a two-dimensional plan-evaluation tool, allowing visualization of target coverage, and critical organ sparing and dose conformity in a single plot. The new approach integrates both volumetric and dosimetric information to review all relevant information for plan quality evaluation for several plans simultaneously.

METHODS: Treatment plans were created to evaluate the differences in dose distribution for Dynamic Arcs (DA) and Intensity modulated fields for 17 patients with Cavernous Sinus Meningioma. Optic nerves were considered Organ at Risk (OAR) with dose limited to 1.8Gy/fraction. DA was delivered with 4 arcs and IMRT with 12 beams to 50.4Gy, in 1.8Gy/fraction were prescribed to the 85% isodose line.

We implemented a novel scoring system for target coverage and OAR sparing, based on Dose Volume Histograms (DVH). General formulation for the SCORE is defined as

\[
\text{FULL \_SCORE} = \text{Target\_Score} - \text{OAR\_Score},
\]

where

\[
\text{Target\_Score} = 1 - (1 - \frac{V_{ptv}}{P_{TV_{min}}}) \times \frac{P_{TV_{min}}}{P_{D}}
\]

and

\[
\text{OAR\_Score} = \frac{V_{OAR}}{P_{D}} \times \frac{O_{AR_{max}} - O_{AR_{tol}}}{P_{D}},
\]

where

- \(V_{ptv}\) - the partial target volume, receiving at least the prescribed dose,
- \(P_{TV_{min}}\) - minimum target dose,
- \(P_{D}\) - prescription dose level,
- \(V_{OAR}\) - partial volume of OAR receiving a dose higher than the defined tolerance level,
- \(O_{AR_{max}}\) - OAR maximum dose,
- \(O_{AR_{tol}}\) - OAR tolerance level.

All these terms have simple geometrical meaning on DVH graphs. We present a new 2D graphical representation of SCORE's system versus Conformity Index (CI, Lomax), which improves our ability to assess trade-offs of the different plans in a simple visual way. For each plan we propose a 2 points graph: Target\_Score and Full\_score versus CI. The length of the line between the two points and its position is correlated to the effectiveness of Target coverage and OAR's sparing. The comparison analysis includes both dose and volume-related target coverage, OAR sparing and overall plan conformity.
RESULTS: The proposed evaluation SCORE's system for quantitative plans comparison was tested and showed a good correlation to target coverage indices, being highly efficient for the evaluation of a specific OAR sparing. SCOREs-CI plots revealed that the IMRT technique was superior to DA in 47% of cases, and inferior in 17.5% of cases. The remainder would be adequately treated with either approach. Conclusions: The proposed method distinguishes between plans in terms of target coverage, conformity and OAR sparing, integrating volumetric and dosimetric information. This SCORING Method is general, can be implemented for several OARs and provides the physician with a time-efficient method for plan evaluation and comparison.

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A Comparison Study of 10MV Unflattened and Flattened (therapeutic) Beams

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Cancer disease is the second leading cause of death, after heart diseases in many countries all over the world, and in some western countries it's even in the first place. Eleven million new cancer patients appear every year and 12.5% of all natural death cases in the world (7 million people) are caused by cancer diseases. According to the UICC (Union Internationale Contre le Cancer = International Union Against Cancer), at 2020, 16 million people will be diagnosed as cancer patients. Those numbers have strong correlation to the continuous increase of the average life expectancy of the world population. In the western world countries (including Israel) 20% of all the death events are caused by cancer diseases and it is related to factors like: wrong nutrition, air pollution, and increase of life expectancy. In Israel there are 23,500 new cancer patients every year.

The main ways to treat cancer are: surgical treatment, radiation treatment, hormonal treatment, and biological treatment that are based on action of antibodies which encourage the immunization system to fight against cancer cells. Up to 60% of all the cancer patients will receive radiation therapy as part of their treatment. This is why in the last few years there is continuous effort in introduction of new radiation techniques combined with new imaging techniques, to improve the radiation treatment quality.

Usually cancer cells are more sensitive to radiation than healthy cells, and by exposing them to high doses of ionizing radiation, we try to kill them without destroying the healthy tissues and healthy organs around the tumor. In teletherapy (external beam radiation) a relative damage to healthy organs and tissues is caused in most of the cases, because of their position near the tumor or in the external beam path. Because of these reasons, considerable importance is given to treatment accuracy. From this point of view, the issue of portal imaging plays an important role in quality assurance in radiotherapy.

Every patient starting radiation treatment needs to undergo portal imaging before the first treatment. Portal image is defined as any patient image taken with the accelerator beam. It can be used to detect errors in patient position, field size, MLC (Multileaf Collimator) location, and other important parameters. The portal image can be done a number of times during the treatment of the patient, depending on the physician decision.

It is a very important tool to assure that the irradiated parts of the body will match exactly to the original treatment planning prescribed for the patient.

There are two ways to perform a portal image: the first one is doing it with special films, and the second one is doing it with DPI (Digital Portal Imaging) system that includes a digital film cassette. As mentioned the portal images obtained with the therapeutic X-ray beam of the accelerator. The portal images taken with the accelerator beam are suffered from insufficient quality because of low contrast between the bones and the other tissues in the body. This problem is due to the high energy range of the beam (4-25MeV). This interval of photon energies is necessary to cover the whole range of human thicknesses. In this range of photon energy, the mass attenuation coefficients for bone and soft tissue are nearly the same (the main interaction mechanism is Compton process) resulting in very low contrast and insufficient sharpness observed in portal images.
In the low photons energy (≤ 150KeV) the main interaction process is photoelectric effect which is strongly dependent on the atomic number of the materials (Z). This, in turn, produces a sharp image with good contrast between the bones and the soft tissue in the low energy range of photons.

In this work we have carried out a comprehensive comparison between the 10MV therapeutic flattened X-ray beam and its unflattened counterpart beam (without the beam flattening filter which is placed in the path of the therapeutic beam). This work was performed at Soroka University Medical Center, with Clinac 18 accelerator (Varian).

The main purpose of this comparison was to give an answer to the question:
Do the diagnostic low energy photons (≤ 150KeV) survive the self- absorption in the thick Cu target of the accelerator? Until recently it was accepted to assume that the low energy photons which do exist in the X-ray spectrum are absorbed in the thick Cu target (4.7mm) and in the beam flattening filter (~ 7cm copper cone).

The role of the beam flattening filter (made of Cu) in the accelerator is to flatten the X-ray beam that comes out from the target being strongly peaked out in the center of the beam and to create an uniform dose around the radiation field (accuracy of 2%).

To check the possible existence of the low energy photons and their relative contribution to the quality of the portal images, we installed in the accelerator a special port for imaging without flattening filter (in this way we have created an unflattened 10MV X-ray beam).

In this work, we present for the first time comprehensive comparison between the 10MV flattened and 10MV unflattened beams in the same accelerator (with original mode of 10MV X-ray flattened beam). As a result of these unique conditions, we have managed to perform a reliable and comprehensive comparison between the two beam modes.

In order to prove the existence of those low energy photons in the unflattened beam, and estimate their relative contribution to the portal image quality we have used a unique irradiation method (developed by researchers from Soroka University Medical Center and Ben-Gurion University) [1].

In this irradiation technique called - one shot absorption measurement we irradiated an EC-L film-screen-cassette combination placed at distance of 35cm from a thin polystyrene plate. On this plate a group of 12 Pb foils with thicknesses varying from 0.25 to 3 mm in step of 0.25 mm arranged symmetrically around the central axis on a 7.2 cm radius circumference. As mentioned before, we have used for this experiment a Kodak EC-L film-screen-cassette combination as a detector in the absorption measurement.

After the one shot absorption film irradiation, the optical density (OD) of the film was measured in the places the Pb absorbers were placed, and the optical density readings were plotted as function of the Pb absorber thicknesses. By using this method we have proved without any doubt, the existence of the low energy photons in the 10MV unflattened X-ray beam and their strong potential to improve the portal image quality.

Another stage of the work was to lower the energy of the new imaging port from 10MV unflattened X-ray beam to 6MV unflattened X-ray beam. This change was performed to investigate if there is a change in the relative contribution of the low energy photons on the portal film, and what will be the influence to the portal image quality when we reduce the beam energy for unflattened beam.
In order to make comprehensive comparison between the three beam modes (10MV flattened and unflattened, 6MV unflattened), we have done at the first stage comparison between PDD curves and profiles (using the Wellhöfer radiation field analyzer system).

At the second stage, we have done a full comparison between the different sensitometric curves. As a result of the comparison between the sensitometric curves of 10MV flattened beam and 10MV unflattened beam, we got clear shift between the two curves. This shift between the curves disappeared when a 0.8 mm thick Pb absorber sheet intercepting the 10MV unflattened beam eliminates almost totally the present of the low energy photons. At the third stage we have done a comprehensive comparison between the absorption curves (that we got by using the one shot absorption measurement method) of the 3 beam modes.

As a result of this comprehensive comparison it was concluded that there is a "benefit" of low energy photons in the unflattened beam (10MV and 6MV unflattened), and we also have managed to estimate their relative contribution to the portal image quality.

In order to perform comparison between the clinical portal images obtained with the three beam modes, we used the Rando anthropomorphic phantom (The Phantom Laboratory). When we used the unflattened beam, improvements in the portal film quality were obtained, due to the participation of the low energy photons that exist in the unflattened beam.

We also have done comparison between portal images of the Las Vegas phantom taken with the three different beam modes.

In the last years there is a growing importance of the accuracy in the radiotherapy treatments, because of the advent of radiation techniques based on small fields like 3D Conformal Radiotherapy and IMRT (Intensity Modulated Radiotherapy). With these new techniques it is possible to increase the total dose to the tumor without damaging too much the organs at risk. Using the unflattened beam for portal imaging can help to physician to examine the portal images and be sure about the location of the radiation fields the patient gets.

The use of unflattened beam to make portal images has good potential to improve the portal image quality with minimum efforts.
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A Genetic Optimizer for Intra-operative Mid-course Correction of a Prostate Implant

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Introduction: The objective of this project is to carry out a mid-course correction during a prostate implant procedure. Situations arise where implanted sources deviate significantly from their intended coordinates. As a result, the dose distribution differs from that intended. Data from the published literature indicate that even in the best hands, less than adequate dose delivery to the planning target volume is possible in up to 20% of patients undergoing prostate brachytherapy. The capability of localizing the implanted radioactive sources and generating a new plan would provide an opportunity for correcting and re-optimizing the dose distribution before completing the procedure. A custom genetic optimizer was coded for generating an updated mid-course treatment plan for the remaining sources, taking into account those sources already implanted. The authors believe that updating the treatment plan in this manner will provide the opportunity for correcting sub-optimal implants before the procedure is completed, thereby preventing medical misadventures. Additionally, this approach may be used at the end of implant (with the patient still on the table) for intra-operative post implant analysis and corrections, when necessary.

Methods: During the brachytherapy procedure, the implanted source coordinates will be ascertained. Markers attached to the US probe will be used to register CT-resolved source coordinates to the US frame of reference. The genetic optimizer will then use this information to calculate the dose distribution resulting from the implanted sources. With the dose distribution known, the optimizer will attempt to determine appropriate coordinates for the remaining sources to meet the prescription criteria as closely as possible. In regions of the prostate where the doses are high, the optimizer will recommend adding fewer new sources. Conversely, in regions where the doses are low, the optimizer will recommend adding more sources. In situations where the initially implanted sources are sub-optimally situated, the dose distribution of a mid-course corrected plan should be superior to an implant performed without such correction.

Results: The optimizer is currently being tested and validated. Work is also under way to integrate the optimizer with Memorial’s treatment planning system.

Discussion: At this stage, the main goal of this project is to correct prostate implants at the time of the procedure, obviating the need for salvage therapy. In principle, this software could also be used for salvage therapy, generating a treatment plan for a second implant performed weeks or months after the first. In the case of a salvage procedure however, the question arises as to what should be the prescription dose for the implant. With this question answered and the implanted source coordinates determined, the optimizer can be applied as in the intra-operative case.
In both the intra-operative and post-implant scenarios, the statement of the optimization problem remains the same, namely where to place additional sources to attain the desired composite dose distribution. In other words, the optimizer would elevate the doses delivered to unintended cold areas within an implant while minimizing the dose elevation to the higher dosed regions.

After the optimizer has been put into clinical service and validated in the intra-operative setting, it will be applied to salvaging sub-optimal implants. In a future version of the optimizer, radiobiological criteria will be used for generating salvage treatment plans. Radiobiological modeling would also provide the opportunity for using HDR for the salvage therapy. The dose distributions of the LDR and HDR procedures could be combined in a radiobiologically meaningful manner. Development of a future HDR optimizer for such application is under consideration.

**Conclusion:** For most medical procedures, it is preferable to intervene and execute a mid-course correction before the problem becomes permanent. During a brachytherapy procedure it is possible to compensate for less than ideal source placement before the prostate implant is completed. This capability however requires accurate intra-operative source localization. The authors are optimistic that the using computer automated intra-operative mid-course correction will reduce the number of implants requiring salvage therapy.

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Target Positioning Accuracy of a Novel Technology for Target Localization in Radiation Therapy

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**Background:** Position accuracy has long been an essential and challenging issue in radiation oncology. With more localized treatment modalities becoming more popular, like IMRT, Body Radiosurgery, etc., the demand for accurate and quick localization has increased and many image guided systems have been introduced to the market (i.e. Cone Beam CT, Orthogonal X-rays, Optical tracking, Ultrasound). All of them require substantial pre-treatment setup time and sometimes physician analysis. A need for a fast, accurate, and objective target localization system has been recognized. This is extremely important for targets which are not attached to bones (like prostate) and one cannot rely on bony anatomy for localization.

**Methods:** The system used is a set of detectors attached to a linac gantry and a radioactive fiducial seed implanted in the target. The active isotope in the seed is a 50 \(\mu\)Ci \(^{192}\text{Ir}\). The detector system does not interfere with the gantry and collimator movements and the reduction in clearance is about the same as for any external MMLC system (about 8 cm). The target was localized with the system and then a cone beam CT was performed to check the localization accuracy. Localization and accuracy measurement was performed six times. The test was performed using a tissue equivalent body phantom.

**Results:** The mean positioning accuracy of the system was 0.3mm, with std. of 0.16mm. The localization was real time and the entire process took less then one minute (including repositioning, not including the CBCT which will not be a part of the clinical system).

**Conclusions:** The radioactive tracking system is a fast, accurate and objective system which is appropriate for real time localization and positioning in radiation therapy. It has great potential utility for rapid and accurate beam positioning and for localization of both targets and critical normal tissues.

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Large Feld IMRT for Malignant Plural Mesothelioma: Comparison with Conventional Technique

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Background:
Death in patients with malignant pleural mesothelioma is most often caused by local disease progression. Aggressive surgical treatment and adjuvant radiotherapy has been widely utilized, which cause significant mortality due to radiation pneumonitis. This severe complication might be related to contralateral lung dose. Two approaches are used for delivery of adjuvant hemithoracic RT:
1. Opposed anterior-posterior photon beams with electron boost to blocked areas.
2. Intensity modulated fields in various configurations.

Purpose:
We applied special IMRT field arrangement with span of 210 degrees around the target, based on manually restricted jaw apertures for potentially hazardous beam directions and compared the resultant dose distribution with standard AP-PA technique.

Methods and materials:
We performed a dosimetric comparison of two techniques, evaluating dose delivery to target, contralateral lung, heart and liver for left and right sided cases. The prescribed planned target dose was 45Gy. New set of constrains was utilized, that included volume of contralateral lung receiving more than 5Gy and mean lung dose. All calculations were performed on Eclipse 7.5 software (Varian inc., Palo Alto, US). Target coverage and homogeneity indexes as well as dose-volume histograms for contralateral lung, heart and liver were compared.

Results:
In both techniques the target volume coverage was acceptable (TC>0.95), while dose homogeneity index was higher for IMRT plans. IMRT approach seems to be advantageous in terms of dose sparing to liver and heart, but increased the low dose volume of contralateral lung as measure by V5, V20 and mean lung dose.

Conclusions:
The balance between toxicity and efficacy seems to be determinant for malignant pleural mesothelioma treatment, because improved local tumor control is crucial for better survival rates. Looking into dose escalation up to 54Gy we may expect IMRT provide better dose distribution in terms of target homogeneity and coverage. With strict attention to limiting the dose to critical organs it will be feasible to deliver higher tumor dose safely.
Figure 1: DVH for IMRT 8 fields fix jaws.

Figure 2: DVH for AP/PA technique.

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Irradiation Accidents in Radiotherapy: Analyze, Manage, Prevent

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Why do errors occur? How to minimize them?
In a context of widely publicized major incidents, of accelerated technological advances in radiotherapy planning and delivery, and of global communication and information resources, this critical issue had to be addressed by the professionals of the field, and so did most national and international organizations.
The ISMP, aware of its responsibility, decided as well to put an emphasis on the topic at the occasion of its annual meeting.
In this frame, potential errors in terms of scenarios, pathways of occurrence, and dosimetry, will first be examined. The goal being to prioritize error prevention according to likelihood of events and their dosimetric impact.
Then, case study of three incidents will be detailed: Epinal, Glasgow and Detroit. For each one, a description of the incident and the way it was reported, its investigation, and the lessons that can be learnt will be presented.
Finally, the implementation of practical measures at different levels, intra- and inter-institutions, like teaching, QA procedures enforcement or voluntary incident reporting, will be discussed.

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Can Record and Verify Systems Eliminate Radiotherapy Delivery Errors?
The Rambam Medical Center Experience

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The use of Record and Verify (R\&V) systems in radiotherapy departments is well known for a long time. Those tools that were originally created for recording and verifying the radiation field parameters, developed into comprehensive systems that now allow for completely paper-less radiotherapy clinics.

In the radiotherapy unit at Rambam Medical Center 200 patients are treated daily (8 am – 7 pm) in four linear accelerators (linacs). During the last 4 years our accelerators (ELEKTA Precise1, VARIAN 600C2 and VARIAN 18002) were connected to the IMPAC Multi-Access system. Because of technology differences between the accelerators, the implementation and connection to the IMPAC system was not identical for the different linacs. However, in all cases the recording and verification of the treatment parameters is possible. Twenty palliative patients are treated on a Varian 6-100 machine that was not connected to the IMPAC system because of its old technology and future planned replacement.

The main reason for the introduction of the IMPAC system in our department was to improve the quality control (QC) of the treatment delivery. We also expected that administrative aspects like patient’s treatment scheduling will be improved. The implementation of the system was not meant to replace our “paper” radiotherapy chart but to be used in parallel with it. As a result we expected an increase in the workload of the staff involved in the treatment planning and delivery process.

Until the introduction of the system the workflow of the patient’s treatment parameters was as follows:

I - Treatment dose is prescribed by the physician and written in the chart
II - Treatment plan is performed (2D or 3D) and all treatment parameters (setup and dosimetry) are defined/calculated by technologist/physicist or dosimetrist and written in the chart.
III - Treatment dosimetric parameters are checked by a senior physicist
IV - Treatment setup parameters are checked by a senior technologist
V - Treatment is delivered and manually recorded

In the above described process we can find several weak links that in a busy clinic like ours and for treatments with constantly increasing complexity may induce treatment delivery errors. Among the most common errors we found: prescriptions that were changed without physician approval;
Setup parameters like SSD, collimator opening, beam modifiers like blocks or wedges that were not clearly documented in the chart; mistakes during treatment delivery like wrong MU used for the irradiation and wrong documentation of cumulative daily doses. The above are the usual errors often presented in the literature and in public Internet sites like ROSIS3.

The Multi-Access system requires electronic approval of the data entered to enable treatment delivery. Each of the users (physicians, physicists, technologists, secretaries) has security permissions that allow him to add/modify/approve a specific type of information. With the implementation of the R&V system the process had to be changed accordingly. Referring to the already described process the following was modified: Data previously written in the paper-chart is also entered in the e-chart and electronically approved by the authorized persons. For 3D plans, parameters are transferred directly from the Treatment Planning System. Treatment delivery is automatically recorded in the IMPAC system.

When comparing and analyzing the new adopted process we can show improvement relative to the previous situation. First, if the prescription is modified by a non-authorized person it is not possible to irradiate the patient. The dose prescription entered by the physician prevents overdosing when trying to irradiate more fractions than the number prescribed by him.

Second, treatment parameters that were approved by a senior physicist can be modified only by an authorized physicist and requires approval. In the past the technologist could change some of the geometrical parameters in order to compensate for inaccurate patient setup. This is not possible anymore and requires physician and physicist intervention. For 3D plans the weak links described previously were resolved where for the 2D plans human errors are still present. Since 2D plans are becoming less common, can we say that this type of R&V system eliminates all the errors? Unfortunately the answer is NO, new weak links appeared.

When people work with computer driven systems they tend to rely almost blindly on them. As a result they will not criticize data appearing on the screen preventing them from finding mistakes. Even more, especially for the advanced accelerators where data is directly transferred by the R&V system, a patient identification problem may arise when trying to speed the treatment process in order to accommodate more patients in the machine schedule. To conclude: from our experience the R&V system allowed us to improve our QC process reducing some of the common errors, but its implementation requires staff change of attitude with respect of the treatment delivery process.

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The 24th Conference of the Nuclear Societies in Israel
Implementing a New Chart in the Radiotherapy Unit at Rambam Medical Center

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The Radiotherapy chart is the paper form used by the multi-disciplinary team of the Radiotherapy [RT] Unit to convey information and register treatment orders. The chart is a major tool which helps the medical and para-medical staff to provide the correct treatment to the patient and to assure that every step is performed appropriately and approved. For this reason, the RT chart is one of the quality assurance tools used in the Unit.

In the last decade, the RT chart used at Rambam Medical Center was very simple, 4 pages long, designed by the staff members themselves. Several errors in delivering RT led to a decision to involve the Research Center for Work Safety and Human Engineering at Technion. This group conducted a human factors and safety analysis study in the RT Unit. After both general and focused analyses, the group recommended redesigning the RT chart which plays an important role as an information and guiding map, affecting the quality of treatment and the patient’s health and safety.

Analysis of different aspects of the old chart, such as its cognitive structure and suitability for the work procedures, demonstrated several problems and deficiencies. These problems could easily influence efficiency and work safety and cause adverse events. The major problems were: memory load – only 50 of 120 necessary fields were present in the old chart; ambiguity of unfilled data fields – a data field that was left empty caused the reader of the chart to assume that it was not needed, when it was left empty by mistake and the required information was missing; inconsistence in the presented information – the location and the streaming of information fields could cause errors while the required information was being copied, thereby complicating the search for the needed information; salience of dynamically changing information – the dynamics of the patient’s status during treatment leads to a need to perform changes in treatment and the information in the chart.

The new RT chart designed by the Technion group in cooperation with representatives of the RT unit includes a human centered approach and implements simple solutions that help to improve the quality and safety of treatment. It is a longer chart, comprised of different sections, each in a different color. The Technion group and the RT team are scheduled to meet and analyze the preliminary problems of the new chart after 3-4 months of use.

The RT staff began using the new chart more 2 years ago. The start was not easy. A change is always hard to implement, and the ones who were not involved in designing the chart had many remarks and objections. The attitude of the staff towards the use of the new chart varies and each discipline handles it differently. The technicians have met to discuss the problems and most of their remarks are concerned with the difficulties of getting used to the new chart. They complain that the new chart is too long and complicated. The physicians and the nurses showed greater readiness to use the new chart. They need to use it less frequently than the technicians and they cope with the challenge and the change more easily. They find the new chart to be more convenient to complete and some think that it has achieved its goals. More attention is needed to complete the chart and, therefore, fewer mistakes will be made.
The physicists, too, were ready to use the new chart and showed no signs of objection. They found the new chart to be more convenient, and only a small amount of change is required for a perfect use. The administrative staff had no problems using the new chart. For them, the only problem was the filing. The old chart fit into the patient's Oncology file, while the new chart is too thick and another filing solution needs to be found.

After one year of use, the new chart was reviewed by all the staff and the comments were analyzed. Some were rejected and the remaining were implemented. A new version of the chart was printed and introduced in the clinical practice. We do not expect changes in the near future.

In summary, applying cognitive psychology principles and implementing a human centered approach in the design of a new RT chart, a very important document used by the RT staff during RT treatment, is essential. The process at Rambam Medical Center that began 2 years ago, and there is no doubt that, after the learning process, the proper use of the chart has minimized critical errors and has contributed to the implementation of quality care and improved patient health and safety.

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Operation and Maintenance of SARAF

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INTRODUCTION
The Soreq Applied Research Accelerator Facility (SARAF) is based on a 5-40 MeV, 0.04-2 mA proton/deuteron RF superconducting linear accelerator. For details about SARAF and its applications, see Ref. 1. The accelerator is currently under commissioning at Soreq NRC. Nevertheless, it has become necessary to initiate operation, maintenance and engineering procedures of the facility even during the accelerator commissioning, especially for the infrastructure systems. This paper presents the operation, maintenance and engineering concept and procedures, with emphasis on personal safety systems that were developed and are used by the small engineering group of SARAF. The group, which will eventually evolve into the operations group, participates in the accelerator commissioning process, which is lead by the accelerator contractor and in parallel, manages and carries out maintenance activities for all infrastructure systems and some of accelerator components. The group is also responsible for gathering and arranging of all the systems documentation which is continuously being delivered as the systems are accepted. A custom software application for managing all these procedures and a Main Knowledge Base are presented as well.

BUILDING AND INFRASTRUCTURE
SARAF includes two main buildings (the Energy Center and the Accelerator Building) and the following infrastructure systems: Cooling Water, HVAC (Heating, Ventilation and Air Conditioning), Compressed Air, Communication, A personal safety system (PSS) including a radiation safety system (RSS), Electrical Supply and Distribution, an ultra high vacuum system and Control Systems.
Most infrastructure systems were accepted and are being used for the accelerator installation and commissioning.
SENIOR ENGINEERING TEAM
Commissioning, acceptance tests, operation, engineering and maintenance of all SARAF systems are under the responsibility of the SARAF Senior Engineering Team (SSET). The SSET includes an Electrical Engineer (Control Systems, Infrastructure, RF, Team leader), a Mechanical Engineer (Cryogenics, Vacuum), a Physicist (Accelerator, Diagnostics, Beam lines), an Industrial Engineer (Maintenance Management, Documentation), a Safety Specialist ("online" safety, procedures for present and future) and two accompanying part time technicians.

SSET members were trained up to supervisor level to operate those systems. In parallel, they are preparing all the procedures for the SARAF systems. Some procedures are already complete and in use. All procedures and systems' documentation and drawings are kept and updated within the SARAF Technical Office, which is managed by the SSET leader and the industrial engineer. The Technical Office tasks also include maintenance scheduling, coordination among SARAF and external contractors and suppliers, spare parts purchasing and stock management.

Engineering activities during accelerator installation includes R&D of accelerator complementary systems such as the control system of the beam halo monitor and of the beam dump, beam lines design and procurement and infrastructure systems improvement.

SARAF MAINTENANCE
In order to preserve the operability of all SARAF systems, maintenance procedures have been developed and implemented. Currently, maintenance is mainly performed on the buildings, infrastructure systems, cryogenics and part of the accelerator vacuum and RF systems. Maintenance is divided into three categories:

1. Breakdown Maintenance - "run to failure", no intervention in plant operation, plant is maintained only when forced by breakdown.
2. Preventive Maintenance (PM) - care and servicing for maintaining equipment and facilities in a satisfactory operating condition, by providing for systematic inspection, detection, and correction of incipient failures either before they occur or before they develop into major defects.
3. Predictive Maintenance (PdM) - techniques that help determine the condition of in-service equipment in order to predict when maintenance should be performed.

For the abovementioned maintenance tasks, a Computerized Maintenance Management System (based on Microsoft Access) has been developed, which enables assets management, PM and PdM maintenance scheduling and management and report generation. As a result of centralized knowledge base demand, a Media Wiki based Main Knowledge Base was created. It includes documents regarding the accelerator and auxiliary systems, presentations, papers and reports related to the project, and an overview of the SARAF staff. The application is based on the Media Wiki engine and has been developed and is updated by SSET.

PERSONAL SAFETY SYSTEM
The SARAF PSS concept has been described in Ref. 2. In the following, we describe the implementation of the SARAF PSS, which is already installed and fully operational. The main interface screen is shown in Fig. 1.
Figure 1. PSS main interface screen

Regarding entry to the accelerator area (red zone), there are three safety statuses: Closed (red), where access is restricted and the accelerator may be on; Controlled (yellow), where access is controlled and the accelerator is off; Free (green), where access is free and the accelerator is of course off. The safety status is displayed by colored ‘traffic lights’ throughout the building and on the PSS screens. In yellow status, entry to the red zone is limited to six users and is done by a strict procedure, which involves communication with the Main Control Room (MCR) and removing a special safety key from the entry panel (see Fig. 3) that disables accelerator operation. In green status, entry is not monitored.

Figure 3. Main entry to the SARAF Accelerator (left) and entry panel (right)
The PSS system is interfaced to the Radiation Safety System (RSS), which monitors radiation levels inside the accelerator area and the areas surrounding it. Using that data the PSS blocks entrance to the accelerator area until the radiation level there is reduced to a defined safe level. PSS Radiation readings above threshold do not drop the enable signal, but generate an alarm.

The interface between the PSS and the accelerator is such that the accelerator can be operated and beam can be generated only if the PSS provides a redundancy characterized hardware ‘enable’ signal. The implementation is performed by safe relays and safe PLC IO cards (see Fig.5) This enable signal is dropped in the following cases: 1) one of the red area external doors is opened, 2) one of the emergency buttons is pushed, 3) one of the special safety keys is removed from the entry panel, 4) the operators requested it via the Human Machine Interface, 5) the main door bypass button is engaged 6) one of the Machine Safety System interlock signals indicates local system failure.

Figure 4. Radiation Safety screen.

Figure 5. Safe Relays (right) and safe PLC IO cards (left)
The control and monitoring over the PSS system is performed from the main control room safety station (see Fig. 6) which includes a CCTV system and main announcement and emergency intercom systems.

Figure 6. Main Control Room Safety Station

SUMMARY
This paper briefly presents the manpower resources, maintenance concept of SARAF and Personal Safety System implementation highlights. The maintenance and engineering procedures that were developed and are used by the small engineering group of SARAF were described. Development and assimilation of maintenance and engineering systems during accelerator installation and commissioning, in parallel to observing safety regulations and interfacing with main contractor personnel presents a significant challenge for the SARAF Senior Engineering Team.

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First Experience at SARAF Intense Low-energy Beams: Radiological Concern

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INTRODUCTION
The linac in Soreq Applied Research Accelerator Facility (SARAF) will be used for the acceleration of a few mA proton and deuteron beams up to 40 MeV [1,2]. Commissioning of the first stages of the accelerator, including the ECR Ion Source (EIS), Low Energy Beam Transport (LEBT) system is currently in progress [2]. The use of intense deuteron beams at the SARAF accelerator as well as at other future facilities as SPIRAL II, EURISOL and IFMIF [3] creates serious radiological problems associated with neutrons production. Even at low energy of few tens keV neutron production takes place via D(d,n)^He reaction. Therefore radiological issue of operation of intense deuteron beams has to be evaluated even for low beam energy. We attempted to obtain maximum information on the subject during recent acceptance tests of the SARAF's EIS and LEBT systems.

PREPARATION AND CONDITIONS OF THE TEST
A schematic view of the low-energy part of the SARAF accelerator is presented in Fig. 1. A deuteron beam was extracted from the ECR ion source by 40 kV extraction voltage. The LEBT optical system consisted of three focusing solenoids and dipole magnet and diagnostic devices for measuring the total current and transversal emittance. A more detailed description of the EIS and LEBT systems can be found elsewhere [4]. The beam might be intercepted at several places along the LEBT line: at a beam stopper before the dipole magnet, at the vertical and horizontal slits and wires for emittance measurements and a Faraday cup. In addition a temporary beam blocker was placed before the third solenoid.

A number of radiation CR-39 tags manufactured by American Acrylic were placed along the beam line for measuring of the total neutron dose (red squares on Fig. 1). At most positions we used pairs of tags in order evaluate the systematic errors. The procedure for processing CR-39 tags is calibrated with a ^{252}Cf source (2 MeV average neutron energy). Therefore one expects to obtain reliable results for 2.45 MeV neutrons. The list of CR-39 tags as well as their approximate distances from the beam axis or to interceptive elements are indicated in Table I.

In addition a mobile neutron monitor, “Snoopy” [5], and a Seforad ^{3}He neutron spectrometer [6] were placed at distance of 45 and 30 cm respectively from the Faraday cup (distances measured to the centers of detectors). The Snoopy monitor has the ability to provide neutron dose normalized to the theoretical ICRP rate within ± 10% [7]. The overall accuracy of the monitor quoted by manufacture is of about 30%. The reading of the monitor's analog scale was observed remotely via an internet camera.
The Seforad spectrometer provides information on neutron energy spectrum. The energy resolution for thermal neutrons obtained with a weak neutron source was 14 keV. This is close to the best values that could be found in literature. Ionization charge, collected in the gas chamber of the counter is amplified by a charge sensitive preamplifier and spectroscopy amplifier and digitized in a PC based multichannel analyzer. The PC and part of electronics were placed at large distance from the detector to avoid possible damage from neutrons. Counting rate in the detector could be also measured with a rate meter (ORTEC 661) and converted to a voltage signal with possibility of digitizing and on-line monitoring using a National Instruments 6008 analog-to-digital converter. The intrinsic efficiency for detection of 2.45 MeV neutrons is known from literature [8]. Finally an indium sample was placed 9 cm from the FC in order to determine the 2.45 MeV and thermal neutron fluxes via off-line measurement of the sample activation.

![Diagram](image)

**Fig. 1** Low-energy part of SARAF linac. The main optical elements, interceptive beam diagnostic elements and elements for neutron diagnostics are shown. Neutron radiation tags are indicated by squares and numbers (see Table I ).
A few mA of 40 keV deuterion beam was extracted from the source and transported to the FC. In order to form the deuterons beam in the ECR we use a 99.8% isotopic abundance D2 gas at 99.999% purity. During the measurements most of the time, the current was on the FC when stability and adjustability of the ion source were measured. The total charge accumulated on the FC was of the order of 96 Coulomb. The charge integral measured on the beam stopper was of the order of 12 Coulomb. A considerable part of the time was devoted to emittance measurement when most of beam was intercepted by the slits and a small fraction was dumped into the beam blocker. A small fraction of the beam of the order of 10% was cut by the aperture. At present one can not measure the charge collected on these last two components.

The ion source was operated with 40 keV, 5 mA of molecular hydrogen (H2+) beam on a day prior to the measurements with deuterons. Thus it is reasonable to assume that the previously implanted deuterium was annealed from the FC surface prior to the operation of deuterion beam.

**RESULTS**

The results of the measured neutron doses together with information on tags positions are presented in Table I. The accuracy of the dose measurements was estimated to be 40%. The results indicate that the neutrons are mostly originated from interceptive beam components.

<table>
<thead>
<tr>
<th>Position</th>
<th>Tug number</th>
<th>Place along the beam line</th>
<th>Distance to the beam axis (cm)</th>
<th>Measured dose (mrem)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>1,2</td>
<td>EIS HV cage</td>
<td>45</td>
<td>42,33</td>
</tr>
<tr>
<td>2</td>
<td>3,4</td>
<td>EIS extraction electrode</td>
<td>10</td>
<td>355,275</td>
</tr>
<tr>
<td>3</td>
<td>5,6</td>
<td>Beam stopper</td>
<td>11</td>
<td>101,174</td>
</tr>
<tr>
<td>4</td>
<td>7,8</td>
<td>Dipole magnet chamber</td>
<td>5</td>
<td>31,35</td>
</tr>
<tr>
<td>5</td>
<td>9,10</td>
<td>Aperture</td>
<td>15</td>
<td>199,174</td>
</tr>
<tr>
<td>6</td>
<td>11,12</td>
<td>Slits cross</td>
<td>14</td>
<td>358,322</td>
</tr>
<tr>
<td>7</td>
<td>13,14</td>
<td>FC/wire cross</td>
<td>14</td>
<td>1297,1511</td>
</tr>
<tr>
<td>8</td>
<td>15,16</td>
<td>Beam blocker</td>
<td>8</td>
<td>679,569</td>
</tr>
<tr>
<td>9</td>
<td>17,18</td>
<td>Diagnostic table</td>
<td>61</td>
<td>21,21</td>
</tr>
<tr>
<td>10</td>
<td>18,19</td>
<td>Dipole magnet table</td>
<td>70</td>
<td>background</td>
</tr>
<tr>
<td>11</td>
<td>20</td>
<td>EIS table</td>
<td>60</td>
<td>background</td>
</tr>
<tr>
<td>12</td>
<td>21</td>
<td>Concrete wall</td>
<td>~200</td>
<td>background</td>
</tr>
</tbody>
</table>

Table I. Summary on radiation tags positions (see Fig. 1) and measured neutron doses.

The rate of the neutrons from the implantation point (FC) was determined by four independent measurements. The results of measurements scattered within a factor of 2 (Table II). As a “rule of thumb” one can expect \( \sim 2 \times 10^6 \) n/s/MA/4\( \pi \) from a 40 keV deuteron beam impinging onto a graphite target.

---

1 The secondary electron suppression is not utilized at the beam stopper. Therefore the measured charge integral should be considered as an upper estimate.
Table II. Summary of neutron rate measurements.

<table>
<thead>
<tr>
<th>Measurement</th>
<th>Deuteron current (mA)</th>
<th># neutrons (n/s)</th>
<th># neutrons per mA (n/s/ mA)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Radiation tags</td>
<td>5</td>
<td>6.5(2.5)·10^6</td>
<td>1.3(5)·10^6</td>
</tr>
<tr>
<td>Snoopy monitor</td>
<td>5</td>
<td>9.0(1.5)·10^6</td>
<td>1.8(3)·10^6</td>
</tr>
<tr>
<td>Indium activation</td>
<td>2</td>
<td>5.0(8)·10^6</td>
<td>2.5(4)·10^6</td>
</tr>
<tr>
<td>Scforad detector</td>
<td>5</td>
<td>1.2(2)·10^7</td>
<td>2.2(4)·10^6</td>
</tr>
</tbody>
</table>

To our knowledge another neutron background measurements during operation of intense (many mA) low-energy deuteron beam were done at CEA-Saclay [9-11]. As it seen from [9-11] the strength of the neutron source at SARAF is somewhat higher than at Saclay. It is clear that the origin of the neutron source is deuterium on the target surface rather than deuterium at the implantation depth. The quantity of deuterium on the surface depends on implantation history and surface temperature. Most likely the main difference between SARAF and Saclay cases is due to the temperature of the target surface.

More systematic studies are needed for a better understanding of the physics and related radiological issues of operation intense deuteron beam. These studies are planned to be done at SARAF in the next future.

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Lattice Study for SARAF 40 MeV Linac and Extended for the EURISOL 60 MeV Low Energy Driver

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INTRODUCTION

In this study we examine a lattice for the SARAF superconducting (SC) linac at the low velocity (\(\beta\)) range. The accelerator at SARAF is composed of an ion-source, a 4-rod 176 MHz 1.5 MeV/u RFQ and a 40 MeV proton and deuteron SC linac. The SC accelerator is described in details in [2] and the accelerator front-end in [3,4]. Driven by the motivation of saving construction and operation cost the SARAF linac starts with SC independently phase 2-gap Half Wave Resonator (HWR) cavities, right after the RFQ. The common solution of DTL in this transition (between the RFQ and the SC linac) was rejected in order to have an efficient high energy gain acceleration of a large range of mass over charge ratio (\(M/q=[1-2]\)) ions. The HWR cavities at the first cryostat are optimized for a geometric \(|\beta_o|=0.09\) and hence the \(\beta=0.0567\) ions from the RFQ are mismatched. The development of an additional SC cavity with \(\beta_o\) at the range of 0.06 at 176 MHz seemed to be technically complicated due to inner surface treatment in such a narrow gaps size.

The effort to accelerate light ions at the low beta range with SC linac in a high acceleration rate is limited by the induced strong longitudinal focusing force\(^{(5)}\). High accelerating gradient could introduce high longitudinal phase advance resulting in beam losses\(^{(6)}\).

In this work we developed an adiabatic tuning method for the low \(\beta\) side of the accelerator that is facing the two problems of mismatched velocity and over focusing at high acceleration gradient. A preliminary approach of this method is applied to the SARAF 40 MeV deuteron accelerator extended for 60 MeV for the low energy part of the EURISOL driver accelerator. We find our study relevant for other linacs that start with SC cavities right after the RFQ, such as SPIRAL2, and maybe IFMIF too, which start with similar \(\beta\) mismatch at the low \(\beta\) range.

BEAM TUNING METHOD

For multi gap cavities, the problematic region for the tune is the low \(\beta\) section. In this area the acceleration in each gap is quite large in comparison to the particles initial velocity at the cavity entrance. The particle deviation from the reference particle at the second gap of the cavity depends on the velocity gain of the particle at the first gap. The evaluation of the particle's trajectory, with significant velocity increase along the cavity, has to be performed at each gap separately, taking into account the particles phase deviation from the reference particle at each gap\(^{(7,p.189)}\). If the accelerating RF field along the bunch in one of the gaps, deviates from the linear range it could introduce a significant emittance growth during the acceleration. The range of the accelerated particle's phase \(\phi_i\), at each gap \(i\) is defined by the reference particle phase \(\phi_{r_i}\) and the bunch half length \(\Psi_i, \phi_{r_i} - \Psi_i \leq \phi_i \leq \phi_{r_i} + \Psi_i\). \(\Psi_i\) can be estimated from the phase distribution of the tuned bunch at the gap entrance. For a two gaps cavity, \(i=1,2\) (Fig.1) the reference particle phase at each gap is evaluated from the synchronous phase \(\phi_s\): \(\phi_{r_1} = \phi_s - \delta, \phi_{r_2} = \phi_s + \delta, \delta = (\beta_0/\beta - 1) * \pi / 2\). Where \(\delta\) is the difference between the synchronous phase and the reference particle phase at both gaps, \(\beta_0\) is the cavity geometric \(\beta\) and \(\beta\) is the reference particle relative velocity at the first gap exit.
To minimize the emittance growth along the linac at the low $\beta$ section, in each gap of the multi gap cavities, it is validated that the acceleration RF field is linear along the bunch. A key factor for a good tuning is a small bunch width along the linac. The longitudinal phase advance per unit length is kept constant (based on [7 p.175]). The method developed here assumed a single cavity per period, further development is needed for few cavities in a period:

$$\frac{d(W - W_s)}{ds} = qV_0 T \left( \cos \Phi - \cos \phi_s \right)/L \quad \frac{d(\Phi - \phi_s)}{ds} = (W - W_s) 2\pi / (mc^2 \gamma_s^3 \beta_s^3 \lambda)$$

$$k_{10}^2 = -2\pi / (mc^2 \gamma_s^3 \beta_s^3 \lambda) q V_0 T \sin \phi_s / L \quad V_{02} T_2 = V_{01} T_1 \left( \sin \phi_{s1} / \sin \phi_{s2} \right)^2 \left( \gamma_{s2}^3 \beta_{s2}^3 L_2 / \gamma_{s1}^3 \beta_{s1}^3 L_1 \right)$$

Where $W$ is the bunch particle energy, $W_s$ is the synchronous particle energy, $\Phi$ is the approximated particle phase along the cavity gaps, $\phi_s$ is the synchronous particle phase, $q$ is the particle charge, $L$ is the distance between cavities, $k_{10}$ is the phase advance per unit distance, $V_0 T$ is the energy gain at zero synchronous phase per unit charge at a cavity based on the cavity voltage $V_0$ (proportional to the cavity field amplitude) and the bunch transient time factor $T$, $i$ is the cavity index ($i=1$ is the upstream cavity, $i=2$ is the cavity to tune) and $s$ is the distance at the beam direction.

This formula implies that after applying a linear RF acceleration field along the bunch, in each gap, then, the energy gain of the cavity at zero synchronous phase ($V_{02} T_2$) is tuned according to the phase advance per unit length at the upstream cavity $k_{10}$, in order to avoid significant local variations of the phase advance per unit length. The applied phase advance per period $\sigma_i$ is evaluated by: $\sigma_i = \int \frac{1}{\hat{\beta}(s)} ds$ Where $\hat{\beta}$ is the Courant Snyder parameter. (8 p. 15)

At the high $\beta$ range, in order to maintain the acceleration efficiency, the synchronous phase is kept between -20 and -15 degrees, but not above -15 degrees in order to maintain stability. The transverse phase advance should be kept higher than 50% of the longitudinal phase advance to eliminate emittance growth and halo development (9).
LINAC LATTICE SETUP
The linac lattice is described in details in ref. [1-3] and references there in. The linac is matched transversely to the RFQ with three quads along the MEBT to convert the beam to a radial symmetric shape. The linac is composed of two $\beta_0=0.09$ cryostats followed by four $\beta_0=0.15$ cryostats. Each period at the basic linac asymmetric configuration is composed of a leading SC solenoid followed by two SC cavities. The first solenoid in the first cryostat serves as a cold trap, which protects the SC cavities from contamination emerging from the injector. The first SC cavity of the linac is used as a buncher. At the symmetric lattice the distance between the first and second MEBT quads is enlarged by 9 cm to increase the transverse focusing. In this case the linac starts with a leading cavity operated as a buncher so the RFQ to buncher distance is reduced by 16 cm with respect to the 107 cm distance in the asymmetric one. Both lattices enable acceleration of a 4 mA deuterons and proton CW beams up to 40 MeV. The EURISOL driver is extended by three $\beta_0=0.15$ cryostats to reach 60 MeV. “End to end” simulations of the SARAF linac have been performed using TRACK(10) from the 20 keV/u ion source to 40 MeV, extended to 60 MeV for the EURISOL driver. The 3D fields of the LEBT solenoids and the fringe fields of the LEBT bending magnet were modeled. The RFQ accelerating structure was generated according to the RFQ design data(4), 3D fields were modeled for the radial matcher with EM Studio. The fields in the regular cells are presented by the 8-term Fourier Bessel expansion. The 3D fields of the SC solenoids were calculated and the 3D fields of the SC cavities were included in the simulation.

BEAM TUNING RESULTS AND COMPARISON
Phase spaces of 500k macro particles at the RFQ exit and at the entrance to the first HWR gap of the symmetric and the asymmetric lattice are presented in Fig. 2. The bunch is kept at a linear range of the RF field at each low $\beta$ gap using the accelerating phases shown in Fig. 3. The effort to bunch the beam along the linac is presented in Fig. 4. The deuterium energy at the linac exit for SARAF and for EURISOL is $[42.2, 65.2]$ and $[42.6, 66.0]$ MeV for the symmetric and the asymmetric lattices for a 4 mA deuterons beam, Fig. 5. The energy difference between the lattices is not significant. One can probably improve the asymmetric lattice by reducing the energy gain at the low $\beta$ section. The emittance growth is similar at both lattices for the rms and 99.5% envelopes both for longitudinal and normalized transverse emittances. The maximum envelope of 500k macro particles is larger at the longitudinal phase space for the symmetric lattice and vice versa for the transverse phase space, Figs 6 and 7. The symmetric lattice gives significantly higher acceptance than the asymmetric lattice, Fig. 8. The asymmetric acceptance can be improved by changing the buncher synchronous phase.
Figure 3: Symmetric (left) and asymmetric (right) lattice bunch acceleration phase at the first cavity gap (top) and the second gap (bottom) along the low $\beta$ SC linac section.

Figure 4: Bunch amplitude length for the symmetric (dots) and the asymmetric (solid line) lattices.

Figure 5: Bunch energy along the linac for the symmetric (dots) and the asymmetric (solid line) lattices.

Figure 6: Bunch transverse normalized emittance for the symmetric (dots) and the asymmetric (solid line) lattices.

Figure 7: Bunch longitudinal emittance along the linac for the symmetric (dots) and the asymmetric (solid line) lattices.
Figure 8: Longitudinal phase space acceptance for the symmetric (left) and the asymmetric (right) lattices vs. the bunch spread at the RFQ exit (relative energy spread vs. phase (deg)).

CONCLUSIONS
A method to tune a SC linac at the low $\beta$ range with a large $\beta$ mismatch was developed. An approach of this method was used to tune two lattices of the SC linac. The two lattices give satisfactory results from the point of view of energy gain, emittance growth and beam losses (not reported here). Further study of applying this method using an external buncher and trying to apply gentle bunching and acceleration is in progress. At present time, the results of the symmetric lattice including the acceptance test are within the design limits. Additional lattice tunes will be benchmarked to the symmetric lattice to test the benefits.

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INTRODUCTION

SARAF (Soreq Applied Research Accelerator Facility) is a high intensity CW proton/deuteron accelerator, which is currently under construction at Soreq NRC. The Accelerator will accelerate a few mA proton and deuteron beams up to energies of 40 MeV\(^1\). The main components of the accelerator are: an ECR ion source to produce a 5 mA DC beam of protons or deuterons at 20 keV/u, a Radio Frequency Quadrupole (RFQ) to bunch the beam and to accelerate it to 1.5 MeV/u, and an RF superconducting linac to accelerate the bunched beam to 40 MeV.

The SARAF linac is designed for hands-on maintenance, with beam loss requirement of less than 1 ppm. We present a new approach for beam loss calculations which places emphasis on the tails of the particle distribution. The tails are formed during the bunching process in the RFQ, and are the main source of subsequent particle loss in the superconducting linac. Our technique of tail emphasis allows us to increase beam loss statistics by at least an order of magnitude.

We employ the General Particle Tracer (GPT) code for the beam dynamics simulations presented in this work. GPT is a position based multi-particle code which uses 5\textsuperscript{th} order Runga-Kutta integration for tracking each particle through the accelerating and focusing elements over the accelerator length. The code can accept the exact geometric dimensions and the 3D time-varying electro-magnetic fields of the accelerating and focusing elements. GPT also determines the space charge (SC) forces – the self-induced fields generated by the charged particles.

Beam dynamics calculations are CPU consuming, therefore a typical simulation run uses only about \(10^5\) macro particles, while a single bunch contains on the order of \(10^8\) particles. The simulations are therefore not sensitive enough for estimation of ppm loss. A recent paper reported on a method for overcoming the CPU problem by implementation of the simulation code on a 2048-processors parallel computer\(^3\). The method described in our work tries to reach the same level of precision on one PC.

RFQ SIMULATION

The RFQ is a strong transverse focusing device, with modulations at the electrodes to provide the desired bunching and acceleration. The SARAF RFQ is a 4-rod type, operating at 176 MHz, and is described in detail in ref. [5]. It takes a DC beam of protons or deuterons at 20 keV/u, bunches it and accelerates it up to 1.5 MeV/u, thereby preparing the beam for the linac acceptance. Typical ion losses along the 4 m long RFQ is on the order of 10\%, but at low ion energies this loss does not result in radioactive contamination.

Reliable simulation of the RFQ is necessary for an accurate assessment of the particle distribution used as input to simulations of the linac. GPT does not contain a built in RFQ element but does have provisions for user defined elements via user supplied routines in C. We incorporated into GPT an RFQ element, with the electric fields as derived from the 8 terms potential function\(^5\), representing the eight lowest order terms of the Bessel-Fourier expansion of the solution of Laplace equation for the RFQ. The coefficients of these terms are calculated by a least mean squared error fit to the boundary defined by the known electrode surface, as described in ref. [6]. The shape of the electrodes, including the modulation parameters, was provided by the RFQ designer\(^7\). Parameterization for the Radial Matching section was taken from ref. [5], with the SARAF RFQ electrodes shape as input.
Space charge (SC) plays a major role in a high current RFQ, because the energy is low while the current is high. To correctly determine the SC, we start with a DC beam of length $3\beta\lambda$, taken at a phase that will result in three symmetric bunches. The bunch we are interested in tracking is the middle bunch, while the two additional "ghosts" bunches are included to provide the correct SC forces. A mesh technique is used to scale the space charge self-interaction linearly with the number of particles by using a multigrid Poisson solver\(^{8}\) to determine the potential due to space charge. Newman boundary conditions are applied in the longitudinal direction with $\partial V/\partial n = 0$, while Dirichlet boundary conditions are applied in the transverse direction, with $V = 0$ at the mean position of the rod electrode to account for the image charges induced on it.

![Figure 1: Longitudinal phase-space snap shots along the RFQ simulation.](image)

- **Top:** $time = 0\, ns$ - DC deuteron beam of length $3\beta\lambda$ at $4\, mA$ and $40\, keV$
- **Middle:** $time = 230\, ns$ - beam bunching at early stage of development.
- **Bottom:** $time = 557\, ns$ - 3 bunches at the RFQ exit.

Figure 1 shows longitudinal phase space plots which sample the development in time of the bunches in the RFQ, starting with a $4\, mA$ DC beam of length $3\beta\lambda$, and ending with three distinct bunches. The blue, green, and red dots represent particles that will end up in the first, second, and third bunches, while the gray dots represent particles that will be eventually lost.
TAIL EMPHASIS SIMULATION
Simulations of the SARAF linac are performed using as input the particles from the middle bunch generated by the RFQ. Typical linac simulations with $10^5$ particles show no particle loss at all. However, this is not sufficient to determine hands on maintenance, since each macro particle represents hundreds of real particles. These simulations yield an envelope of beam losses on the level of a few nA, while we require sensitivity of at least an order of magnitude better.

The idea behind Tail Emphasis (TE) simulations is to increase statistics only in the interesting region of the beam. Simulations of the linac indicate that losses are caused from longitudinal displacements from the synchronous phase and/or energy, where the displaced particle first loses the bunch longitudinally, enters subsequent cavities at the incorrect phase, thereby experiencing transversal defocusing, and finally ejection from the vacuum pipe. This insight raised the idea of representing the longitudinal tail – the outer part of the bunch in longitudinal phase space – in more details than the inner part. To maintain a correct account of space charge forces, the amount of charge for each macro-particle in the tail is lowered by the same factor as the enhancement factor for the number of particles in this tail.

![Figure 2 RFQ simulations: 4 mA deuterons with tail emphasis.](image)

**Top:** time = 0 ns - DC beam of length $3\beta\lambda$ showing main bunch (green), ghost bunches (red – $1/3$rd statistics), and tail (blue – $\times 25$ statistics).

**Bottom:** time = 230 ns - at early development. The blue particles populate the tail.
Since the longitudinal tail is a result of dynamic processes occurring during bunch formation in the RFQ, the tail emphasis technique needs to be applied at the outset to the RFQ simulations. The criterion for reliably selecting an enhanced tail region is that no mixing should occur with other regions. Figure 2 shows calculations for the RFQ and a comparison between calculations with and without tail emphasis. Figure 2a shows the longitudinal phase space for the initial particles entering the RFQ, including the tail enhanced regions. The main bunch, shown in green, contains macro-particles each representing 250 ions. The tails, shown in blue, include the regions between bunches, and contain 3.6% of the whole simulated DC beam. The density of tail macro-particles is increased by a factor of 25, each macro-particle representing 10 ions, for a beam loss level of 0.3 nA. Since the ghost bunches only serve to provide the correct boundary conditions for space charge, we reduced the statistics for these bunches by a factor of 3 compared to the middle bunch (with a corresponding increase of factor 3 for charge of each macro-particle).

![Figure 3: Beam at RFQ exit.](image)

**Left:** non-weighted simulations. Each macro particle represents 250 particles.  
**Right:** simulations with tail emphasis. Each macro particle in the tail (blue) represents 10 particles.

Figure 3 shows plots for the longitudinal phase space for the beam exiting the RFQ for 1.7 million macro-particles for non-weighted calculations (left), and with calculations containing tail emphasis (right). Both plots show similar overall trends, but the plot with tail emphasis contains a factor of 25 more statistics in the tail (shown in blue).

**BEAM ENVELOPE AND LOSSES IN SARAF LINAC**

We have simulated the SARAF linac by taking as input the distributions generated by the RFQ simulations. Details concerning the linac lattice, basic tune and error analysis are found in ref. [9]. Figure 4 shows the radial extent of the beam along the length of the linac for the r.m.s. size and for the envelope of the most extreme particle. The r.m.s is essentially identical for the regular simulation shown in green, and the tail emphasis simulation shown in blue. However, clear difference is seen for the beam envelope between the non-weighted simulation, which represents beam loss sensitivity at a level of 7 nA, and the simulation with tail emphasis, which represents beam loss sensitivity at a level of 0.3 nA.. While the difference in beam envelope is not dramatic, increasing statistics tenfold or more using the tail emphasis technique will be valuable when performing error analysis and simulations where single-particle precision is needed.
CONCLUSION
We have described a method for significantly increasing statistics of beam loss by placing emphasis on the tails of the particle distributions. We have developed this method for simulating the SARAF accelerator, starting with RFQ simulations with tail emphasis, and ending with increased sensitivity for beam loss in SARAF linac. We have shown that this method is reliable and can increase statistics in the tail by at least an order of magnitude. This approach should be suitable in beam loss prediction, error analysis, and any other simulation where single-particle precision is needed.

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Simulations of Aperture Collimation and Ion Neutralization Effects in the SARAF Low Energy Beam Transport

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INTRODUCTION

The SARAF superconducting linac is currently being commissioned at Soreq NRC\(^{(1)}\). As part of the commissioning process transversal emittance measurements were made of the beam coming out of the ECR ion source and passing through the Low Energy Beam Transport section (LEBT).

Two notable phenomena were observed in the results of these measurements; the aperture cuts away parts of the beam's transversal phase space in an unexpected orientation, and the deuteron beam's phase space includes an apparent secondary beam. We describe the beam dynamics simulations done in an effort to explain these phenomena.

MEASUREMENTS

All of the measured data and simulations that follow relate to part of the SARAF LEBT, described in detail in \(^{(2),(3)}\) and shown in Figure 1.

The section examined begins after a dipole bending magnet that acts as a mass analyzer, allowing only particles of a specific rigidity to pass. At this point a variable-radius aperture is set that enables beam current control by manipulation of the opening size. Following the aperture is a solenoid (a magnetic lens) and further downstream is the slit and wire emittance measurement system.

The slit/wire system\(^{(4)}\) yields a snapshot of the phase space of the beam in one transversal dimension. In this work we study x axis results.

![Figure 1: The SARAF Low Energy Beam Transport](image-url)
APERTURE EFFECTS

Initial measurements were done to ascertain the effect of the aperture installed after the dipole magnet. Figure 2 shows the particle density in the x phase space for a 4.7 mA proton beam with the aperture completely open (all particle density graphs are shown on logarithmic scale). Figure 5 shows one measurement taken with the aperture partly closed, to allow only 3.6 mA of current through.

The aperture acts as a collimator in real space. Therefore the collimation in the momentum axis, as seen in Figure 5, requires explanation. To understand the results, beam dynamics analysis was conducted.

The phase space data gathered by the slit/wire system was taken as a matrix giving particle density as function of the phase space location \((x,x')\) (Figure 2). The full-current results were taken, with open aperture. Using this distribution, made up of roughly 2000 data points, a set of about 80,000 macro-particles was randomly created. The \(y\) phase space was generated with the same parameters as those of the \(x\) phase space distribution. This approximation has been shown to be roughly correct earlier\(^{(3)}\).

The simulated beam was then run backwards through the system described above (from the slit back to the aperture), taking into account space charge. We received a picture of the beam at the aperture (Figure 3).

The effect of the aperture was then applied in real space, as seen in Figure 4, and the beam was run forward to the measurement slit again, applying the same algorithm as before.

![Figure 2: Measurement of proton beam 4.7 mA beam, aperture open](image)

![Figure 3: Beam at aperture (simulation)](image)

![Figure 4: Beam cut by aperture to 3.6 mA (simulation)](image)
The algorithm used to run the particles both back and forwards along the LEBT was based on the linear optical matrices found in the TRACE 3-D code\(^5\), though applied to single particles rather than beam envelope matrices. In addition, TRACE 3-D’s space charge algorithm was also incorporated.

Figure 6 shows the result of the simulation described above for the case of the aperture set to 3.6 mA. The result clearly reflects the same cut seen in the equivalent measurement (Figure 5) though the simulation result uses only the input seen in Figure 2 and the known current of the measurement. We see that though the aperture collimates in the real space, the effects of drift, space charge and the solenoid exchange this into the momentum space. At this location, using a single aperture, most of the beam tail can be cut and the transversal emittance improved\(^2\).

**CHARGE EXCHANGE PHENOMENON**

Another result is shown in Figure 7. This measurement is of a 6.1 mA deuteron beam. It is similar to the previous result (Figure 2), but we note an additional offshoot that appears to the top right side of the beam.

Such a well defined offshoot appears also in other similar EIS+LEBT systems\(^6\). We identify it as a beam of secondary-species particles, differently affected by the solenoid and so arriving with a different phase-space profile at the detector. The secondary beam is probably the result of charge exchange\(^9\), a process in which the beam interacts with residual gases in the beam line, most likely deuterium gas coming from the ion source, exchanging momentum with this neutral gas or creating other charged particles of different rigidity.

The possible interactions for various particles can be investigated. For example, for 20 keV protons colliding with H\(_2\), the calculated ion neutralization rate is 1% m/10\(^{-6}\) mbar, based on the cross-sections found in \(^8\).
However, such cross-sections generally assume ions are at their ground state, while the hot ECR plasma ions can be excited and their cross sections then are not well known and may be higher\(^9\).

For the deuteron beam the most significant interaction seems to be neutral exchange, in which a charged deuteron is incident upon a deuterium molecule and passes either its charge or momentum, creating a neutral particle of the same momentum:

\[
\begin{align*}
d^- (\vec{p}) + D_2^+ (\vec{0}) & \to d^0 (\vec{p}) + D_2^+ (\vec{0}) \\
d^- (\vec{p}) + D_2^+ (\vec{0}) & \to d^- (\vec{0}) + H_2^0 (\vec{p}) + X
\end{align*}
\]

Of these, the first process is more likely. In the simulation we apply an unspecified process, replacing charged particles with neutral particles of the same momentum.

In this case, our goal with the beam dynamics calculation is to verify the origin of the offshoot. The analysis performed is similar to that discussed in the previous section. However, our initial input into the algorithm was modified. We assume that the data presented in Figure 7 is composed of two beams, one of deuterons and one of neutral particles. While the slit/wire system cannot separate these particles, we can identify the offshoot as the neutral beam and remove it by hand. This creates the deuteron-only beam shown in Figure 9, which was used as input for the deuteron simulations. Roughly 3% of the particles were removed in this manner, giving us an idea to the rate of this process.

As before, a set of particles was created and the beam dynamics algorithm was used to transport it back to the dipole exit. The particles were then run forward with the same algorithm, this time including a neutral exchange effect.
The rate of this reaction was used as a free parameter in the simulation, and taken roughly at 3%/m. The result of this simulation is shown in Figure 8.

We can clearly see the creation of the same offshoot that had been removed, with the same angle seen in Figure 7. This verifies our initial assumption and shows the offshoot is indeed a secondary neutral beam.

A significant difference between simulation and measurement is that the measured emittance (Figure 7) exhibits only one offshoot towards top right, while the simulation (Figure 8) shows an additional offshoot towards bottom left. This is explained by misalignment of the solenoid, and indeed manipulation of its alignment in the simulation moves the location of the offshoot compared to the main beam.

CONCLUSIONS

The beam dynamics simulations shown here give clear explanations for the results measured by the slit/wire system. This helps in determining the actual quality and characteristics of the beam in SARAF, allowing for better planning for the future.

With regards to the aperture, the simulation gives a clear explanation of its effect and the ability to predict it.

With regards to the charge exchange, the simulation allows us to determine what portion of the measured beam will enter the linac, and what portions will be lost in the initial bunching and acceleration components.

Future work in this area includes improvement of the beam dynamics code for the space charge, and simulation of further measured cases including an $H_2^+$ beam.

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Beam Dynamics Simulation of the 1.5 MeV Proton Beam Measured at the SARAF RFQ Exit

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INTRODUCTION

The Soreq Applied Research Accelerator Facility (SARAF) accelerator's [1] front-end is composed of a 20 keV/u protons and deuterons ECR ion source, a 5 mA low energy beam transport (LEBT) [2] and a 1.5 MeV/u, 4 mA, 176 MHz, 4-rod RFQ [3] (Fig. 1).

In this work, beam dynamics simulations of the SARAF accelerator front-end is compared to the first proton beam measurements taken during commissioning [4]. Beam transmission, ion energy and bunch width as a function of the RFQ power have been measured in the medium energy beam transport (MEBT) diagnostics and using a dedicated diagnostic plate (D-plate) (Fig. 1 insert). The simulations and measurements show similar trends. This agreement allows calibrating the RFQ power to its electrodes voltage, in the low electric field range, where the common x-ray measurement method is not feasible. The SARAF RFQ is designed as a deuteron accelerator and therefore is potentially capable of producing four times the power needed for accelerating protons. This unusual condition enables us to measure the RFQ proton beam properties, not only up to the optimal power, but also in a wide range above the optimal power. Even though common measurement devices are limited to the beam core, due to the detection techniques sensitivity and resolution, the simulation in this work covers the beam tails as well. All beam dynamics simulations presented here were performed using the TRACK code [5].

Figure 1: The SARAF linac front-end setup, as used for the protons beam measurements and simulations in this paper.

The insert shows the location of the diagnostic devices downstream the RFQ that were used. BPM- beam position monitor. FFC- fast Faraday cup. MPCT- multi parameter current transformer.
MEASUREMENT AND SIMULATION SETUP
Typical 5 mA proton beams are analyzed and measured in the LEBT emittance apparatus and show an rms normalized transversal emittance of 0.15 π-mm-mrad at the location of measurement [2]. For the measurement described here the beam current was reduced to 3 mA (instantaneous current). The LEBT beam optics was simulated starting downstream of the ion source extraction electrodes with a 4D Waterbag distribution and normalized to generate an emittance of 0.15 π-mm-mrad at the emittance apparatus. We assume a beam waist at this starting point. In order to match the beam to the RFQ and reach the maximum measured transmission, the LEBT steerers were used in the measurement. In the simulation the solenoids were operated with magnetic fields as similar as possible to the measurements values, but steerers were not included. The beam was operated at a pulse mode in order not to exceed the intercepting diagnostic rated power of 200 W. The ion source and the RFQ RF power supplies (PS) are pulsed as described in [4] in order to shape the pulses. The beam current was measured in the LEBT, upstream of the RFQ entrance, using a Faraday cup. Downstream of the RFQ, the beam current was measured using the MEBT BPM sum signal and using the calibrated Bergoz MPCT at the D-plate (Fig.2). Beam energy was extracted from the time-of-flight between the two MEBT BPMs, used as phase probe pickups, and confirmed by a measurement in additional two phase probes at the D-plate [4]. The longitudinal bunch width was measured by Fast Faraday cups (FFCs) located 1.064 m and 2.649 m downstream the RFQ exit flange. The transversal beam profile was measured using a set of x/y wire scanners [4].

![Graph](image)

Figure 2: proton beam current, as function of the RFQ PS power, measured using the MPCT (blue) and the MEBT BPMs (green and purple). The red dashed curve presents the current predicted in the simulation, peaked at 32.0 kV.
RESULTS

The comparison between the simulated and measured beam current as a function of the RFQ power is presented in Fig.2. The simulation predicts well the BPM signal trend. However, the current value of the BPM is not calibrated. Similar trends are found in the SNS and ISIS RFQs measurements and simulations [6-8]. The MPCT overestimates the current at low power due to dark current that limits its sensitivity. The underestimation at high power might be due to beam loss in the D-plate. This assumption is supported by the measured wire scanner profiles.

When using LEBT operating conditions, the maximum transmission in the simulation is limited to ~70% due to beam mismatch at the RFQ entrance. For optimal matching the simulated transmission is 96%. In the measurement, the transmission is limited probably also due to misalignment found at this preliminary stage of the installation. The proton energy predicted by the simulation is low at low electrode voltage, reaches the design value of 1.5 MeV at 30.0 kV and stays constant at higher voltages. Similar step-function behavior was measured as a function of the RFQ power [9]. In the measurement, this designed proton energy is reached at a PS power of 55 kW. This PS forward power includes <1 kW losses along the supply line to the RFQ and ~3.5 kW of beam power, during the 3 mA injected protons pulse. The match of the simulated and the measured energy "step-functions" predicts that the design voltage of 32.5 kV is reached at 60 kW without beam load. It means that with a 3 mA injected beam load and ~70% transmission, 32.5 kV corresponds to a PS power of 63.5 kW.

The simulated and the measured longitudinal bunch profile at FFC1 are presented in Fig.3 for the optimal RFQ voltage/power. At these conditions the profile shape is a Gaussian with 1STD=10 deg. The beam measurements, with lower precision, have been performed also for an injected current of 0.5 mA and the simulations reported above have been repeated for this case. The transmission at low current is higher and goes up to 90% in the simulation. The time and energy distribution does not present a typical Gauss shape but a double peak distribution is observed, due to the reduction of the space charge effect.

The proton bunch width as function of the RFQ PS power is presented in Fig.4. There is a good agreement between the simulated and measured values at FFC1 and optimal RFQ power. At FFC2 and optimal RFQ power the simulated value is significantly higher than the measured value.

Figure 3: proton time distribution inside a bunch at FFC1. Time is presented as degrees of 176 MHz (360 deg=5.68 ns). Left: simulation of one bunch with 40,000 micro-particles. Right: an average of 100 measured bunches. Both taken at optimal RFQ voltage/power.
This is probably due to a smaller energy spread in reality (not yet measured) compared to the energy spread in the simulations. The origin for this discrepancy is not yet clear. We simulated the fact that the FFC measures only a small fraction (<1%) of the beam at the beam center. The result shows that at optimal RFQ power, this effect introduces a maximum error of 10%. The optimum operating power region of the RFQ was evaluated, taking into account minimum expected beam losses, maximum transmission and a low operating electrode voltage. The minimization of the expected losses along the superconducting accelerator downstream of the RFQ is achieved in the simulation by eliminating the low energy tail at the RFQ exit. This can be analyzed by minimizing the halo parameter defined in [10]. Our results show (Fig.5) that increasing the RFQ voltage above the design value will improve the beam quality. A similar analysis of the TRASCO RFQ is found in [11].

All the simulations presented above have been repeated using initial 4D Gaussian distributions at the ECR exit and gave very similar output behavior at the RFQ exit.
This behavior is explained by the low non linear space charge effects (needed to remix the distribution) and the larger separatix width in low current that enabled the beam to expand in longitudinal phase space [11]. The SARAF RFQ bunching and focusing strengths were optimized for deuterons at 5 mA [3]. In terms of the space charge effect, the optimal current for protons is then 2.5 mA (since protons have half the electric rigidity \(E/q\) of deuterons). This makes clear the origin of the beam characters at low intensity.

**CONCLUSION**

The benchmark between the simulation and measurement show that the RFQ model in our simulation can well predict the measured values. The simulation is used for calibrating unknown parameters such as the electrode voltage and is used for understanding the measurements. There are still several discrepancies between the simulation and the measurements that may be explained by misalignment inside and outside the RFQ, which is not taken into account in the simulation, and by the fact that the simulation starts with a Waterbag distribution at the ion source exit and not with a realistic distribution. These last differences can be improved after realignment and if the beam emittance will be specifically measured at the LEBT at the matching condition to the RFQ, and the results will be introduced to the simulation as a starting distribution.

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Cross Section Tuning

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INTRODUCTION

Generally available evaluated cross-section libraries were entirely based on differential cross-section measurements (and on nuclear-reaction models). In fact, the evaluators, as a matter of principle, have always been very careful to avoid "contamination" of their libraries by any integral-measurement information \(^{(1)}\). These libraries have thus been manifestly uncorrelated to integral responses. Even for the simplest critical assemblies, the deviations of the calculated reactivities from their respective measured values, in general, were considerably greater than the uncertainties in the latter. Results of an MCNP criticality validation for 31 benchmarks using the new ENDF/B-V11.0 nuclear data library \(^{(2,3)}\) show an incredible agreement between the calculated and the measured reactivities: 19 of the 31 results fall within a single standard deviation of the corresponding benchmark value for \(k_{\text{eff}}\), and only 6 differ by more than two standard deviations. This "improvement," though not a result of a rigorous adjustment exercise, clearly indicates that what is known as "tuning" has been applied to the cross sections in the evaluation process. Such tuning manifestly introduces response-parameter correlations. However, since it is an art rather than a science, it is difficult, if not impossible, to quantify the response-parameter correlations it introduces. In a recent work \(^{(4)}\) we have examined the cross-section adjustment by measured integral responses formalism and estimated the effect of such correlations on the adjusted parameters and responses, and on the uncertainties in these quantities. And in order to lay stress on this effect, and avoid unnecessary technical complications, we limited our discussion to a simple one-parameter/one-response criticality problem.

ONE RESPONSE CRITICALITY MODEL

In the one-group isotropic-scattering approximation of the neutron transport equation, two parameters determine the solutions of the equation for homogeneous uniform-density bare spheres. These parameters might well be the macroscopic total cross section \(\Sigma\), or in other words the sphere’s dimension in units of mean free paths, and the average collision multiplication \(c = \frac{\nu\sigma_f + 2\sigma_{n,2n} + 3\sigma_{n,3n} + ... + \sigma_s}{\sigma_{\text{tot}}}\).

It is plain to see that for any bare sphere, of which the medium is characterized by a given value of \(c\), the effective collision multiplication, or reactivity, is \(\gamma = cp\), where \(P\) is the non-leakage probability (i.e. the probability to make the next collision in the sphere).
In Figure 1, the non-leakage probability $P$ and its derivative are depicted as function of the sphere's radius in mean free paths. In our previous work, in which we limited our discussion to a simple one-parameter/one-response criticality problem, $c$ was the only parameter considered. However, since the total macroscopic cross section $\Sigma$ also has an uncertainty, we consider in this work uncertainties in both $c$ and $\Sigma$. These two uncertainties are assumed not to be correlated. On the other hand, the average collision multiplication $c$ and the sphere's "measured" multiplication $\gamma$ are assumed to be correlated. However, since in the tuning process the correlation is unknown, we vary this correlation between -1 and 1 to span the whole range of possibilities.

![Graph showing the non-leakage probability P and its derivative as function of the sphere's radius in mean free paths.]

**RESULTS**

In Figure 2 we compare the relative uncertainty in the sphere's "adjusted" multiplication (reactivity) $\gamma$, for adjustments performed with one ($c$), or two ($c, \Sigma$) parameters. In our previous work\(^{(4)}\), on the one-parameter/one-response criticality problem adjustment, we have shown that the relative uncertainties in both the adjusted response $\gamma$ and in the adjusted parameter $c$ are equal and fully correlated. Moreover, they are equal to zero for a full prior correlation or full prior anti-correlation and are always smaller than the relative uncertainty in the "measured" reactivity $\gamma$. Having more than one parameter the relative uncertainty in the adjusted $\gamma$ does not vanish for any prior correlation and the uncertainties in the adjusted parameters are reduced for any prior response-parameter correlation. Unlike the one parameter/one response adjustment, the posterior response-parameter correlations are not constant and are not equal to one, this is clearly seen in Figure 3.
CONCLUSIONS
We have studied adjustment in the elementary case of the one-group bare critical sphere, and demonstrated that even when response-parameter correlations are known to be present, but are quantitatively unknown, adjustment is still advisable and practical. Considering that the main purpose of adjustment is to generate parameters leading to more reliable calculations, i.e., reducing uncertainties in calculated responses, we have again shown that, to the extent that the measured responses are accurate (compared to the reliability of the calculated responses), and whatever the assumed correlations are, adjustment will result in better libraries.

Figure 2. Comparison of the new relative uncertainty in the sphere’s "adjusted" multiplication \( \gamma \) as function of the assumed prior correlation between \( c \) and the "measured" \( \gamma \), for adjustments considering uncertainties in one \( (c) \) or two \( (c, \Sigma) \) parameters. The relative uncertainty in the "measured" \( \gamma \) is 0.001.
Figure 3. The new correlation between the uncertainties in the adjusted $c$, the average collision multiplication, and the uncertainty in the sphere's "adjusted" multiplication $\gamma$ and the new correlation between the uncertainty in the adjusted radius $R$ (in units of mean free path, i.e. uncertainty in $\Sigma$) and the uncertainty in the sphere's "adjusted" multiplication $\gamma$, as function of the assumed prior correlation between $c$ and the "measured" $\gamma$.

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High Pressure Study of Intermetallic Compound Hf(10)B2 and TDPAC Studies of Radiation Damage

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ABSTRACT. Irradiated Hf10B2 was studied as a function of the annealing temperature up to 1100°C, utilizing the TDPAC method. Three different sets of frequencies were needed to fit the data at all temperatures having the same \( V_{zz} \) and \( \eta \). \( V_{zz}^{1} = 13.1 \times 10^{17} \text{ V/cm}^2 \) (\( \eta = 0.26 \)) is attributed to the hexagonal symmetry of the lattice while \( V_{zz}^{2} = 21.9 \times 10^{17} \text{ V/cm}^2 \) and \( V_{zz}^{3} = 13.1 \times 10^{17} \text{ V/cm}^2 \) are due to the damage induced by the neutron irradiation. The Electric Field Gradients (EFGs) fractions (\( f_{1} \)) changed during the isochronal annealing were \( f_{1} \) increases from 20% at 100°C to 35% at 1100°C and \( f_{2} \) decreases from 70% to 55% while \( f_{3} \) remains constant, 10%. The c/a ratio as function of pressure shows a linear reduction tendency. Fitting the data with the B-M equation to determine the \( B_0 \) as \( 232 \pm 13 \text{ GPa} \) with \( B_0' \) fixed at 4. There is no indication for phase transition in the measured reign.

Key Words: High-Pressure, hafnium diboride, radiation damage, quenching.

1. INTRODUCTION

HfB2 compound has nuclear, mechanical and physical properties making it a suitable candidate for a variety of applications in the nuclear industry. This is due to the large cross section of the \( ^{10}\text{B} \) for thermal neutron absorption, high erosion resistance and high melting temperature. Indeed, the \( ^{10}\text{B} (\text{n},\alpha)^{7}\text{Li} \) reaction generates recoiled He and Li atoms with an average kinetic energy of 1.48MeV and 0.83MeV respectively. The recoiled atoms together with the neutron scattering process may cause degradation in the bulk properties of the HfB2. The present work aimed to study the irradiation damages in Hf10B2 utilizing the TDPAC method and is part of a program to study neutron irradiation damages in HfB2 by comparing the Hf10B2 with Hf1B2.

2. EXPERIMENT

Sample Preparation. Hf was irradiated in a reactor to produce the radioactive isotope \( ^{181}\text{Hf} \) with an activity of \( \sim 45\mu\text{Ci} \) and arc-melted in a furnace with \( ^{10}\text{B} \) powder to produce HfB2. From Rietveld analysis we ensured a typical purity of \( \sim 98\% \) of HfB2 with main impurities of boron, hafnium metal and hafnium oxide. During annealing stages samples were sealed in an evacuated quartz capsule. The HfB2 was irradiated with a neutron dose of \( 2 \times 10^{16} \text{ n/cm}^2 \).

TDPAC measurements. The above samples were measured using the \( ^{181}\text{Ta} \) 133-482 KeV cascade with an intermediate state half-life of 10.8ns. This cascade is obtained via the decay of the 18µs 615 KeV state in the \( ^{181}\text{Ta} \) probe, populated in the \( \beta \) decay of \( ^{181}\text{Hf} \rightarrow ^{181}\text{Ta} \). The time spectra at \( \pi \) and \( \pi/2 \) were recorded simultaneously using a four BaF2 detector system with time resolution of 600ps and conventional fast-slow electronics coupled to a multi-parameter acquisition system.
High Pressure measurements.
The high-pressure energy dispersive X-ray diffraction studies were taken at the X17-C beamline of the National Synchrotron Light Source (NSLS) \(^{(1)}\). The energy dispersive data was collected with a high pure germanium detector at a fixed Bragg angle \((2\theta=12^\circ)\). The high-pressure X-ray powder diffraction measurements were taken at discrete pressure steps in the range of 0-30.8 GPa. The data was collected by the EDS technique, using the white beam of the superconducting wiggler at the X17-C beamline. Typical data collection time was about 1 minute. In the lower range of the High-Pressure range, angle dispersive measurements where carried out in transmission configuration using the image plate technique. The data was analyzed using a commercial Rietveld analysis software packages \(^{(2,3)}\).

The pressure (up to 30.8 GPa) was applied via a Merrill-Bassett type diamond anvil cell (DAC), "Tel-Aviv"-type \(^{(4)}\). The experiments were conducted using a polychromatic X-ray beam, and the pressure was measured utilizing the fluorescence of Ruby technique \(^{(5)}\). The sample dimensions for the high pressure measurements were about 50\(\mu\)m in height and 130\(\mu\)m in diameter. The beam size was about 30\(\mu\)mX50\(\mu\)m vibrating in the sample region. A diamond-anvil cell with 500\(\mu\)m culets was used, with silicon oil as a pressure medium in the sample cavity. A 200\(\mu\)m stainless L-301, foil was used as a metallic gasket between the two diamonds. The pressure distribution inside the sampling volume was checked at different regions, and was determined to vary by less than 5%.

3. RESULTS and ANALYSIS

Fig. 1 shows the TDPAC time spectra \(^{(6)}\) of the same sample obtained in the following processes: (a) after quenching from arc-melting to room temperature (b) after annealing at 1000°C for 60h (c) after neutron irradiation with a dose of \(2\times10^{16}\) n/cm\(^2\). The solid line is the theoretical fit of \(G_2(t)\) to the data. The fitted \(V_\parallel\) (Electric Field Gradient EFG) and \(\eta\) (asymmetry) parameters are given in table 1. After annealing, \(V_\parallel^2\) disappeared and only \(V_\parallel^1\) was left, as expected from the hexagonal structure of the HfB\(_2\). However the asymmetry parameter, \(\eta\), after the long annealing stage, was found to be 1. This result strongly indicates an incomplete annealing process. This may also explain the higher value of \(V_\parallel^1\) obtained at this stage compared to references \(^{(7,8,9)}\). The damage caused by neutron irradiation shows up via the additional two EFG components reappearing in the spectrum (Fig. 1; table 1).

Table 1: The fitted EFG and \(\eta\) values prior to the isochronal annealing study. The statistical errors are \(\pm1\times10^{16}\text{V/cm}^2\), see Fig. 2.

<table>
<thead>
<tr>
<th></th>
<th>(V_\parallel^1) V/cm(^2) ((\times10^{17}))</th>
<th>(\eta^1)</th>
<th>(V_\parallel^2) V/cm(^2) ((\times10^{17}))</th>
<th>(\eta^2)</th>
<th>(V_\parallel^3) V/cm(^2) ((\times10^{17}))</th>
<th>(\eta^3)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Quench(a)</td>
<td>12.9</td>
<td>0.2</td>
<td>22.7</td>
<td>0.1</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Anneal(b)</td>
<td>14.7</td>
<td>1</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Irrad. (c)</td>
<td>14.6</td>
<td>1</td>
<td>20.9</td>
<td>0.4</td>
<td>10.7</td>
<td>0.7</td>
</tr>
</tbody>
</table>

The irradiated HfB\(_2\) was isochronally annealed from 100 to 1100°C in steps of 100°C, and duration of 30 min. per step. Three EFGs were fitted with the following averages: \(V_\parallel^1=13.1\times10^{17}(1)\ (\eta^1=0.25)\), \(V_\parallel^2=21.9\times10^{17}(3)\ (\eta^2=0)\), \(V_\parallel^3=8.3\times10^{17}(1)\ (\eta^3=0.3)\) V/cm\(^2\). Figure 1 presents the time spectra for three representative annealing temperatures. The solid lines are the fits of the theoretical \(G_2(t)\) function to the data.

Figure 2 shows the fraction of each EFG during the isochronal annealing.
At the first annealing temperature of 100°C the fractions are re-ordered and then they remain constant up to 800°C. At this point $V^1_{zz}$ ($V^2_{zz}$) fraction begin to increase (decrease).

The X-Ray data as function of pressure is depicted in figure 4.
The difference in the reduction of the unit cell parameters as function of pressure yields a change in c/a ratio which is indicating the differences between the plains densities, the c/a ratio is changing from 1.10589(5) to 1.10016(5) at 0 and 30.8 GPa, respectively. The c/a ratio as function of pressure is shown in figure 5.
The c/a ratio as function of pressure shows a linear reduction tendency.
Fitting the data with the Vinet (10) equation to determine the $B_0$ as 232±13 GPa and $B_0'$ fixed at 4, figure 6.
4. Calculations

The calculations were done using the full potential linearized plane wave (LAPW) method, as embodied in the WIEN97 code. The unit cell in these calculations is divided into two parts, the atomic spheres and the interstitial region. The wave functions inside the atomic spheres are described by atomic-like functions, while in the interstitial region plane waves are used \(^{(11)}\). Exchange and correlation effects are treated within the density functional theory, using the generalized gradient approximation (GGA) \(^{(12)}\), where not only the local density, but also its gradient determines the magnitude of the effect. In most cases, the charge asymmetry inside the atomic sphere, where the \(V_{zz}\) value is being calculated, determines more than 90% of the \(V_{zz}\) value. Inside this sphere, the contribution to \(V_{zz}\) can be further divided into s-d, p-p and d-d components, so that the physical origin of the EFG can be analyzed \(^{(11)}\).

The calculated \(V_{zz}\) values at the hafnium and boron sites are compared in Table 2 with previously calculated results \(^{(13)}\) and with experimental \(^{181}\)Ta TDPAC values obtained in this work and in references \(^{(7,8,9)}\). The nonzero \(\eta\) value observed in our experimental work is indicative for the disorder that exists in our examined sample. In a single phase and well ordered HfB\(_2\) sample, the main principle of the EFG will be parallel to the c-axis, and a zero \(\eta\) value is expected. The discrepancy between the calculated and measured \(V_{zz}\) values at the hafnium site can be explained by the fact that TDPAC measurements are performed with a \(^{181}\)Ta probe situated in the hafnium site, while the calculations were performed on a pure HfB\(_2\) sample.

**Table II:** TDPAC measured \(V_{zz}\) value at the hafnium site and LAPW calculated \(V_{zz}\) value at the hafnium and boron sites in HfB\(_2\).

<table>
<thead>
<tr>
<th>The source of the result</th>
<th>Hafnium site</th>
<th>Boron site</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>(V_{zz}) (V/cm^3 \times 10^{17})</td>
<td>(\eta)</td>
</tr>
<tr>
<td>Calculated</td>
<td></td>
<td></td>
</tr>
<tr>
<td>This work</td>
<td>9.27</td>
<td>0</td>
</tr>
<tr>
<td>Hass work [13]</td>
<td>9.86</td>
<td>0</td>
</tr>
<tr>
<td>Measured</td>
<td></td>
<td></td>
</tr>
<tr>
<td>This work</td>
<td>13.1 ± 0.1</td>
<td>0.25</td>
</tr>
<tr>
<td>references [7,8,9]</td>
<td>12.9 ± 0.1</td>
<td>0</td>
</tr>
</tbody>
</table>

The partial valence s-d, p-p and d-d and the lattice contribution to the EFG at the hafnium and boron sites are listed in Table 3. From these results, it is evident that the major contribution to the \(V_{zz}\) value at the hafnium site comes from a p-p contribution next to the probe nucleus, with a relatively large d-d contribution of about 25%. This unusual large d-d contribution arises from the hafnium p-d electrons coupling shown in figure 5 at the energy of \(\approx 3\) eV below Fermi level.
The s, p and d-partial electron DOS at the hafnium and boron sites are plotted in figure 8. The results are similar to the ones reported by Vajeeston (14) who used the self-consistent tight binding linear muffin tin orbital (TB-LMTO) method. The major figure observed in this plot is the hafnium p-d and boron p hybridize at the energy of 2-5 eV below E_Fermi.

5. SUMMARY
The long annealing of the quenched sample with radioactive Hf and non-irradiated B (60h at 1000°C) should reveal the EFG of the HfB₂ hexagonal structure. The value \( V_{zz} = 14.7(3) \times 10^{-11} \text{V/cm} \) coupled with \( \eta = 1 \) obtained at this stage, indicates an incomplete annealing process. Indeed this EFG value is higher than those quoted at references (9,89). After the isochronal annealing study, \( V_{zz} \) was reduced to \( 13.1(1) \times 10^{-11} \text{V/cm} \) which is consistent with the previous publications. The asymmetry parameter \( \eta \) was reduced by the isochronal annealing process to 0.25.

After neutron irradiation of the sample, three different sets of frequencies were needed to fit the data. These EFGs are: \( V_{zz} = 13.1(1) \times 10^{-11} (\eta = 0.25) \), \( V_{zz} = 21.9(3) \times 10^{-11} (\eta = 0) \) and \( V_{zz} = 8.3(1) \times 10^{-11} (\eta = 0.3) \) \text{V/cm}². These values remain constant through the process of the sample up to 1100°C. \( V_{zz} \) is related to the crystallographic structure and it is higher than the LAPW based calculations \( 9.27 \times 10^{-11} \text{V/cm}² \) in this work and \( 9.86 \times 10^{-11} \text{V/cm}² \) in ref. (9).

The irradiation damage is expressed in displacement atoms due to recoil energies of the reaction products and neutron scattering. \( V_{zz} \) and \( V_{zz} \) can be related to these processes. Note that \( V_{zz} \) appears also in the quenched sample, which is a known phenomena in metal defects (16). During the first step of the isochronal annealing, at 100°C, the relative fractions of the three EFGs were significantly changed and re-ordered. This phenomenon is not clear by now and will be subject to further investigations. During all other isochronal stages \( V_{zz} \) fraction remains nearly constant. Up to 800°C \( V_{zz} \) and \( V_{zz} \) fractions were also almost constants. However, at 800°C \( V_{zz} \) fraction was decreased while \( V_{zz} \) fraction was increased, as can be expected by the disappearance of defects as a function of continues annealing. The stability of the \( V_{zz} \) fraction through the isochronal annealing in comparison to \( V_{zz} \) fraction indicates smaller migration energy for the latter related defects. Annealing of the sample at 1100°C for longer times is in progress to reach the full annealed stage.

The c/a ratio as function of pressure shows a linear reduction tendency. Fitting the data with the Vinet equation to determine the B₀ as 232±13 and B₀' fixed at 4.

There is no indication for phase transition in the measured reign.

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Absence of Low-Temperature Dependence of the Decay of 7Be and 198Au in Metallic Hosts

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INTRODUCTION
The issue of the possible dependence of β-decay and electron-capture (EC) rates of radioactive nuclei on the nature and temperature of the host matrix and environment is a long standing subject [1,2]. In particular, numerous recent experiments have claimed that the half-life of radioactive nuclei embedded in metals would be significantly affected by screening of the electrons in the metal and this effect further be strengthened at low temperatures [3-7]. Several of the most recent articles cite a longer half-life for the EC of 7Be (0.8±0.2% [8]) and a shorter half-life for the β+-decay of 22Na (1.2±0.2% [9]), where these nuclides were implanted in the metals Pd and In and cooled to T = 12 K. For the β+-decay of 198Au in a Au metallic environment the half-life was observed to be longer by 0.4±0.7% at room temperature and by 4.0±0.7% when the metal was cooled to T = 12 K, both compared to the literature value of the half-life [10]. For α-decay as well, the half-life of 210Po nuclei, located inside Cu metal at T = 12 K was measured to be shorter by 6.3±1.4% as compared to room temperature [11]. However, such effects could not be confirmed by other experiments to the accuracy level of parts of a percent [12-15].

On the theoretical side, we note that the Debye-Hückel screening model, used in previous publications to explain the apparent temperature dependence, is not applicable for a solid (strong-coupled plasma) but rather for a weak-coupled plasma at a very high temperature (thousands of degrees).

EXPERIMENTAL
We report on the simultaneous measurement of the temperature dependence of the EC decay rate of 7Be in metallic Cu host and of the β+-decay rate of 198Au in the host alloy Al-Au, ranging from 0.350 K to 293 K. The “double-source assembly” in the present experiment was based on the 2 mm diameter 7Be source, prepared by direct implantation at ISOLDE (CERN) onto a copper disk of 12 mm diameter and 1.5 mm thickness. The characteristics of this 7Be source have been provided in detail in Refs. [16-18]. The half-life 53.353(50) d of 7Be in copper was measured in a previous study [19]. Adjacent to the 2 mm spot of 7Be activity on the Cu disc, a 198Au (T1/2 = 2.6956(3) d [20]) source was attached. A 0.51 mm thick Al-Au (0.135% of Au) alloy wire of mass 13.7 mg was irradiated in the nuclear reactor at the Soreq Nuclear Research Centre, Israel, to produce 198Au by neutron activation - 197Au(n, γ)198Au. The 20 min. irradiation in the pneumatic transfer tube facility (“Rabbit”) of the reactor induced an activity of approximately 2.8×10^5 Bq at the end of irradiation, 1-2 days prior to the commencement of the γ measurements. The Ge detector was placed at 0° and 90°, alternatively, to the axis of the cryopump, mounted on a massive stand at a distance of 50 mm from the sample for both geometries, ensuring mechanical stability of better than 0.1 mm (Fig. 1(a)). The temperature was measured with a silicon diode, mounted close to the source and was recorded continuously.
RESULTS AND DISCUSSION

The count rates of $^7$Be and $^{198}$Au were calculated for every measured spectrum and corrected for decay using the published half-lives 53.353 and 2.6956 d. In this correction, the two reference times for the $^0$° and 90° measurements were the middle times of the first room-temperature runs in each of these measurements. The weighted averages of the ratios of count rates at different temperatures are presented in Fig. 2.

The individual normalized count rates of $^7$Be and $^{198}$Au at the $^0$° position for 293 K were: 2.8462(44) and 236.05(6). For 12.5 K the normalized rates were: 2.8121(29) and 232.59(20), respectively. These results demonstrate that at this position, the count rates of both radionuclides are lower by about $(1.20\pm0.18)$ to $(1.46\pm0.09)$% at the low-temperature (12.5 K) compared to the room temperature (293 K). This finding can be qualitatively explained by the geometrical contraction/expansion of the cold head of the cryopump (on which the copper disc was mounted) that is fixed by 70 mm long stainless steel rods that contracted during cooling down. The distance between the sources and the Ge detector was 50 mm and therefore the contraction of the rods increased the distance and subsequently induced a 1.2% decrease of the count rate, which agrees very well with the measured decrease 1.2 to 1.5%. This effect may have influenced the previous measurements [3,9]. The corresponding count rates of $^7$Be and $^{198}$Au at the 90° position for 293 K were 0.8030(22) and 94.53(24) and for 12.5 K these were 0.8092(29) and 95.57(44), respectively. This indicates an opposite finding at the 90° position, but with much smaller precisions.
A statistical test supports a null-effect, which is anticipated because the thermal contraction induces a negligible change in the distance between the detector and the sources. We emphasize again that these effects cancel out in very good approximation by the \(^7\text{Be}/^{198}\text{Au}\) ratios.

![Graphs showing the ratio of rates \(^7\text{Be}/^{198}\text{Au}\) at different temperatures and times.](image)

Figure 2. Top: Ratio of rates \((^7\text{Be}/^{198}\text{Au})\) at 0° for the temperature at 12.5 K and 293 K; Middle: Ratio of rates \((^7\text{Be}/^{198}\text{Au})\) at 90° for the temperature at 12.5 K and 293 K; Bottom: Ratio of rates \((^7\text{Be}/^{198}\text{Au})\) at 90° for the temperature at 4 K and 0.350 K. The solid line represents the weighted average and the broken lines correspond to a ±1σ interval.

The half-life of \(^{198}\text{Au}\) in the alloy Al-Au (0.135% of Au) was measured and a null-effect of (0.02±0.11)% was found at 12.5 K relative to 293 K. The count rate of \(^7\text{Be}\) in copper was measured at these temperatures relative to that of \(^{198}\text{Au}\). The normalized ratio of ratios (ratio \(^7\text{Be}/^{198}\text{Au}\) at 12.5 K to ratio \(^7\text{Be}/^{198}\text{Au}\) at 293 K), at the 0° and 90° are 1.0015(16) and 1.0032(58), which are equal within the experimental uncertainty, yielding a null temperature effect of (0.15±0.16)%, i.e., the half-life of \(^7\text{Be}\) at 12.5 K is shorter by 0.15±0.16 than the value of 53.353 d as was determined at 293 K [19].
The same ratios of \(^{7}\text{Be}\) to \(^{198}\text{Au}\) yields 0.0590(3) and 0.0589(3) at 0.350 K and 4 K, respectively. The ratio of ratios is therefore 1.0028(71), again a null temperature dependence of \((0.28\pm0.71)\)% for these two temperatures. We note that the Debye-Hückel screening model, used in previous publications to explain the apparent temperature dependence, is not applicable for a solid (strong-coupled plasma) but rather for a weak-coupled plasma.

**CONCLUSIONS**

In summary, the present decay rates of \(^{7}\text{Be}\) (EC) and \(^{198}\text{Au}\) (\(\beta\)) in the wide temperature range 293 to 0.350 K do not support earlier temperature-dependence observations for metallic hosts and validate the theoretical picture above.

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Radiation Damage Accumulation in Thin Metal Films:  
A Novel Alpha Emitter Technique

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The physical properties of irradiated materials have been subject of intense research during the last five decades [1]. Understanding the effects of radiation on materials is especially important in materials related to nuclear reactors and those that are used in the space industry. The goals of these studies are the characterization of the defects, understanding how defects are introduced and annealed as well as to quantify the changes in physical properties caused by those defects.

In most of these studies, samples are exposed to an external flux of radiation, such as neutrons, electrons or light ions. A different type of research problem involves the properties of self-irradiated materials such as in actinide metals. Self-irradiation is fundamentally different from external exposure in two main aspects: (i) Self-irradiation is a process that is present throughout the measurement. (ii) It involves two particle processes i.e., the alpha particle and the recoil atom. Experimental studies of self-irradiation are complicated by the highly radioactive and toxic nature of the materials involved.

Currently our theoretical understanding of the microscopic processes of radiation damage is based on phenomenological models and some molecular dynamics simulations.

In this work we develop a novel technique for studying radiation damage effects in thin metallic films irradiated by alpha emitting ions, which can be applied to the study of self-irradiation.

A long lived alpha emitting isotope \(^{224}\text{Ra}\) is collected on the tip of a metallic needle by the electrostatic ion collection method [2]. The tip is then placed above the sample, a thin metal film. As a result of the \(^{224}\text{Ra}\) decay, a short lived isotope \(^{220}\text{Rn}\) is emitted and implanted in the sample. This implanted \(^{220}\text{Rn}\) ion itself then undergoes alpha decay, causing internal irradiation of the sample. This type of irradiation mimics the effect of self-irradiation in a controlled manner.
Since the initial implantation of $^{220}\text{Rn}$ reaches typical depth of 100Å, the sample thickness should be of that order. Hence, we used a sample with a thickness of 200 Å. By collecting the radioactive ions on a tip one can achieve high density of activity needed for the measurement. The tip diameter which gave the highest density of activity was 500 microns. Therefore, for uniform irradiation, the sample width should be an order of magnitude smaller. The samples were prepared using photolithography technique in a Hall bar shape with 2 current terminals and 4 voltage probes [Fig. 1]. The sample has dimensions of 750μm X 50 μm.

![Figure 1: thin metal film in a Hall bar shape.](image)

The damage accumulation is measured by four probe electrical resistivity technique, i.e., constant ac current is driven between two current electrodes and the voltage drop is measured between the additional two electrodes by lock-in amplifier[Fig. 2]. The damage accumulation is indicated by an increase in resistivity. In practice the sample is cooled to a constant low temperature (~15K) by a cold head cryostat in order to prevent diffusion of defects. The resistivity is measured as a function of time.

The diffusion of defects in metals is characterized by several stages as function of temperature. These steps represent different mechanisms of damage annealing. The low T measurement should be done below the 1st stage, typically around 20K.
Figure 2: Schematic diagram of four probe resistivity measurements.

Figure 3: Damage accumulation in 200Å thick Platinum sample.
Results and discussion

Figure 3 shows the increase in electrical resistivity. The additional resistivity, $\Delta \rho$, is plotted as function of ion fluence, $\phi$, for a 200Å thick Platinum sample. The sample was irradiated at a temperature of 18K. The resistivity change rate $d\Delta \rho/d\phi$, is proportional to the damage accumulation rate.

Our preliminary results show a typical value for the rate of resistivity increase to be $8.26 \times 10^{-10}\ \mu\Omega\text{cm}$. This value is much larger than for a corresponding neutron irradiation measurement on 0.1mm bulk platinum sample [1]. We speculate that high damage accumulation rate arises from the damage caused by the recoil heavy ions in the cascade. We note that a single ion having these high energies induces typically couple of thousands of Frenkel pairs (vacancy and self interstitial). These defects are confined to a volume of a typical length of 200Å, whereas neutron irradiation induces typically 10-100 Frenkel pairs.

Acknowledgements

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Crystal and Exchange Fields in the Singlet Ground State System
TbCo3B2 Studied by Inelastic Neutron Scattering

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INTRODUCTION
Materials of the Rn1.0Co3n1.0B2n (R=lanthanides; n=1, 2, 3, infinity) family are derived from the RCo5 structure (P6/mmm space group), (n=0), by gradual substitution of B for Co on one of the Co sites, until complete substitution is achieved on that site for RCo3B2 (n=oo). In the past, these materials have attracted a lot of interest because of their potential use as permanent magnets. Magnetic structure and properties of the TbCo3B2 compounds were previously studied in our group.1,2 This paper will show that in this system, Tb3+ serves as a good example for a 4f electron shell in a singlet ground state, which is formed under the influence of crystalline electric field. Nevertheless this system orders magnetically at T~30 K where the resulted ground state magnetic structure at 1.5 K is ferrimagnetic, in which ordered Co and Tb moments are aligned in an anti-parallel formation, perpendicular to the unique axis.1 This magnetic ordering is achieved due to mixing of higher lying crystal field levels into the ground state by magnetic exchange interaction.3 Above 30 K, TbCo3B2 crystallizes in a hexagonal unit cell (space group P6/mmm) with lattice parameters a=b=5 Å and c=3 Å.1 The Tb3+ site symmetry is 6/mmm and the crystal field Hamiltonian (CEF) is

\[ H_{CEF} = B^0_2O_2^0 + B^0_4O_4^0 + B^6_6O_6^6 + B^6_8O_8^8 \]

Where the B_i and O_i are crystal field parameters and Stevens operators respectively.5 The non diagonal term, B^6_6O^6_6, mix the pure J_z states so that the eigenstates of (1) contain admixtures of different values of J_z's. An addition of an exchange term perpendicular to the unique axis (hexagonal (001) direction, i.e. J_x) will admix more J_z states into the ground state. The relative strength between the exchange interaction and the crystal field will determine the magnetic behavior of such induced moment systems. In the Weiss field approximation, this behavior can be well described by the Hamiltonian

\[ H = H_{CEF} + g_\mu_B H_m J_z \]

In this work we proved that in our system, the above mentioned mechanism manifests in quenching of the observed magnetic moment in the Tb3+ ion from its free ion value of \( \mu_{1B} = 9 \mu_B \) to approximately 5.2(1) \( \mu_B \).

RESULTS AND ANALYSIS
Figure 1 shows Inelastic neutron scattering (INS) data above (T=40, 80 and 200 K) and below (T=10 K) Tb-Tb ordering temperature with incident energy of E0=35 meV. Additional incident energy of E0=60 meV was used to verify that no signals appear at higher transfer energies. The experimental results were corrected using instrumental background measured at the same experimental conditions and put on absolute scale using Vanadium measurement.
The inelastic non magnetic background was determined by measuring the non magnetic analog YCo$_3$B$_2$ under identical experimental conditions (also in figure 1). The non magnetic background (e.g. phonons) is undetectable.

![Graph](image)

Figure 1 - Scattering law, $S(0,\omega)$, vs. energy transfer $\hbar\omega/2\pi$, with incident energy of $E_i=35$ meV, at four temperatures (see top left corner, dotted lines are guides to the eye). For $T=10$ K, Tb-Tb ordering took place.$^1$ All non-elastic signals are of magnetic origin, as was verified using measurement of the non magnetic analog YCo$_3$B$_2$ at 200K. Results were normalized to zero momentum transfer, $Q=0$, using magnetic form factor of the Tb$^{3+}$ ion$^6$ and put on absolute scale using Vanadium measurement. The measured momentum and angle regions are $Q\in(2.01,3.88)$ Å$^{-1}$ and $2\theta\in(12,68.4)$ deg.

Above Tb-Tb ordering temperature, three crystal field transitions are observed. Detailed balance$^7$ condition was used to verify the validity of the experiment and temperature readings, by comparing the intensities of energy gain and energy loss transitions. Below Tb-Tb ordering temperature, two transitions are observed at different energy transfers, suggesting different magnetic behavior (Hamiltonian (2)).

"Least squares" (LS) fitting algorithm, "FRILL", was used to simultaneously fit data for all four temperatures. The procedure was based on the cross section for magnetic scattering of an isolated ion,$^7$ CEF and Weiss fields in equations (1) and (2), respectively. Resulted fitting parameters are presented in table 1.

Table 1 - Parameters of the CEF and molecular field generated by a least squares fit to data. Numbers in parenthesis represent the statistical error on the last figure.

<table>
<thead>
<tr>
<th>$\chi^2$</th>
<th>$B_2^{(0)}$ (meV)</th>
<th>$B_4^{(0)}$ (meV)</th>
<th>$B_6^{(0)}$ (meV)</th>
<th>$B_6^{(6)}$ (meV)</th>
<th>$\mu_B H_m$ (meV)</th>
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<td>8.99</td>
<td>1.40(3)</td>
<td>-6(1)×10$^{-4}$</td>
<td>3.1(5)×10$^{-5}$</td>
<td>4.0(4)×10$^{-5}$</td>
<td>5.0(1)×10$^{-5}$</td>
</tr>
</tbody>
</table>
DISCUSSION
The quantum energy eigenvalues and eigenstates, above and below the transition temperature, are summarized in figure 2.

![Energy levels scheme](image)

Figure 2 - Energy levels scheme of each level for the Tb$^{3+}$ ion in the TbCo$_2$B$_2$ system. Above (left) and below (right) the transition temperature, the quantum wells are described by equations (1) and (2) respectively. Double arrows on the levels scheme represent the experimentally observed transitions. Levels separated by less than 1 meV were considered as doublets. The ground state above Tb-Tb ordering was chosen as the zero energy level.

Using the ground state eigenstate below Tb-Tb ordering temperature, we calculate a magnetic moment of $\mu_{\text{Tb}}=5.6(3) \mu_B$ on the hexagonal plane. In previous work, using neutron diffraction (ND), we measured a moment of $\mu_{\text{Tb}}=5.2(1) \mu_B$. The two values agree within the uncertainty limit.

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Magnetic and Crystallographic ND Study of Tb$_{1-x}$Y$_x$Co$_3$B$_2$

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INTRODUCTION
Recently, the magnetic behavior of a polycrystalline sample of TbCo3B2 has been studied.$^{(1)}$ This compound, paramagnetic at room temperature (space group P6/mmm), undergoes a high temperature [$T_C = 170(15)$K] ferromagnetic ordering with moments aligned along the c axis, and a lower temperature [$TSRT = 30(3)$K] spin-reorientation transition (SRT), with moments rotating towards the basal ab plane as temperature is lowered. The high temperature transition was attributed to the Co-Co exchange interaction while the lower temperature one was attributed to the Tb-Tb exchange interaction. In addition, below TSRT a symmetry decrease sets in, leading to an orthorhombic distortion of the crystal lattice. Attributing the high temperature transition to the Co-Co exchange interaction was supported by the study of YCo$_3$B$_2$.$^{(2)}$ Where only the $T_C$ transition was observed. In order to decouple the Co-Co magnetic transition from the Tb-Tb magnetic transition, 5 samples of the Tb$_{1-x}$Y$_x$Co$_3$B$_2$ family (with $x = 0.05, 0.1, 0.25, 0.4,$ and 0.5) have been prepared and studied by means of Neutron Powder diffraction (NPD).

EXPERIMENTAL DETAILS
The samples were prepared by arc melting the constituents in an argon atmosphere followed by a subsequent annealing at 1270 K in vacuum for 117 h. Xray diffraction show high sample quality. NPD measurements were carried out at 26 different temperature points (4.6 - 300 K) on both the Medium Resolution Neutron Powder Diffractometer (MRPD) and the High Resolution Neutron Powder Diffractometer (HRPD) at the High Flux Australian Reactor operated by Australian Nuclear Science and Technology Organization. The MRPD ($\lambda = 1.665(1)$ Å, $2\theta = 2 - 138^\circ$, step $0.1^\circ$) was preferred for magnetic data due to higher flux, while the HRPD ($\lambda = 1.492(1)$ Å, $2\theta = 10 - 150^\circ$, step $0.05^\circ$) was used to study the crystallographic properties in detail. Diffraction patterns at 16 different temperatures (6.5 – 300 K) were collected on the MRPD, and 10 different temperatures (4.6 – 240 K) were collected on the HRPD. The same powder sample (~10 g) was used on both instruments. NPD data were analyzed using the Rietveld refinement method with the FULLPROF code.$^{(3)}$
RESULTS AND ANALYSIS

A significant intensity increase in some of the NPD reflections (Fig. 1) where observed at the 3 low doped samples, i.e. x = 0.05, 0.1 and 0.25, at the lowest temperature measurement of each sample in comparison with the high temperature measurement. This increase in the intensity can be related to the appearance of either ferro- or ferrimagnetic ordering. Based on previously published data on TbCo$_2$B$_2$ (1) the NPD data collected at the MRPD below $T_{SRT}$ were refined using collinear magnetic model. According to this model, Tb sublattice is ferromagnetic and the Co sublattice is ferrimagnetic. Whereas, the magnetic axes of the two sublattices are allowed to be either parallel (ferromagnetic) or antiparallel (ferrimagnetic) and are aligned with an angle $\theta$ to the hexagonal axis c. This model was found in TbCo$_2$B$_2$ in the ferrimagnetic state. (1) This analysis yielded the temperature dependence of the Tb ordered magnetic moment (Fig. 2), as well as the angle $\theta$ (not shown) in all the samples which undergoes a phase transition. The $T_{SRT}$ linearly decreases as a function of the non magnetic atom Y as expected (Fig. 2). Furthermore, The analysis yielded a clear decreas of both the magnetic moment of a single Tb ion (Fig. 3), and the angle $\theta$ (not shown) as a function of Y doping.

![Diagram](image.png)

Fig. 1. MRPD diffractions of the Tb$_{0.75}$Y$_{0.25}$Co$_2$B$_2$ sample at 300K (top) and 6.5K (bottom), the data represented by points, Rietveld refinement profile by solid line, and their difference by the solid line at the bottom. Lines are indexed using the $P6/mmm$ notation.
Fig 2. The Tb ordered magnetic moment vs. Temperature in the 3 samples (x = 0.5, 0.1 and 0.25) which undergoes a phase transition.

Fig 3. The Magnetic moment of a single Tb ion (solid dots) and The T_{SRT} (open dots).
The basal plane component of the magnetic moment is inconsistent with hexagonal symmetry. (4) Therefore, SRT must be accompanied with a symmetry reduction similar to what was found in TbCo3B2,(3) where an orthorhombic (Cmmm) unit cell was used to describe the crystallographic. Furthermore, any value of $\theta$ except $\theta = 0^\circ$ and $\theta = 90^\circ$ is inconsistent with the orthorhombic Cmmm space group as well. (4) Thus, further symmetry reduction must be taken into consideration for the structure found below $T_{SRT}$. The maximal subgroup of Cmmm which is consistent with such value of $\theta$ is the monoclinic $C2/m$ space group. An attempt to use the orthorhombic $C2/m$ model for the crystal structure in the analysis of the MRPD data at 6.5K did not improve the fit. This is probably due to the limited resolution of the MRPD. The refinements of suggested structural models for the high resolution HRPD data clearly show that the crystal symmetry reduction to the $C2/m$ space group in all 3 samples below $T_{SRT}$.

CONCLUSIONS

In general, the magnetic properties of Tb$_x$Y$_{1-x}$Co$_3$B$_2$ presented here strongly support our model of the magnetic behavior of the RCo$_3$B$_2$ compounds. (1) It is shown here that $T_{SRT}$ strongly depends on R. Substitution of Tb with 25% of Y reduces $T_{SRT}$ from 30K for TbCo$_3$B$_2$ (1) to 17 K for Tb$_{0.75}$Y$_{0.25}$Co$_3$B$_2$. Adding this to the absence of measurable $T_{SRT}$ in YCo$_3$B$_2$ (2), strongly supports our conjecture that the spin reorientation is governed by the R-R exchange interaction. The decreasing value of $\theta$ as function of Y doping (From $\theta = 74(2)^\circ$ in the undoped compound to $\theta = 32(9)^\circ$ in the $x = 0.25$ compound) is easily explained by the magnetocrystalline anisotropy. As is discussed in our previous work (1), the magnetocrystalline anisotropy energy contributed from the Tb site favors alignment of magnetic moments perpendicular to the hexagonal unique axis c, while the magnetocrystalline anisotropy energy contributed from the Co site favors alignment of magnetic moments along the hexagonal unique axis c. Magnetocrystalline anisotropy energy strongly depends on the magnitude of the magnetic moment (Ref. 1, and references therein). Thus, the dilution of the magnetic site with the non-magnetic atom Y weakens the anisotropy contributed from the Tb site and as a result the ordered magnetic moments are aligned closer to the hexagonal unique axis.

NPD analyses show a considerable decrease in the magnitude of the magnetic moment of the Tb ion, $\mu_{TB}$, as a function of the non-magnetic atom Y (Fig. 3). It is difficult to see how the small change of the Crystalline Electric Field (CEF) due to Y doping will cause this significant reduction in $\mu_{TB}$. We postulate that for TbCo$_3$B$_2$ the relatively large exchange field below 31K admixes the CEF ground state, resulting in a finite, yet reduced $\mu_{TB}$. When Tb is diluted with the non-magnetic Y atom, the exchange field diminishes significantly, changing the admixture in a way that lowers $\mu_{TB}$. A study of inelastic neutron scattering is currently underway which may contribute to our understanding of the Y doping effect on the electric field in the Tb$_{1-x}$Y$_x$Co$_3$B$_2$ system.

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Possible Liquid-Liquid Phase Transition in Bismuth

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INTRODUCTION

The density and structure factor of liquid bismuth have been explored from melting point to 1100°C by high resolution measurements of the density and neutron diffraction. These physical properties display complex temperature dependence. In particular, we have observed the presence of a possible temperature driven liquid-liquid phase transition that takes place at ambient pressure. This liquid-liquid phase transition is characterized by an abrupt density discontinuity that occurs at 740°C. According to neutron diffraction measurements, the phase transition manifests itself in the static structure factor by the disappearance of a high q shoulder on the first peak and a shift in the position of the third peak. The presence of this transition, at ambient pressure, challenges the common understanding of pure liquid metals as simple fluids having properties that vary continuously with temperature.

RESULTS

Density measurements

Density measurements were performed using the γ ray attenuation method. The experimental setup of the measurements consists of a 137Cs high energy (662Kev) γ ray source of 35mCi. The density \( \rho \), at each point was calculated from the attenuation of the γ ray intensity, I, in which

\[
\rho = \frac{1}{\mu d} \ln \frac{I_0}{I},
\]

where \( \mu \) is the absorption coefficient per unit mass and d is the sample’s thickness. I0 was measured separately using the same experimental setup, in the absence of the sample. The sample was then mounted inside a heated quartz crucible with a 7mm X 7mm cross section. The density of liquid bismuth was measured in the temperature range from melting to 1000°C in discrete steps of 5°C. In order to achieve stability, the sample was allowed to attain thermal equilibrium before starting the measurement. The same procedure was implemented while measuring I0. At each measured point, several spectra were acquired to reduce statistical error. To obtain the final spectrum, the background spectrum was subtracted from the recorded I and I0 spectrum. The intensity (I or I0) was determined by the area under the 137Cs peak (positioned at 662Kev). Consequently, a shift in the peak’s position resulted in an error in the intensity determination. We define the centroid of a spectrum as \( \Sigma X_i Y_i \), where \( X_i \) represents the channel number (i.e. energy) and \( Y_i \) represents the number of counts in the particular channel. The centroid of the 137Cs peak, in all recorded spectra, were computed prior to the intensity calculations. The thermal stabilization of the CsI, solid state γ photons detector, lead to an overall minor shift (less than one channel in average) in the centroid’s position. Such a negligible shift in the peak’s centroid had a large contribution in reducing the error in I0, as well as I, hence decreasing the total error in the calculated density. The total measurement error was found to be about 0.1% in average. At approximately 740–5°C a discontinuous change in the density of approximately 1%, made its appearance, suggesting the presence of a liquid-liquid phase transition. This change is considerably larger than the experimental scatter of ca. 0.1% observed in our measurements and was verified in several experiments. The results are presented in figure 1.
FIG. 1: Density of liquid bismuth as a function of temperature. At approximately 740°C a density discontinuity of ca. 1% is observed. The error of the measured points includes the systematic and statistical errors and does not exceed 0.1%.

**Neutron Diffraction Measurement**

Bismuth is an ideal element for neutron scattering measurements due to its high coherent scattering cross section (9.1477 barns), its negligible incoherent scattering cross section (0.0084 barns) and a very small absorption cross section (0.0338 barns @ λ=1.8Å). Neutron diffraction measurements were conducted initially at Glass Liquid and Amorphous Diffractometer (GLAD) at the Intense Pulsed Neutron Source at the Argonne National Laboratory. Time of flight measurements were carried out at 7 different temperatures upon heating and 5 different temperatures upon cooling. Additional independent measurements were carried out at the 7C2 diffractometer at Laboratoire Leon Brillouin at Saclay in order to confirm and extend the results that had been obtained. In this latter case the diffraction of a monochromatic beam (λ= 0.007Å) was recorded. Measurements at 14 different temperatures were performed upon heating and 5 different temperatures upon cooling. In both experiments a high purity (5N) bismuth cylinder, 8mm in diameter and 30mm high, was used. The cylinder was mounted in a sealed quartz capsule. The quartz capsule (8mm inner diameter and 10mm outer diameter), containing the sample, was evacuated and filled with high purity He to approximately 300mm Hg after which it was sealed. Empty quartz capsule diffraction patterns were also determined, at two different temperatures in GLAD and seven different temperatures in 7C2. As a part of the data analysis procedure, these diffraction patterns were corrected for background, cell and furnace contributions, for multiple scattering and for inelastic effects \(^{(2,3,4,5)}\). In both cases, in order to validate absolute S(q) values, a diffraction pattern was separately determined for a vanadium cylinder. The temperature was measured by a thermocouple attached to the outer bottom of the capsule. Within the limit of experimental systematic errors, both the GLAD and 7C2 absolute S(q) data are in reasonable agreement over the whole temperature range. Moreover, both our measurements at melting temperature stand in good quantitative agreement with previous publications \(^{(6,7)}\).
In Fig. 2 we present the structure factor of liquid bismuth at several temperatures as measured by the 7C2 diffractometer. At and above melting the structure factor is characterized by a shoulder located on the high q side of the first peak, which is usually associated with directional bonding. At approximately 700°C and above, a qualitative change in the structure factor takes place and which manifests itself by:

(i) The disappearance of this shoulder, quantified in Fig. 3 middle panel.
(ii) Stabilization of the width of the first peak, measured at S(q)=1, which increases from ca. 0.7Å⁻¹ at melting to approximately 0.85Å⁻¹ (Fig. 3 Top panel).
(iii) A shift of 0.15Å⁻¹ in the position of the third peak (Fig. 3 Bottom panel).

These results were confirmed upon heating and cooling in both GLAD and 7C2 diffractometers (except (iii) which due to experimental noise could not be corroborated on the GLAD data). We therefore conclude that a transition in the temperature dependence of the structure factor occurs at ca. 700°C. The difference in the transition temperature as determined by neutron diffraction and density may arise from the dissimilarity in positioning the thermocouples in different measurements. We note in passing, that, unusually for liquid metals, the position of the first and the second peaks in the structure factor is independent of temperature over the entire measurement range.

![Graph](image)

**FIG. 2:** The structure factor S(q) of liquid bismuth at several temperatures as measured by the 7C2 diffractometer. A shoulder on the high q side of the first peak is clearly visible at low temperatures and disappears at higher temperature. The shift to higher q values of the third peak is evident as the density decreases.
FIG. 3: Temperature dependence of parameters characterizing the static structure factor. Top panel - width of the first peak at $S(q)=1$. Middle panel - difference of width of first peak between $S(q)=0.9$ and $S(q)=1$. Bottom panel - position of third peak.

CONCLUSIONS
In conclusion, we have observed a possible liquid-liquid phase transition at atmospheric pressure in liquid bismuth at a temperature well above the liquidus. The existence of this transition has been confirmed by density measurement and supported by a change in the temperature dependence of the peaks of the structure factor obtained from neutron diffraction. The discovery of this transition, at ambient pressure, challenges our fundamental understanding of liquid metals as simple fluids having properties that vary continuously with temperature.


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Measurements of Residual Radiation Spectra from Neutron Activation in a Medical Linear Accelerator

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Purpose: Linear accelerators are known to produce neutrons by the atomic reaction ($\gamma$,n). It is of an interest to measure the residual spectra of activated materials in the room.

Method and Materials: The accelerators used was an Elekta Synergy S unit with beam modulator. The measurements were done on both 6MV and 18MV photon energies at institute #1 and at 10 & 15 MV at institute # 2. The detector was connected to a laptop with QtmMCA software. The detector was calibrated using Co⁵⁷, Ba¹³³, Cs¹³⁷ sources. An Ir¹⁹² (at inst. 1) and Cs¹³⁷ (at inst. 2) was placed in the detection area for reference.

Results: There was no observable change in the spectra before and after the 6MV irradiation. Following the 18MV / 10MV / 15 MV radiations there was a significant peak at about 150KeV with half life of about 145 sec. The suspected isotope is one of the following: Sr⁷⁹, Fr⁸⁷, Cd⁴⁸, Pm⁶¹, Pr⁵⁹, Os⁶⁶. Another pick was about 1.5MeV, which probably belongs to Al²⁸.

Conclusion: There is no significant activation from 6MV photon beam. There is a significant activation following an 18MV / 10MV / 15 MV beam with relatively long half life (145 sec.). It should be taken into consideration when department policies are written. When using mixed beams set up, it should be preferred to treat the high energy beams first.

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Automated Calculation of TG-43 Parameters for a Brachytherapy Source

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Introduction: The AAPM TG-43 dosimetry formalism has become the standard methodology for calculating the dose distributions surrounding brachytherapy sources. The original TG-43 formalism was published in 1995. An updated refinement of the formalism was published in 2004. A program was coded in Fortran-95 for calculating the relative TG-43 parameters based on tabulated dosimetry data.

Methods: The program computes the TG-43 parameters given the dose distribution (expressed in Cartesian coordinates) surrounding a brachytherapy source. The dose distribution is stored in a data file. The parameter definitions were coded in software. The program reads the data file and in turn calculates the parameters defined in the updated TG-43 formalism. Program output is stored to data files. At this stage, basic numerical methods were applied. Linear and bilinear interpolation were used for converting from rectangular to polar coordinates. Trapezoidal integration was used to generate 1D anisotropy data by integrating the 3D dose distribution over a 4π steradian solid angle. For testing and validating the program, the dose distribution around an ideal brachytherapy source was calculated and tabulated using deterministic physics. A separate dose calculation program was written for this purpose. The doses are calculated by dividing the line source into smaller segments each assumed to be a geometric point. At each calculation point, the exponential attenuation of the dose in the medium from each segment was calculated individually and the inverse square law applied. The TG-43 parameter calculation program reads the files generated by the dose calculator program.

Results: The calculated TG-43 parameters were verified to be correct for ideal brachytherapy line sources.

Discussion: This software was written to fill a need for automating the calculation of the TG-43 parameters for photon-emitting brachytherapy sources. Although calculated data were used for generating the input data file for the TG-43 calculator program, it is immaterial how the input data are generated. The data could likewise be derived from measurements or Monte Carlo simulations. The interpolation and integration routines used in this version of the software are simple. In the future, polynomial or spline interpolation and more sophisticated integration routines may be implemented if added accuracy is required. At this stage, only relative TG-43 parameters are calculated by the program. In the future, the absolute dose rate constant will be calculated provided that the air kerma data are available for the isotope.
Conclusion: Thus far, TG-43 parameters were only calculated for ideal brachytherapy sources. Based on the preliminary findings, the authors are confident that with further refinements, the dose calculator program can be extended to generate dosimetry tables for real brachytherapy sources, accounting for attenuation in the encapsulation material. The dose calculator program could then serve as a secondary check for dosimetry data acquired from measurements and Monte Carlo simulations. Such capability will enable additional testing and validation of the TG-43 calculation program. Once validated, the TG-43 calculator can be used with confidence for calculating the TG-43 parameters for real brachytherapy sources.

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A Micro-MOSFET In Vivo Dosimetry System

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INTRODUCTION
Radiotherapy delivers doses to the target or region of interest base on treatment planning system calculations. In vivo Dosimetry allows the measurement of doses in specific organs while irradiation treatment is conducted. This Dosimetry is an essential tool for quality and quantity assurance in radiotherapy.

In Vivo dosimetry allows on-time treatment validation with no perturbation to treatment procedure. It has overcome the limitations of dimensions, spatial and temperature corrections and is well fitted for a wide range of energies. Recent developments in treatment planning and radiation delivery methods raise the demand for In Vivo dosimetry that becomes an integral part in radiotherapy.

This Dosimetry system measures the doses given to the skin and critical organs in a radiation field and adjacent to it. The results obtained from the In Vivo measurements compared to the doses estimated by the treatment planning system that is commercial software.

CONCLUSIONS
This project shows a pioneer local treatment routine that uses microMOSFET In vivo Dosimetry system. The measuring system (by Thomson & Nielsen Electronics Ltd. model TN-RD-70-W) consists of five microMOSFET dosimeters destined to measure the dose in versatile arrangements. The system was first calibrated to all the Linacs in Chaim Sheba medical center in Tel-Hashomer using an acrylic phantom that was primarily developed for IMRT QA procedures. Doses were measured in different patients during their radiation treatments.

The measured doses showed a good compatibility to the treatment plan conducted by a commercial analytic treatment planning program.
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The Use of a Mini-chamber for Small Photons Beam Dosimetry

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Radiotherapy, Medical Physics, Rabin Medical Physics, Petah Tikvah, Israel

INTRODUCTION

For Intensity Modulated Radiation Therapy and Stereotactic Radiotherapy, a technique for dosimetry of small MV-photon beams (below 4x4 cm²) is required.

Mazal et al. [1] presented the method and materials to perform the dosimetry of high energy (6-23 MV) circular photon beams with diameters ranging from 10 to 40 mm at the isocenter of linear accelerators, used in stereotactic radiosurgery. Mazal et al. [1] showed that the lack of electronic equilibrium was the source of the macroscopic characteristics of the small beams of high energy photons. Mazal et al. [1] marked special problems of different detectors in the small beams dosimetry: large chambers loose part of the signal and have a low resolution; diodes establish an artificial electronic equilibrium around the detector, giving a response higher than the real one under non-equilibrium conditions; small chambers have the problem of a lower signal, giving in many cases more noisy data. In consequence, the use of different detectors was recommended to analyze the accuracy of results: the dosimetry can be performed with small ion chamber (which has cavity dimensions much smaller than the beam dimensions) and the beam profile that can be measured by film dosimetry must be used to evaluate and to correct the accuracy of the ion chamber measurement.

The concept of IMRT succeeds in limiting damage to normal tissue, while high doses can be delivered to target volumes. To achieve conformal dose distributions, IMRT fields are composed of small subfields; the size of subfields is on the order of 1x1 cm². Leybovich et al. [2] performed comparison of 0.6, 0.125, and 0.009 cm³ ionization chambers for IMRT absolute dose verification. He observed that the chamber sensitivity drops with decreasing volume: sensitivity of 0.009 cm³ chamber was obtained about 60 times less than that of a 0.6 cm³ chamber. Leybovich et al. [2] observed also, that the effect of leakage on the measured charge is relatively greater for small chambers, and it is significant since the delivery of IMRT takes a relatively long time. Without leakage corrections, with 0.6 cm³ chamber the measured dose was equal to calculated within 0.5%; with 0.125 cm³ chamber the error of measurement was 2.6%; with 0.009 cm³ chamber the error was 16%.

In this work, we performed the dosimetry of small 6 MV photon beams below 4x4 cm² using the Wellhoffer-CC13 cylindrical mini-ion chamber of 0.13 cm³ as the reference dosimeter. Kodak XV2 film dosimetry were performed for check the obtained results. Empirical calculations were carried out in order to implement a clinical viable dosimetry method.
1 MATERIALS AND METHODS

1.1 Instruments

- SCANDITRONIX/WELLOFFER 0.13 cc cylindrical ion chamber type CC 13 (3)
  It is a standard chamber for clinical use in water phantoms, vented through waterproof silicon sleeve. Wall material is Shonka (tissue-like plastic). Total active length is 5.8 mm. The chamber dimensions are presented in Figure 2.1

![Figure 2.1: Dimensions of 0.13 cc SCANDITRONIX/WELLOFFER ion chamber](image)

We measured the leakage current for this chamber and NE 2570 Farmer Dosemeter. The chamber was exposed to a 6 MV photons beam and collected the charge of 19.715 nC. After 600 sec the charge reading was 19.789 nC. It means that the measured leakage current is $1.3 \times 10^{-14}$ C/sec.

- NE 2570 Farmer Dosemeter.
  The NE2570 Farmer is a battery-portable electrometer with a proven performance record for long-term stability and precision. The electrometer has entry of correction factors for pressure $P$, temperature $T$ and chamber response. According to specifications and measurements, the dosemeter leakage current is $1 \times 10^{-14}$ C/sec of range full scale.

- KODAK radiographic films.
  In this work we used two KODAK radiographic film types: X-OMAT V (4). X-OMAT V is a relatively low-speed film designed for verifying the orientation and for estimating patient dosage in radiotherapy procedures.

1.2 Dosimetry with ion chamber

1.2.1 Output Factor measurement

We performed dose measurements with the CC13 chamber for a 6 MV photon beam at field size: 10x10 cm², 8x8 cm², 6x6 cm², 5x5 cm², 4x4 cm², 3x3 cm², 2x2 cm², 1x1 cm². The measurements were performed in water phantom 15 mm deep, the $d_m$ for 10x10 cm².

We determined the smallest field size for which the chamber is useful, by irradiating rectangular fields of 4x6 cm² and 6x4 cm², 2x4 cm² and 4x2 cm², 1x3 cm² and 3x1 cm². For each pair of the rectangular fields we calculated the ratio of charge readings and determined the pairs for which the difference between the readings exceeded 1.5%.
1.2.2 Check of Inverse Square Law

Inverse Square Law should be used for dose ratio calculations at different SSD in water phantom, with the detector at depth $d_m$. We performed dose measurements at SSD [cm]: 100, 110, 120, 130, 140 for field sizes [cm²]: 3x3, 4x4, 6x6, 8x8, 10x10. At each SSD two measurements were performed: at the depth 1.5 cm (relative $d_m$ for a 10x10 cm²) and at the depth 5 cm (used for Mayneord Factor checking).

1.2.3 Mayneord Factor accuracy checking

The accuracy of the Mayneord Factor was checked according to the measurements performed as follow: At each SSD the PDD ($d = 5$ cm) was calculated by the ratio of chamber readings at $d_m = 1.5$ cm and at $d = 5$ cm. Three readings pairs were used for PDD($d = 5$ cm) ratio calculation and comparison with Mayneord Factor.

3. EXPERIMENTAL RESULTS

3.1 Dosimetry with ion chamber

According to the results, the 0.13cc chamber is useful for dosimetry of the field dimensions larger than 2 cm and field sizes larger than 2x4 cm² (accuracy better than 1.5%).

In the Figure 3.1 measured and calculated values of the Output Factor are shown.

![Graph 2: Output Factor results](image)

**Figure 3.1: Estimated and measured Output Factor results**

An empirical function for Output Factor based on our results was determined for a clinical environment. The difference between calculated and measured values obtained was less than 0.5%:

$$OF(r) = 47.5 \times [1 + 0.366 \times \log(100 \times r + 20)].$$

A check based on film dosimetry, using XV2 Kodak film have shown similar results.
4 COMMENTS AND DISCUSSIONS

As we can see in

Figure 3.1, the empirical function correlates well with the calculated results for small fields in (the differences are less than 0.5%). Further work to complement this study has been carried out by means of film dosimetry.

5 REFERENCES:


4. KODAK publication: X-OMAT V Film / 4508 Technical Information Data Sheet, Revised 3-94.

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**Farmer Ion Chamber Dosimeter Response in Strong Magnetic Fields**

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² Oncology Division, Radiotherapy Department, Chaim Sheba Medical Centre, Tel Hashomer, Israel

**INTRODUCTION**

The interest in applying strong magnetic field in radiotherapy has grown recently. Linac or Cobalt machine coupling with an MRI facility for Image Guided Radiotherapy IGR(1) and for confined dose distribution(2), have been suggested in many publications. Ion chambers dose response should be affected due to the modified electron trajectories by the applied magnetic field. Ion chambers are very common in radiotherapy dosimetry. This work inspects the dependence of ion chamber response to radiation in magnetic fields. This study shows a "Farmer" type ion chamber 0.6 cm³ dose responses to a 20 mCi Cs-137 gamma source under the influence strong magnetic fields of a few Tesla. The response was measured in 0.5, 1.5 and 3.0 Tesla fields originated by GE™ MRI facilities with respect to the dosimeter angular orientation. This orientation maintained by a staging device allowed deferent field to chamber central line measurements, while keeping the source to chamber central axis constant.

**SETUP**

a. Ion chamber

The dose reading in IC is obtained through the charge collected and analyzed by the electrometer with respect to temperature, barometric pressure and recombination. In our work a "Farmer" type chamber 0.6 cm³ (PTW manufactured) was investigated.

The chamber is composed of a 1.1 mm Aluminum electrode, a 0.335 mm outer PMMA (C₅H₈O₂)ₙ wall, and a graphite peripheral electrode 0.09 mm inner layer. The densities of the PMMA and graphite are 1.19 g/cm³ and 1.85 g/cm³, respectively.

Figure 1: Right, PTW made "Farmer" type design dimensions, taken with permission from PTW-Freiburg. Left, the simulation geometry setup view, plotted with CGview program. All dimensions and materials were taken with permission from PTW-Freiburg.
a. Magnetic fields and radiation source
A 20 mCi Cs-137 source pellet with 1 cm long and 0.3 cm radius used. The gamma source was kept in shield constantly and placed perpendicularly to the chamber central axis by the staging device. Figure 1 shows a scaled sketch of source and chamber locations. Since the source and the chamber central reference point lay on a central line pivot their positioning maintained still while chamber was tilted. A 30 degree tilting steps of the chamber allowed measurements in different magnetic field directions. During irradiation the shield cover was removed, fulfilling radiation protection restrictions and regulations.

RESULTS
Results show a growing charge collection where magnetic field is present with dependence to magnetic field direction. In order to compare between the results, correction was needed according the barometric pressure and temperature. Thus, all results were corrected according to STP parameters. The results presentation of the open MRI 0.5 T, and the close MRI of 1.5 and 3 T were separate due to yield obtaining from reflection by structure MRI materials. For this reason, references with no presence of B field were measured inside an out-of-order 2 T MRI manufactured by Elscint. References regarding the open MRI 0.5 T was taken inside a brachytherapy room.

![Graph showing ion chamber charge collection as a function of 0.5 Tesla B field directions](image)

Figure 2: Ion chamber charge collection as a function of 0.5 Tesla B field directions

Figure 2 shows electric charge collection in pico-Coulombs, obtained by irradiation of 100 sec periods, versus the field to chamber central axis opening degree.
Figure 3: Ion chamber charge collection in 1.5 and 3.0 Tesla B fields vs. field to chamber central axis opening degree.

In all the cases where magnetic field is present the dose obtained by the dosimeter is increased. For the 0.5 and 1.5 T field the charge collected was growing when the field direction became more transverse to the chamber central line. For the 3.0 T field the tendency is kept for a 30 degrees opening angle. Then its growth reduces until it reaches some sort of saturation.

CONCLUSIONS and FUTURE WORK

Magnetic fields presence increases the ion-chamber dose readings. A buildup distance shortening when magnetic field is present and electron return effect are playing major role in the chamber reading modifications. A reduction in contribution of some electrons can be compensated by others reaching the cavity from surrounding build-up region. The contributing effect to the higher ionization formed, leading to a higher electrometer reading is referred to the electrons feasibility to maintain larger trajectories within the air cavity, or to form more electrons that reach cavity and collected by the electric field between the chamber electrodes.

A FLUKA Monte-Carlo simulation was carried out to further investigation of these results allowing future development of magnetic field approval ion chambers.

REFERENCES


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Lateral Spine X-Ray Diagnostic Examination: Left or Right Projection?

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INTRODUCTION

The total annual number of spine x-ray diagnostic examinations for adults and children in Israel amounts to approximately 800,000 procedures in 2007. The common procedure usually covers one front/back position and one lateral position. Cervical spine examinations are usually performed when the patient stands on his feet but thoracic and lumbar spine examinations are usually performed in a horizontal position. This fact influences the patient's effective dose during spine examinations. The current work focuses on lateral spine examinations.

A good x-ray radiographer will put the expected body findings near the image receptor. The same distance between the spine and the image receptor will be preserved on both lateral sides. There is no Israeli standard regarding the performance of lateral spine x-ray examinations. Every hospital chooses its own method – right lateral projection (R LAT) or left lateral projection (L LAT), or according to what the radiographer sees fit. Different hospitals perform the lateral spine x-ray examination when the patient lies on his left side and the radiation enters through his right side (i.e., L LAT position). Others perform this examination when the patient lies on his right side and the radiation enters through his left side (i.e., R LAT position). The basic factor that influences the lateral spine examination direction is what the radiographer prefers. A patient in a standing position does not care if the cervical spine examinations are performed in the L LAT or R LAT position. When the patient lies on his side, it is easier for the radiographer to perform the lateral examination with the patient's back near him. Without referring to any special needs and any national guidelines, the structure of different x-ray rooms, i.e. the location of the bed related to the room entrance and the control panel constitute major parameters that influence patients' positioning during spine examinations: lying on their right side or lying on their left side.

The largest radiation absorption can be found on the side where the radiation enters the body. From that point onwards towards the image receptor, there is a photon flux reduction inside the body as a result from the Compton and Photoelectric effects. Since the body organs aren't distributed symmetrically on one side of the body, the organs which are located close to the area where the radiation entered the body are exposed to a larger radiation dose than the organs on the other side. The unsymmetrical positioning of organs inside the body, together with the different radiation sensitivity of each organ, the photon energy, the patient's body size and the irradiation direction, constitute the major factors that influence the patient's effective dose.
This work deals with lateral exposure directions in spine x-ray examinations at a sample of 10 Israeli hospitals, the influence of the right and left lateral exposure on the radiation's effective dose and recommendations for the preferable exposure directions in consideration with the patient's safety in terms of radiation.

**RESULTS**

Observation has been made in radiology departments in 10 medium and large hospitals regarding the preferable right or left lateral projection in diagnostic spine x-ray examinations. That observation showed that 30% of the cervical spine examinations were performed at an L LAT projection and the remaining 70% have no preferable direction. In 70% of the thoracic and lumbar spine examinations, L LAT projections were used and the remaining 30% were comprised of R LAT projections.

Calculations of the radiation dose for these examinations are shown in table 1, using adult conversion coefficients to provide an effective dose from Dose-Area product\(^{(1)}\) (mSv/Gy*cm\(^2\)) and actual kVp and Dose*Area product\(^{(2)}\).

<table>
<thead>
<tr>
<th></th>
<th>Cervical Spine</th>
<th>Thoracic Spine</th>
<th>Lumbar Spine</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>L Lat</td>
<td>R Lat</td>
<td>L Lat</td>
</tr>
<tr>
<td>KVP</td>
<td>70.0</td>
<td>70.0</td>
<td>75.0</td>
</tr>
<tr>
<td>Filtration</td>
<td>2.5</td>
<td>2.5</td>
<td>2.5</td>
</tr>
<tr>
<td>Dose Area (cGy*cm)</td>
<td>75.0</td>
<td>75.0</td>
<td>516.0</td>
</tr>
<tr>
<td>Ovaries</td>
<td>0.00</td>
<td>0.00</td>
<td>0.00</td>
</tr>
<tr>
<td>Testes</td>
<td>0.00</td>
<td>0.00</td>
<td>0.00</td>
</tr>
<tr>
<td>Lungs</td>
<td>0.02</td>
<td>0.02</td>
<td>1.93</td>
</tr>
<tr>
<td>Stomach</td>
<td>0.00</td>
<td>0.00</td>
<td>0.05</td>
</tr>
<tr>
<td>LLI</td>
<td>0.00</td>
<td>0.00</td>
<td>0.00</td>
</tr>
<tr>
<td>RBM</td>
<td>0.04</td>
<td>0.04</td>
<td>0.45</td>
</tr>
<tr>
<td>Thyroid</td>
<td>0.10</td>
<td>0.10</td>
<td>0.10</td>
</tr>
<tr>
<td>Breasts</td>
<td>0.00</td>
<td>0.00</td>
<td>0.09</td>
</tr>
<tr>
<td>Esophagus</td>
<td>0.06</td>
<td>0.10</td>
<td>0.69</td>
</tr>
<tr>
<td>Liver</td>
<td>0.00</td>
<td>0.00</td>
<td>1.20</td>
</tr>
<tr>
<td>U Bladder</td>
<td>0.00</td>
<td>0.00</td>
<td>0.00</td>
</tr>
<tr>
<td>Skin</td>
<td>0.11</td>
<td>0.11</td>
<td>0.70</td>
</tr>
<tr>
<td>Tot Bone</td>
<td>0.16</td>
<td>0.16</td>
<td>1.14</td>
</tr>
<tr>
<td>Brain</td>
<td>0.05</td>
<td>0.05</td>
<td>0.01</td>
</tr>
<tr>
<td>Thymus</td>
<td>0.01</td>
<td>0.01</td>
<td>0.24</td>
</tr>
<tr>
<td>Adrenals</td>
<td>0.00</td>
<td>0.00</td>
<td>0.67</td>
</tr>
<tr>
<td>Pancreas</td>
<td>0.00</td>
<td>0.00</td>
<td>0.24</td>
</tr>
<tr>
<td>Spleen</td>
<td>0.00</td>
<td>0.00</td>
<td>0.06</td>
</tr>
<tr>
<td>Kidneys</td>
<td>0.00</td>
<td>0.00</td>
<td>0.16</td>
</tr>
<tr>
<td>Small Int.</td>
<td>0.00</td>
<td>0.00</td>
<td>0.01</td>
</tr>
<tr>
<td>ULI</td>
<td>0.00</td>
<td>0.00</td>
<td>0.01</td>
</tr>
<tr>
<td>Uterus</td>
<td>0.00</td>
<td>0.00</td>
<td>0.00</td>
</tr>
<tr>
<td>Muscle</td>
<td>0.06</td>
<td>0.06</td>
<td>0.36</td>
</tr>
<tr>
<td>Effective Dose (mSv)</td>
<td>0.020</td>
<td>0.022</td>
<td>0.432</td>
</tr>
</tbody>
</table>

**Table 1.** Organ dose (mSv) and effective dose (mSv) in spine x-ray examinations
The above data clearly show that the L LAT projection induces the lowest x-ray effective dose. The calculated effective dose in R LAT cervical spine and thoracic spine projections is 9% larger than the L LAT projections. The effective dose of R LAT lumbar spine projections is 60% larger than the L LAT projections. The organs that distinguish the R and L LAT effective dose are the stomach, LLI, esophagus, pancreas and spleen which induce a larger dose in the R LAT projection, and the liver and the ULI which induce a larger radiation dose in the L LAT projection.

A comparison between the L LAT and R LAT lumbar spine effective dose conversion coefficients differences in adults\(^1\) and children\(^3\), shows a larger impact in children than adults, up to 79% for 10 year old children and 86% for 15 year olds.

<table>
<thead>
<tr>
<th>Age (y)</th>
<th>kV</th>
<th>R LAT</th>
<th>L LAT</th>
<th>R/L Ratio</th>
</tr>
</thead>
<tbody>
<tr>
<td>10</td>
<td>70</td>
<td>0.322</td>
<td>0.18</td>
<td>1.79</td>
</tr>
<tr>
<td>15</td>
<td>80</td>
<td>0.214</td>
<td>0.115</td>
<td>1.86</td>
</tr>
<tr>
<td>Adult</td>
<td>85</td>
<td>0.119</td>
<td>0.074</td>
<td>1.60</td>
</tr>
</tbody>
</table>

Table 2. Conversion coefficients comparison, to give an effective dose from a Dose-Area product (mSv/Gy\(^2\)cm).

An assessment of the national potential for radiation effective dose reduction, if all the Israeli spine examinations will be performed using an L LAT projection is presented in table 3. This table shows an annual reduction potential of 57 man*Sv.

<table>
<thead>
<tr>
<th>Annual No. of Examinations</th>
<th>Cervical Spine</th>
<th>Thoracic Spine</th>
<th>Lumbar Spine</th>
</tr>
</thead>
<tbody>
<tr>
<td>R/L LAT Examinations Ratio (%)</td>
<td>35</td>
<td>30</td>
<td>30</td>
</tr>
<tr>
<td>R LAT minus L LAT dose (mSv)</td>
<td>0.002</td>
<td>0.038</td>
<td>0.435</td>
</tr>
<tr>
<td>Reduction Potential (man*Sv)</td>
<td>0.17</td>
<td>1.71</td>
<td>55.46</td>
</tr>
<tr>
<td>Total Reduction Potential (man*Sv)</td>
<td>57.3</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Table 3. Annual reduction potential of the Israeli collective dose if all lateral spine x-ray examinations were performed using an L LAT projection.

The second effect that influences a patient's dose during lateral x-ray spine examinations is the Heel effect. This effect refers to the reduction of the x-ray intensity (up to about 15% in radiology x-ray equipment) toward the anode side of the x-ray field. By proper use of this effect, when possible, the radiographer can reduce the radiation parameters and dose by alignment of the patient's thicker part toward the cathode and the thinner part toward the anode. The thicker parts of the body in spine examinations are the pelvis in a lumbar spine examination and the shoulders in a thoracic spine examination. In most Israeli x-ray examinations, this effect doesn't take into consideration the patient's decision regarding positioning.
CONCLUSIONS

1. In most of the 10 Israeli hospital samples, the decision about the spine lateral x-ray projection side (left or right) is taken according to the room structure and the radiographer's preferences. There aren’t any national guidelines regarding the patient's bed position in the x-ray room related to the patient's room entrance.

2. According to the body organs' distribution and sensitivity, when possible, lateral x-ray examinations should be taken using the L LAT projection (i.e., the radiation entry side is right). The R LAT projection can cause an effective dose excess of up to 60% in lumbar spine examinations and up to 9% in cervical and thoracic spine examinations.

3. Compared to adults, pediatric patients get a larger effective dose in R LAT spine examinations than in the L LAT side (up to 86% more in R LAT lumbar spine examinations for 15 year old patients).

4. If all the Israeli spine x-ray examinations would have been done using an L LAT projection, the annual collective dose reduction would be 57 man*Sv (0.6% of the annual Israeli medical x-ray collective dose). Using the Heel effect properly can cause another reduction in the patient's effective dose.

REFERENCES


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A Three Year Undergraduate Program (B.SC) in Medical Radiation Physics in the Ariel University Center of Samaria

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GENERAL
A three year undergraduate program (B.Sc.) in Medical Radiation Physics was established in the Ariel University Center of Samaria. The program was submitted to the Council of Higher Education (MALAG) in 2003 and was finally approved by the Council on October 2005. Registration for the first class was announced in January 2006. Studies started on October 2006. Of 24 candidates who applied, 16 were admitted. 12 of the 16 students completed their study duties in the first year. All of them started their second year studies in October 2007.

RATIONALE
The rational for the establishment of a first faculty for the training of medical radiation physicists at undergraduate level stems from the actual needs of the National Health Service system and of the local high-tech industry.

IMAGING AND RADIOTHERAPY SERVICES IN ISRAEL
The extent and the professional level of radiotherapy, nuclear medicine and diagnostic X ray imaging services in Israel are similar to those in EC states.
In the year 2005 the number of diagnostic radiology procedures performed in Israel reached about 1 per capita (population: ~7x10^6).
In the same year the number of new radiotherapy patients was close to 10,000.

TASKS OF THE MEDICAL PHYSICIST
The medical physicist is engaged (among other tasks) in:
- Radiotherapy treatment planning for individual patients.
- Planning and implementation of QA and QC programs for radiotherapy, diagnostic Radiology and Nuclear Medicine equipment.
- Acceptance testing of imaging and of radiotherapy equipment.
- Precise calibration of dosimetry equipment.
- Optimization of diagnostic radiology and NM imaging procedures.
- Radiation protection program (patient and staff).

THE NEEDS
The 1997 EC medical exposure directive (EC 1997) requires that a qualified expert in radiophysics will be available to sophisticated departments of radiotherapy and medical imaging. The government of Israel and Israel's health authorities are making efforts to upgrade the local health services and bring them to comply with the European Directives requirements. In 1995 the average number of qualified medical physicists employed in the EC countries was estimated to reach 10-15 per million citizens (CEC 1995).
Ten years later, in 2005, the number of qualified medical physicist employed in Israel reached 5-6 per million citizens.
In view of these there is a recognized need to double or even triple the existing qualified medical physics manpower in Israel.
AIMS OF THE PROGRAM
As of 2003 none of the Israeli Universities or Technical Institutes offered a training program in Medical Physics (undergraduate or graduate).
The new program offered by the Ariel University Center tries to fill this lack.

The aim of the program is to produce B.Sc. graduates with basic knowledge of medical radiation physics as applied in Israel.
Graduates of the program are expected to join the professional workforce as junior medical physicists in departments of imaging and radiotherapy in Israeli hospitals and private clinics.

CURRICULUM
Subjects studied include:
- Classical and modern physics.
- Pure and applied mathematics.
- Computer science and application of computers in medical imaging and radiotherapy.
- The application of x rays, nuclear radiation, magnetic fields, ultrasound and RF radiation in medical diagnosis and therapy.
- Anatomy and physiology.
- Biochemistry.
- Radiobiology and radiation damage in the human body.
- Radiation dosimetry.
- Image processing.
- Radio-therapy treatment planning.
- Radiation protection and legislation related to the exposure of the patient, the worker and members of the public to ionizing and non ionizing radiation.
- Quality Assurance and Quality Control requirements and their practical implementation.

REFERENCES

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AC Powered Linear Power Supply Units ("AC Adaptors") - Magnetic Fields and Unexpected Electric Current Dependence

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INTRODUCTION

Household appliances are an important source of indoor 50 Hz extremely low frequency magnetic fields (ELF MF) exposure\textsuperscript{(1,2)}. Furthermore, a number of epidemiological studies have pointed toward a possible association between leukemia risk and personal use of household appliances\textsuperscript{(3)}.

AC powered linear power supply units (LPSU) usually known as "AC adaptors", "power adaptors", "power bricks" or just "chargers" (figure 1) are very extensively used at home and workplace. Such LPSU may be located inside some appliances (e.g. radio cassette players) or as an independent unit sometimes known as "wall wart" or "plug-in adaptor". Another common type of AC adaptor is the so-called "Universal adaptor", which attempts to replace missing or damaged ones, using multiple plugs and selectors for different voltages and polarities. Another type of PSU, named "Switched Mode Power Supply" (SMPS) will not be discussed here.

A simple LPSU usually uses a transformer to convert the voltage from the wall outlet ("mains") to a different, usually a lower voltage. If it is used to produce DC, as is usually done, a rectifier circuit is employed either as a single chip or as an array of diodes called a diode bridge. A capacitor is then used to smooth output voltage ripple. Finally, depending on the requirements of the load, a linear regulator may be used to reduce the ripple, sometimes also allowing for adjustment of the output to the desired but lower voltage.

Three electric currents can be defined within the adaptor:

1. Input AC current (I\textsubscript{LN}) - from the mains (220 V) to the transformer itself.
2. Output AC current (I\textsubscript{OUT}) – from transformer into the rectifier/ regulator.
3. DC load current (I\textsubscript{LOAD}) – from the rectifier to the outer load.

This study examines the ELF MF emitted by standard domestic AC powered LPSU (herein AC adaptors) and their dependence on the electric currents within the adaptor. The possible health effects are discussed elsewhere\textsuperscript{(4)} and will not be dealt with here.
MATERIALS AND METHODS
Two kinds of measurements were conducted within this study:

1. ELF (40-800 Hz) MFs were measured near 3 different domestic AC Adaptors, herein called "Universal", "Phone" and "15 V" (figure 1). For each adaptor, 3 sets of measurements were taken at three perpendicular planes (X Y Z). For each of these 3 sets, the MFs were measured at 5 different distances (Contact, 12, 25, 50 and 100 cm), while at each distance measurements were taken every 10 degrees around the adaptor (at contact - every 90 degrees). This process yields a total of 108 measure points at each distance (36 angles × 3 planes) and nearly 540 points (108 × 5 distances) for each adaptor. All measurements were performed without a load connected to the adaptor (ILOAD=0).

2. In order to analyze the exact source of the MFs and their dependence on the currents within the adaptor, another series of measurements were carried out: The three currents defined above and the MFs at an intermediate distance from the adaptor (i.e. 25 cm, X plane, every 90 degrees) were recorded at different load currents. Two "zero current" measurements were taken - one simply without a load and the other while the transformer’s output was disconnected. These series of measurements were conducted for the "Universal" adaptor only.

All MF measurements were conducted using an EMDEX II MF meter (40-800 Hz).

![Figure 1. The AC Adaptors (LPSU) that were used in this study (from left to right - "15 V", "Universal" and "Phone")](image)

RESULTS
Typical results of MFs around (360°) one of the adaptors ("15 V") in one plane (X) and at different distances are presented in figure 2. The next figure enables us to compare the MF levels in the 3 planes for one adaptor ("Phone") after the results have been averaged over 360°, while the last figure summarizes the results and presents MFs vs. distance for each of the 3 adaptors (averaged over all 3 planes and 36 angels). All results are presented on a logarithmic scale. For technical reasons, the standard deviations are displayed for the intermediate graphs only (figures 3-4). Final results of the second series of measurements can be found in table 1 (MFs were averaged over the 4 angles).
Figure 2: MFs versus angle around the adaptor for five different distances ("15 V" adaptor, X plane)

Figure 3: Average MFs versus distance for three different planes ("Phone" adaptor)

Figure 4: Average MFs versus distance for the three different adaptors
<table>
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<th>$I_{IN}$ (mA, AC)</th>
<th>$I_{OUT}$ (mA, AC)</th>
<th>$I_{LOAD}$ (mA, DC)</th>
<th>AVG MF (μT)</th>
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<tr>
<td>47</td>
<td>298.6</td>
<td>146.7</td>
<td>1.715</td>
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</tbody>
</table>

Table 1: Currents and average MFs for different loads

DISCUSSION AND CONCLUSIONS

MFs in vicinity of LPSU ("AC Adaptors") are very high relative to average typical residential levels and to other electric appliances, and thus might have significant contribution to the total daily exposure. MFs of the order of 10 μT can be found at distances of 5-20 cm from the adaptor, 1 μT at 25-45 cm, and background levels (0.05-0.1 μT) at distances typically exceeding 50-100 cm. MFs in contact with the adaptor may exceed ICNIRP 1998 general public guidelines (100 μT), and reach up to 150-200 μT. The levels are roughly isotropic.

Furthermore, the results indicate that MFs in the vicinity of AC adaptors are not linearly correlated with any of the electric currents within it. To the best of our knowledge this phenomenon was not yet discussed in the literature, at least not in the context of human exposure, although it might be of importance in this aspect. These somewhat unexpected results will be discussed at the conference.

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Magnetic Fields in Vicinity of Stand Alone Transformer Stations

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INTRODUCTION

Since 1979 a large number of epidemiologic studies examining the potential effect of residential exposure to extremely-low frequency magnetic fields (ELF-MF) on childhood cancers have been conducted and published. Two pooled analyses\textsuperscript{(1,2)}, which included major epidemiologic studies of ELF-MF and childhood leukemia, pointed to a consistent, roughly twofold increase in childhood leukemia risk in association with residential ELF-MF exposure above 3-4 mG. Based on “limited” epidemiologic evidence linking ELF-MF exposure to childhood leukemia and “inadequate evidence” for carcinogenicity of ELF-MF in rodent bioassays, the International Agency for Research on Cancer (IARC) classified ELF-MF as a possible human carcinogen (2B classification) in 2002\textsuperscript{(3)}. This classification implies that a potential causal relationship between MF and childhood leukemia is not the only possible explanation for the observed epidemiologic association. Based on available evidence, alternative explanations cannot be excluded; Selection bias remains one of the most frequently mentioned non-causal possible explanations for the MF-childhood leukemia association. However, only very limited and indirect evidence is available in support of or against the selection bias hypothesis. To reduce the scientific uncertainty surrounding the association between MF and childhood leukemia, further scientific evidence is required.

In some countries, including Israel, it is common that electric transformers are placed in the basement or on the first floor of multi-level apartment buildings. MFs in the apartments located immediately above or next to these transformers are assumed to be significantly higher than in other apartments in the same building. A multi-country case-control study of childhood leukemia focusing on populations living in apartment buildings with transformers was launched by the US Electric Power Research Institute (EPRI). This study, named "TransExpo", holds the potential of providing a new and informative test of both high magnetic fields and the selection bias hypothesis. In the currently discussed pilot study, MF levels will be assessed taking into consideration the location of the residence in relation to the nearby transformer and the characteristics of the specific transformer. If MF exposure can be reliably categorized based on the relative location of the transformers and of the apartments in the same building it may be possible to conduct a large-scale study that minimizes the potential for selection bias, and maximizes the selection of highly exposed individuals.

Studies on exposure to MF in such apartments have been performed in Hungary\textsuperscript{(4)} and Finland\textsuperscript{(5)}. Results from Israel, which is one of the participating countries in this project, are reported here for the first time.
MATERIALS AND METHODS
First we characterized the engineering and structural aspects of different types of transformer stations which might be relevant to exposure of nearby inhabitants to MF. Additionally, we conducted MF measurements in the vicinity of three stand-alone transformer stations. All stations had the same major parameters, which are probably the most common parameters for transformer stations in Israel (i.e. 22kV input voltage, 630 kVA nominal power, single transformer per room and upper Low Voltage feed lines). Furthermore, all had similar dimensions (approx. 4×7×3 m³). In each transformer station, 3 types of MF measurements were conducted, usually using the EMDEX II MF meter (40-800 Hz):

1. 1×1 m (or an even more dense) grid around the room, approx. 1 m above ground level.
2. 1×1 m grids on top of the room, each at a different height (from 0 to 4 m above the roof level).
3. In addition to these spot measurements, continuous measurements were performed at 1 or 2 locations during the whole measurement process (i.e. a few hours).

The results, which are briefly reported and discussed here, enable a better understanding of the characteristics of the MF around transformer stations, and will serve as a basis for the dosimetry of the future large scale epidemiological study.

RESULTS
Typical results are presented in figures 1-3. More detailed results will be presented and discussed at the conference.

Figure 1. Magnetic fields in the vicinity of a typical transformer station (1 m above ground)
Figure 2. Contour plots of magnetic fields above a typical transformer station

Figure 3. Magnetic field levels versus time at 2 locations near a typical transformer station
DISCUSSION AND CONCLUSIONS
A few important facts can be concluded from the results. First, MF in the vicinity of all 3 transformer stations showed the same pattern, namely an ellipsoid centered along the Low Voltage (LV) cables feeding the LV switchboard which, as this study shows, are the major source of MF within the transformer stations. High Voltage (HV) cables and switchboard seem to have no or negligible contribution to the overall exposure. Secondly, MFs above the transformer station were shown to be very sensitive to the exact height above the source. Although it is too early to draw final conclusions, it may be preliminarily suggested that

MF around the transformer room can be classified as High (> 10 mG) at distances up to approx. 1-3 m from the LV cables and as Low (< 1 mG) at distances greater than approx. 4-6 m from those cables. Above the room, the situation is more complex: High MFs were found inside a cap-like area (see figure 2) with a radius of approx. 2-4 m (on the roof itself) and height of 0.5-1 m (at the center of the cap). For Low MFs the radius and height were greater than 5 m and 3-4 m, respectively.

The time dependence of the MFs was found to be very complicated and it is yet unclear whether or how it can be modeled or predicted. Generally speaking, a significant rise in the MF levels was observed at noon (approx. 12:00-13:00). Another rise is expected in the evening (approx. 18:00-19:00), but was not measured for technical reasons. However, rapid changes (2-3 fold) were also observed during short periods of time (few minutes). In some cases, a correlation was found between continuous measurements taken at the two different locations (figure 3), but no association was found between those measurements and typical load curves supplied by the IEC.

The authors would like to thank the Israeli Electric Company for providing technical information needed to perform this study. This study is sponsored by the Electric Power Research Institute (EPRI) in Palo Alto, California.

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Laser Hazard Assessment with an Eye Simulator

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INTRODUCTION
Laser radiation entering the eye is focused on the retina and produces there an extremely high energy or power density. This is the essence of laser eye hazards and the various laser safety standards aim mainly at preventing this from happening.

Laser classification is thus based on stating an Accessible Emission Limit (AEL) which ensures that under reasonably foreseeable conditions, the amount of energy/power entering the eye will be too weak to cause damage.

The main consideration is the size of the focal spot on the retina - the smaller this becomes, the higher the power/energy density, and consequently - the higher the hazard.

The smallest focal spot size considered in the leading laser safety standard - the IEC 60825, is 25 μm. Such a small spot can be produced only by a laser beam. This standard considers spot size of 250 μm to be producible by extended light sources with a much reduced hazard.

Determining the retinal focal spot size computationally from beam data is rather complex and involves calculating the AEL depends on various parameters and correction factors, stated in the said standard.

One correction factor is termed C6 (see Tables 10 of the standard ref 1) and represents the beam angular divergence. The angular beam divergence affects the size of the apparent source size in the eye. This is the real or virtual object that forms the smallest retinal image for a given distance. The angular subtense of the apparent source is determined by the smallest retinal image size that the eye can produce by accommodation (i.e., by varying the focal length of the eye lens). As described in IEC 60825-1, D.2 - the effects of laser radiation on biological tissue (ref. 1), the above reference to the retinal spot diameter of 25 μm is equivalent to considering the maximum focal length of the eye, f = 17 mm (for optical eye modeling - see figure 1).

Limited by the aberrations of the refracting surfaces of the eye, the minimum angular subtense, $\alpha_{\text{min}} = 1.5$ mrad. The angular subtense of the apparent source is used as a measure of the retinal image size. This angular subtense is the planar angle subtended by the diameter of the apparent source at the eye lens (for illustration - see Figure 2).
The minimum retinal spot size is given by: $D_{\text{min}} = \frac{fx \alpha_{\text{min}}}{0.0015 \text{ radians}} = 17 \text{ mm} \times 0.0015 \text{ radians} = 25 \mu\text{m}$.

The problems inherent in determining accurately the angular subtense in laser systems in which additional optics were implemented, can be easily solved if actual measurements of the retinal focal spot size (and distribution of laser energy within the spot) could be conducted.

To this end we have designed and constructed an eye simulation device, the design and performance of which are presented in the following.

**THE EXPERIMENTAL SYSTEM**

In order to measure the image size on the retina, different measurement methods can be used. Following are some of the methods:

1. **Circular Aperture Measurement**, using the following equation,

   $$D_L = \sqrt{-\frac{D_{fa}}{ln \left[1 - \frac{\Phi_d}{\Phi_0}\right]}}$$

   Where:
   - $D_L$ = the source spot size
   - $D_{fa}$ = pinhole diameter
   - $\Phi_d$ = beam energy transmitted through $D_{fa}$ the aperture (the transmitted energy should be less than 80% and more than 40 %)
   - $\Phi_0$ = the laser output

2. **Using "Knife Edge" measurements**, where the intensity passing a sharp edge is measured

The method presented here utilizes a beam profiler (OPHIR BeamStar FX 33- 1/3") which incorporates a high resolution CMOS array (pixel size - 7.5 \mu\text{m}) with a linear intensity response, to simulate the retina. The eye optics are simulated by a gradual index 18 mm focal length lens (Gradim \textsuperscript{16}) equipped with a 7 mm aperture and focusing mechanism.
Figure 3, the eye simulator
(OPHIR BeamStar FX 33- 1/3" beam profiler with an
gradual index  F=18 mm lens and a 7 mm entrance aperture)

The great advantage of this experimental approach is that it allows us to observe non-uniformities in the intensity distribution within the retinal image - a feature which might escape detection by the other - above mentioned methods.

Experimental:
First measurement was made using Helium-Neon laser, with low beam divergence. Figure 4 shows the image and beam profile on the CMOS without the 18 mm lens.

Figure 4, Helium Neon at 100 mm distance - bare beam
The exit beam $D_{37}$ that had been measured was 0.456 mm. In comparison, figure 5 represents the "retinal" focal image as obtained with our "eye". It can be seen that the beam was focused to a spot of 23 µm diameter only.

![Image](image_url)

**Figure 5.** Same HeNe laser at 100 mm distance in our "eye"

The extremely small size of the focal spot is apparent in the beam profile, and the fact that the whole power emitted by the laser is concentrated in a very small spot is evident. This result is expectable, but in the next example the eye simulator is indispensable; this is the analysis of diffused laser beams.

IEC 60825-1 (1) uses the term "totally diffusing" (transmitting or reflecting) surface. The basic assumption is that once a laser beam has passed a transparent diffuser (or reflected by a diffusing reflecting surface) the beam becomes divergent and loses its coherence, thus becoming a non-laser beam with a reduced hazard. The conventional measurement methods can measure the increased divergence of the beam after diffusion, and it is then assumed that 100% of the beam energy is so diffused. The presence of a small (a few percent or less) part of the original laser beam which has not been diffused will escape these measurement methods. However, using our "eye" on the original beam path is capable of detecting and assessing the intensity of the non-diffused component. It is important to note that such a small component might pose an eye hazard, which otherwise might not be considered at all. An example is given in figure 7.

Here we examined the above HeNe beam after passing a "perfect diffuser", expecting it to lose its original high directivity and coherence.
The measured spot size was 2843 µm and the resulting angular substance is therefore \( \alpha=2.843/18 \rightarrow \alpha > \alpha_{\text{max}} > 100 \) mrad. This looks like an extended - non laser light source, but a close examination of the intensity distribution reveals clearly that a part of the beam continues to behave as a laser beam (note the prominent peak in the 2D and 3D representations).

To verify that this peak is indeed a coherent and low divergence beam (namely: it behaves as a non-diffused beam, and consequently poses the conventional laser beam hazards) we have measured the power density reduction of the "perfectly diffused" beam as a function of distance, expecting it to drop as \( 1/R^2 \).

In fact, the power entering a 7 mm aperture at 50 cm and at 200 cm was the same, rather than the \((50/200)^2 = 1/16\) reduction expected, clearly indicating that a small part of the beam continued to be coherent and has a very small angular divergence. (measurement with an GIGA HERTZ OPTIK LP 9901 power meter)

**DISCUSSION**

We have designed and constructed an optical eye simulator consisting of a gradual index lens with optical properties similar to those of the human eye, and of a beam profiler as a "retina". This device greatly simplifies the laser hazard assessment for real-world laser beams, especially those implemented in optical systems.

We used this device to examine beams assumed to be non-hazardous due to diffusion, and found non-diffused and consequently - hazardous remnants of the original beam.

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Interference of Extremely Low Frequency (50 Hz) Electric and Magnetic Field Emitted by HV Power Lines with RF Broadband Field Meters

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INTRODUCTION

The increased use of radio frequency radiation (RFR) emitting technologies has raised the public concern of possible hazardous health effects caused by exposure to RF fields. These concerns increased the demand for intensive RFR environmental and occupational measurements. Most RFR measurements are conducted with commercial RF broadband electric field meters (RFFM). In many cases, these measurements are performed in the vicinity of high voltage (HV) power lines. Although RFFM are expected to be immune to the Extremely Low Frequency (ELF) electric and magnetic fields (EMF) emitted by HV power lines, in recent measurements near such lines we have noticed some distortions in the reading of the RFFM.

The aim of the present work was to examine possible disturbances to the proper functioning of several commercial RFFM in the vicinity of HV power lines, apparently as a result of strong ELF EMF. In addition, the influence of surveyor body on the RFFM readings was assessed.

MATERIALS AND METHODS

The experiment was conducted at two open sites located near HV (161 kV) power lines. In each site, approx. 15 measuring points were defined. These reference points were located along a straight line, approximately perpendicular to the power lines, and 5 m apart from the each other. In each point, ELF electric and magnetic fields were measured first, using PMM 8053A connected via fiber optic to EHP-50B distant probe (5 Hz – 100 kHz) placed on non-conductive tripod. The frequency span was set to 5 - 100 Hz (no significant difference was found using wider span of 6 - 500 Hz). Immediately afterwards, six commercial RFFM were placed one by one in each reference point, and their readings were recorded. The RFFMs were manually held, as done in daily RF surveys. In both sites, RF background electric field levels were measured away from the power lines and found to be below the detection level of most RFFM used. The RFFMs used were:

1. PMM 8053A meter with two distinct probes:
   a. EP-408 (1 MHz – 40 GHz, herein pmm40G)
   b. EP-330 (100 kHz – 3 GHz, herein pmm3G)
2. Wandel & Goldermann EMR-300 meter with two distinct probes:
   a. E-8 (100 kHz – 3 GHz, herein wg3G)
   b. E-9 (10 MHz – 18 GHz, herein wg18G)
3. Holaday HI-2200 meter with E-100 probe (100 kHz – 5 GHz, herein Holaday).
4. EMC RFP-04HF meter with PI-03 probe (3 MHz – 18 GHz, herein Smart) It should be emphasized that this specific RFFM was designed for fast operation and hence declared by the producer as more vulnerable to ELF EMFs.
RESULTS
Results of the measurements in both sites are presented in figures 1-2. Figure 1 indicates that two RFFMs ("Holaday" and "Smart") suffered serious interference when electric and magnetic field exceeded approx. 400 V/m and 0.8 μT respectively. At the second site, where ELF EMFs were higher (up to nearly 4 kV/m and 6 μT), significant disruptions were observed for the wg3G, and minor interference for the wg18G. Moreover, at these high fields, the readings of the Holaday became inordinate. Both PMM probes showed only minor interference under all conditions.

Fig 1: RFFMs readings vs. ELF electric magnetic field in the first site

Fig 2: RFFMs readings vs. ELF electric and magnetic field in the second site
However, these measurements could not indicate whether the source of the said interference is the electric field or the magnetic field. To resolve this issue, another set of measurements was conducted inside a vehicle located beneath a HV power line. The electric field within the vehicle was significantly attenuated (26 V/m) while the magnetic field was actually unchanged (1.65 µT). Nearly no interference was observed under these conditions, leading to the conclusion that the electric fields are mainly or exclusively responsible for this undesired interference.

Figure 3 shows the RFFMs readings as a function of the distance from the surveyor body/hand to the meter (at 3650 V/m and 4.85 µT ELF EMFs). It can be seen that interference becomes significant only at distances less than approx. 0.5 m.

Fig 3: RFFMs readings vs. surveyor body's distance from the meter
CONCLUSIONS
All six tested commercial RF broadband electric field meters (RFFM) were somewhat interfered by ELF *electric* fields at the vicinity of HV power lines. Moreover, two of them were very seriously interfered by relatively low electric fields (approx. 400 V/m). Another one was intermediately interfered at higher ELF electric field levels (above approx. 1.5 kV/m) and the two others were quite immune up to very high field (approx. 5 kV/m). It appears that these interferences are a result of the *electric* ELF field, and not the magnetic field. In addition, it seems that this interference is caused primarily by the presence of human body/ hand, as well as other conducting objects, in close proximity (less then 0.5 m) to the RFFM.

To summarize, attention should be taken while conducting RFR measurements at high ELF electric field environments, e.g., near HV power lines. In the absence of reliable information related to the susceptibility of the specific RFFM model being used to ELF EMFs, it is recommended that the probe should be placed on a non conductive tripod, at least 0.5 m away from any conducting object.

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Acoustic Waves Generation by Pulsed Microwave Radiation

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INTRODUCTION
The ability of humans and mammals to sense appropriately modulated pulses of microwave radiation (2.4 MHz – 10 GHz) is known for some decades. The common explanation of this phenomenon is the thermoelastic expansion theory: The electromagnetic energy, which is absorbed by the soft tissues, produces a small but rapid rise of the temperature (in order of 10⁻⁵ °K). This heating causes an elastic expansion of the material that elicits acoustic waves in the head, which are heard via bone conduction.

In the past, several analytical models were presented. In this study those models were examined and some shortcomings were found. In addition a generalized model was developed and its results will be presented in the lecture.

THEORETICAL FORMULATION
Let us assume a plane wave of pulsed microwave energy impinges on a homogeneous spherical model of the head. The absorbed microwave power density at any point is given by:

\[ I(\vec{r}, t) = \frac{1}{2} \sigma |\vec{E}|^2 f(t) \]  \hspace{1cm} (1)

where \( \sigma \) is the electrical conductivity of the tissue, \( f(t) \) is the shape of the pulse in time, and the electric field \( \vec{E} \) is given by:

\[ \vec{E} = E_0 \exp(-i\omega t) \sum_{j=1}^{\infty} \frac{2j+1}{j(j+1)} (a_j \tilde{M}_{ej} - ib_j \tilde{N}_{ej}) \]  \hspace{1cm} (2)

where \( E_0 \) is the incident electric field strength, \( \omega \) is the angular frequency of the microwave radiation, \( a_j \) and \( b_j \) are the magnetic and electric oscillations, respectively, and \( \tilde{M}_{ej} \) and \( \tilde{N}_{ej} \) are vector spherical wave functions.

The previous analytical models have approximated the absorption pattern for the frequency of 918 MHz to possess radial symmetry and to have the form:

\[ I(r, t) = I_0 \left[ \eta + \frac{\sin(N\pi r/a)}{N\pi r/a} (1-\eta) \right] f(t) \]  \hspace{1cm} (3)

where \( a \) is the radius of the sphere and is equal to 7 cm, \( N \) is a positive integer which specifies the number of spatial oscillations of the absorbed energy approximation, and \( \eta \) is a real number expressing the degree of uniform heating of the sphere (0≤\( \eta \)≤1). Lin assumed in his models that \( N=6 \) and \( \eta=0.6 \), whereas Shibata et al. used the values: \( N=4 \) and \( \eta=0.4 \). In Lin's approximation the total absorbed energy is negative. This fact leads to unphysical results.

Those models were implemented only for this frequency and a rectangular pulse. In the present study a generalized model for any shape of pulse and for any pattern of absorption with radial symmetry (in its Taylor's series form) was developed:

\[ I(\vec{r}, t) = \sum_n I_n \left( \frac{t}{a} \right)^n f(t) \]  \hspace{1cm} (4)
This model was applied to some shapes of pulse (ramp and half sinusoidal wave) with a carrier frequency of 2.45 GHz impinging on a spherical head with a radius of 10 cm. The outstanding difference between the two frequencies is the place at which the absorption is centered. For 918 MHz it is at the center of the sphere while for 2.45 GHz it is at its surface.

RESULTS
According to the analytical models the perceived frequency of sound depends on the physical and geometrical properties of one's head and does not depend on the energy absorption pattern. The calculated frequency of the 1\(^{st}\) harmony for adult human (a=7cm) is 10.4 kHz for a free surface and 14.9 kHz for constrained one.

The amplitudes of the first 20 harmonies can be seen in figure 1. The model of Shibata et al.\(^{(4)}\) predicts the highest intensity. The calculation of Lin for a free surface yields negligible strength for all harmonics except for the 6\(^{th}\) harmony (above human's hearing threshold). For a constrained surface, this model predicts that the 6\(^{th}\) harmony has the largest amplitude.

![Figure 1. The sound pressure amplitudes according to the previous models\(^{(2-4)}\) in the center of 7 cm radius spherical head for the first 20 harmonies. The pulse width is 20 µsec and the peak absorbed power is 1 W/cm\(^3\). Harmony 0 is the change of the base level.](image-url)
Figure 2. A comparison between previous models\(^{(3,4)}\) results (1\(^{st}\) harmony) and experimental data of perceived loudness\(^{(5,6)}\). Acoustic pressure amplitude generated in a 7-cm-radius spherical head, which is exposed to microwave pulse, as a function of pulse width (a - constant peak power, b - constant peak energy).
The results of the previous models corresponded well to the experimental data as can be seen in figure 2. The dependence of the acoustic loudness was compared to experimental results. At a study conducted by Tyazhelov's et al.\(^{(5)}\) 18 examinees were exposed to 800 MHz microwave pulses with PRF (Pulse Repetition Frequency) of 2 kHz. Frey and Messenger's\(^{(6)}\) exposed the heads of 4 volunteers to 1.245 GHz microwave radiation. This match and other outcomes strengthen the thermoelastic expansion theory as the leading explanation of the effect.

**CONCLUSIONS**

The thermoelastic expansion theory may explain the microwave hearing effect. However, some unphysical features of previous models\(^{(2,3)}\) are demonstrated. The model of Shibata et al.\(^{(4)}\) is found to be free of these deficiencies. Some numerical results of our generalized model, applicable to arbitrary absorption patterns of radial symmetry, will be presented.

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Source Term Estimation for a Radiological Dispersive Device Scenario

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ABSTRACT
As a part of developing a dispersion model for a Radiological Dispersive Device (RDD) scenario, it is important to understand the evolution of the radioactive cloud at the first stage after its formation as a function of time, amount of TNT and atmospheric stability. Cloud rise models were developed in the past for large quantities of explosives, in the range of hundreds of kilograms of TNT. Therefore, a gap in knowledge exist for TNT amount less than 100 kg, which is relevant to an RDD event. In this work, the results two sets of explosions (GFI and GF2), made using 0.25-100 kg of TNT are given. A new semi-empirical cloud rise model, based on the results of these two experiments, is then described.

Key Words: Cloud rise model, RDD, risk assessment

1. INTRODUCTION
In the first few hours after the explosion of a Radiological Dispersive Device (RDD), it is not likely to assume that reliable field measurement will be available. Therefore, the only way of getting estimation for ground contamination distribution and air particles concentration in this period is by using theoretical models\[1,2,3\]. One of the necessary data needed for getting reliable results from a dispersion model is a precise characterization of the source term. This can be done by getting a full description of the cloud top and particle distribution as a function of the type and amount of explosive used\[4\], the geometry, type and activity of the radiation source and on the atmospheric conditions in the close vicinity of the detonation point. In this report, the Green Field (GF) task plan conducted in order to measure the source term parameters as a function of time, explosive amount and atmospheric stability is describe and the semi-empirical cloud rise model written according to these results is presented.

2. EXPERIMENTAL
The GF task plan performed so far included two sets of explosions conducted in varies atmospheric conditions. In the first set (GFI) 0.25-2 kg of TNT were detonated. In the second set (GF2) 2-100 kg of TNT were detonated. The clouds created by the explosions were recorded using an array of several visible light video cameras, one thermal infrared camera (used only in GFI) and a fast camera (used only in GF2). The cameras were located in different angles and distances around the detonation point. The meteorological array included a meteorological mast 6 m in height with temperature, relative humidity, sun radiation and wind velocity measurements. The location of the mast was determined to be the closest possible to the detonation point. A radiosonde was also launched from the detonation point before the beginning of each series in GFI and before each explosion in GF2. The radiosonde readings of vertical atmospheric profiles (temperature, relative humidity, wind velocity and localization) were recorded in a ground station with a measurements frequency of 1 Hz.
3. RESULTS

The time dependence of the cloud top measured in the explosions conducted during the GF1 and GF2 tests is depicted in figures 1 in a logarithmic (log-log) scale. Based on the linear dependency shown in these results, it is possible to assume that the time dependent of the cloud’s top height, for a given amount of explosive, apply a simple power law,

\[ h(t) = h_{t=1} \cdot t^a \]  

(1)

\( h(t) \) – cloud top height, measured in meters.

\( h_{t=1} \) – cloud to height in meters, a second after the explosion.

\( t \) – time elapsed after the explosion, measured in seconds.

**Figure 1.** The time dependence of the cloud top as measured in the GF tests, for stable (full marks) and unstable (hollow marks) atmospheric conditions, in a logarithmic scale.

The calculated values of the constants \( h_{t=1} \) and \( a \), as resulted from the fit of equation 1 to each of the GF measured results given in figure 1, are depicted in figure 2 and 3 respectively.
Figure 2. The cloud height after one second, $h_{t=1}$, as a function of explosive amount.

Figure 3. The power coefficient, $a$, as a function of explosive amount, calculated for stable (full marks) and unstable (hollow marks) atmospheric conditions.
4. SEMI-EMPIRICAL CLOUD RISE MODEL

Based on the fit results for, $h_{t-1}$, and, $a$, depicted in figures 2 and 3, a semi-empirical model of the cloud top as a function of time, explosive amount and atmospheric stability was obtained. The constant, $h_{t-1}$, (altitude of the cloud after a second in equation 1) is depicted in figure 2 as a function of explosive amount. From the linear distribution of the points in the logarithmic presentation, it is obvious that a linear fit should be used in this case. The relation between, $h_{t-1}$, and the amount of the explosive, $w$, can therefore be expressed using exponential function of the form,

$$h_{t-1}(w) = 6.06 \pm 0.08 \cdot w^{0.265 \pm 0.016}$$

(2)

The constant, $a$, (time exponent in equation 1) as resulted from fitting of the GF experiment results, is plotted in figure 3 as a function of explosive amount, $w$, and atmospheric stability. The linear lines shown in this log-log representation are the result of a semi-empirical model assuming an exponential dependency between, $a$, and $w$,

$$a(w) = a_0 \cdot w^d = \begin{cases} (0.507 \pm 0.028) \cdot w^{0.0318 \pm 0.0015} & \text{Stable} \\ (0.471 \pm 0.012) \cdot w^{0.0174 \pm 0.0071} & \text{Unstable} \end{cases}$$

(3)

Insert the expressions given for, $h_{t-1}$, and for, $a$, in equations 2 and 3 respectively, into equation 1, will result in the semi-empirical model, given in equation 4. This model describes the cloud top as a function of time, until the time, $t_m(w)$ when the cloud reaches its effective altitude, amount of TNT, up to 100 kg, and atmospheric conditions.

$$h(w,t) = h_{t-1} t^a = 6.06 \cdot w^{0.265} \begin{cases} t^{0.507 \cdot w^{0.0318}} & \text{Stable} \\ t^{0.471 \cdot w^{0.0174}} & \text{Unstable} \end{cases}$$

$$0.25 \leq w \leq 100$$

$$t_m(w) = 21 \cdot w^{0.455}$$

(4)
5. SUMMARY
In this work, the results of explosion tests performed using 0.25-100 kg of TNT, done in order to measure the vertical dimension of a cloud formed in a RDD explosion, are presented. Based on these results, a semi-empirical model for cloud rise as a function of time, amount of TNT and atmospheric stability is formulated.

REFERENCES

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Health Risks from Exposure to Low Doses of Ionizing Radiation according to BEIR VII

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The purpose of this paper is to review the main findings and conclusions of the recently published BEIR VII report.

BEIR (Biological Effects of Ionizing Radiation) is the Committee appointed by the U.S. National Research Council of the National Academies to assess health risks from exposure to low levels of ionizing radiation. The BEIR VII report, which was published in 2006, is the seventh in the series of reports published by the BEIR Committee. It focuses on the health effects of low levels of low linear energy transfer (LET) ionizing radiation. Its main purpose is to update the 1990 BEIR V report by reviewing the new information available from epidemiologic and experimental research.

The conclusions of the BEIR V report served as a primary reference for the 1990 Recommendations of the ICRP (International Commission on Radiological Protection) and the 1996 International BSS (Basic Safety Standards), which are the basis for radiation protection regulations in most countries. The BEIR VII report was widely cited in the recently published 2007 Recommendations of the ICRP and it is to be expected to serve as an authoritative reference in the updating of the International BSS as well.

The LSS (Life Span Study) cohort of survivors of the atomic bombings in Hiroshima and Nagasaki played a central role in developing the Committee's recommended risk estimates of cancer incidence and mortality, as in past risk assessments. Data from epidemiological studies related to medical and occupational exposure were also evaluated.

The main findings and conclusions of the BEIR VII report are as follows:

- The current scientific evidence is consistent with the hypothesis that there is a linear, no-threshold dose-response relationship between exposure to ionizing radiation and the additional risk of cancer development in humans.
- Current knowledge on the cellular/molecular mechanisms of radiation-related tumorigenesis tends to support the application of models that incorporate the excess relative risk projection over time.
- The risk of incidence of solid cancers increases in a linear fashion with increasing dose at low doses (in the range 0-100 mSv). At medium to high doses (in the range 100-2,000 mSv) the risk increases with dose according to a linear-quadratic model.
- The risk of leukemia incidence increases with increasing dose in a linear-quadratic fashion, even at low doses.
- A combined analysis of A-bomb epidemiological information and experimental data has been used to provide an estimate of the dose and dose-rate effectiveness factor (DDREF), which is the ratio of the risk per unit of dose at "high dose" to the risk per unit of dose at "low dose". A reasonable range of DDREF values has been found to be 1.1-2.3 and a median value of 1.5 has been used to estimate solid cancer risks.
The cancer risk has been found to decrease with increasing age at exposure.

The cancer risk has been found to decrease with increasing attained age and with increasing time since exposure.

The cancer risk is different for males and for females, and the gender dependency varies for the different cancer sites.

On average, assuming a sex and age distribution similar to that of the entire US population, the BEIR VII lifetime risk model predicts that approximately 1 (one) individual in 100 exposed persons would be expected to develop cancer (solid cancer or leukemia) following a dose of 0.1 Sv (for comparison approximately 42 of the 100 individuals would be expected to develop solid cancer or leukemia due to causes unrelated to radiation). Overall, the cancer mortality risk is approximately half of the incidence risk.

Although adverse hereditary health effects have not been found in children of exposed parents, there are extensive data on radiation-induced transmissible mutations in mice and other organisms. There is therefore no reason to believe that humans would be immune to this sort of harm.

Non-cancer health effects (such as heart disease and stroke) occur at high radiation doses, but additional data must be gathered before an assessment of any possible dose response can be made between low doses of radiation and non-cancer health effects.

The Committee emphasizes the potential risks involved in the growing use of medical radiation in general and of whole body scanning by computed tomography (CT) as a way of screening for early signs of disease among asymptomatic adults in particular. The Committee recommends that follow-up studies should be established of cohorts of persons undergoing CT scans, especially children. In addition, it recommends studies of infants who experience diagnostic radiation exposure related to cardiac catheterization and of premature infants who are monitored with repeated x rays for pulmonary development.

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Human Cytoplasmic and Nuclear Non-Exchangeable Organic Bound Tritium (OBT) Increase Due to Exposure to Different Extracellular Media and Potassium Concentrations

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INTRODUCTION

Tritium is found in the environment mostly as tritiated water, in gaseous or liquid form (HTO, T₂O). Tritium atoms can replace the hydrogen atoms in organic compounds, forming the Organic Bound Tritium (OBT). If this process occurs in the human body, considerable portions of the OBT may bind to biological molecules or remain as OBT with various degrees of exchangeability. Since C-H bonds are stable, except in the presence of strong acids, strong bases, or catalysts, this fraction is called "non-exchangeable OBT". OBT can be retained in the human body much longer than tritiated water (HTO) and therefore the dose arising from OBT can reach 50% of the total tritium dose. In human beings, the effective half-life of HTO is 10 days. OBT, has a higher retention time in the body (theoretically, up to a biological half-life of 500 days), and therefore the doses to the body are higher when compared to HTO.

ICRP publication 78, the commonly used publication on tritium exposure estimation, states that when HTO enters the body, 97% of the tritium systemically absorbed enters the 10 days compartment and the remaining 3% enters the 40 days compartment (i.e., their t₁/₂ is 40 days). It was mentioned (¹) that the total amount of OBT molecules formed after acute exposure to HTO, was 6% (double as the model's value) of the total activity. Relatively recent data on chronic exposure of rats to HTO yielded an unexpected OBT percentage of 22% of the total activity (²). Other studies (³) indicate also tritium accumulation.

The purpose of the present work was to investigate the influence of various environmental and physiological conditions on OBT concentrations and characteristics (exchangeable versus non-exchangeable OBT) formed in human living cells, due to tritiated water (HTO) exposure. We exposed malignant osteoblast MG-63 cells to isotonic and hypotonic solutions supplemented with or without different concentrations of KCl, known to have a water swelling potential effect on cells.

RESULTS AND DISCUSSION

⁴⁰K is known as one of the main contributors of human internal radiation exposure. It is a natural radioisotope contaminant with an abundance of 0.0118%. It emits beta radiation at a much higher energy than tritium, but as the beta spectra are continuous, the interaction in the scintillator produces also pulses in the low region, characteristic of tritium. Potassium. In order to check the potassium effect on OBT formation in cells, we determined first the ⁴⁰K contribution ("equivalent ³H activity") to the ³H quantitative determination. Samples containing KCl in various concentrations, tritium free water, and liquid scintillation were prepared and counted. It was found that 4.27±0.98% from the spectrum area is contained in the ³H spectral region.
Physiological-relevant $^{40}\text{K}$ concentrations were prepared by using a 1M KCl (analytical grade, Frutarom, Israel) solution in samples of 20 mL final volumes. Each $^{40}\text{K}$ standard concentration contained a constant amounts of water (verified before and practically containing no tritium atoms), liquid scintillator (to achieve a constant quench parameter (SQPE) and counting efficiency), and $^{40}\text{K}$ contribution due to its total amount in the constant ingredients. The minimal detection activity (MDA) of $^{40}\text{K}$ calculated from the integral area of the energy region of $^{40}\text{K}$ was 3.04E-05 Bq/sample for a 95% confidence level, for 45 min of counting time. Figure 1 describes the $^3\text{H}$ equivalent activity of $^{40}\text{K}$ as a result of the various concentrations of KCl.

![Graph showing the relationship between KCl concentration and activity](image)

Figure 1. $^{40}\text{K}$ activities int the $^3\text{H}$ spectral region (0-18.6 keV) for different KCl concentrations.

Because of its contribution in the $^3\text{H}$ region, $^{40}\text{K}$ is a disturbing factor when measuring $^3\text{H}$ activity. The subtraction of the $^{40}\text{K}$ contribution was checked by adding an HTO spike of 0.72±0.1 Bq to each KCl concentration and subtracting the $^{40}\text{K}$ part from the total activity of the $^3\text{H}$ region. All $^3\text{H}$ determinations were within a range of ±2σ.

Table 1: OBT content in cell components grown in an environment which contained 200 Bq/ml HTO.

<table>
<thead>
<tr>
<th>Fraction</th>
<th>Activity (Bq)</th>
<th>% of Total Activity</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cytoplasmic fraction*</td>
<td>6.1±0.6</td>
<td>48.8±4.9</td>
</tr>
<tr>
<td>Nuclear fraction*</td>
<td>0.7±0.1</td>
<td>5.6±0.6</td>
</tr>
<tr>
<td>Lipids and membrane fraction**</td>
<td>0.4±0.0§</td>
<td>3.2±0.3</td>
</tr>
<tr>
<td>DNA, RNA fraction**</td>
<td>5.3±0.5</td>
<td>42.4±4.2</td>
</tr>
</tbody>
</table>

* The fraction is represented by 1500μg proteins.
** The fraction was normalized to 1500μg proteins.
§ The uncertainty is less or equal to 0.05Bq.
The OBT activity in components of cells grown in isotonic medium containing 200 Bq/ml HTO were investigated, and the results are shown in table 1. OBT activity determinations for the MG-63 malignant cells treated with various concentrations of KCl and a constant activity of HTO (200 Bq/ml), were carried out by measuring a quantity of 1500μg protein samples. $^{40}$K contributions to the $^3$H region were subtracted as described before. The OBT part was calculated by subtracting HTO activity from the total $^3$H activity (5). Figure 2 presents the OBT and the HTO activities of the samples relative to the control sample. The measurements were done after the three days of the cells incubation, in the different KCl and NaCl concentrations solutions. NaCl was supplemented to the isotonic medium in order to simulate extracellular osmolarity increase (as was caused by supplementing high potassium concentrations), and to emphasize the potassium effect. As can be seen in figure 3, the combination of exposing the cells to hypotonic medium (80 mM NaCl) with raising KCl molarity, results in an enhanced bound of cytoplasmic and nuclear proteins to tritium (OBT), by up to $37\pm5.6\%$ and by up to $20\pm5.6\%$ in the DNA and the membrane (OBT) fraction (united), relative to the isotonic control. Thus, increasing the KCl concentrations was found to increase HTO content in cells in any medium (isotonic or hypotonic) by 80-136%.

Figure 2: Cells OBT and the HTO activities formed and accumulated, as a result of exposure to different KCl concentrations and to 200Bq/ml of HTO.

- I1- Isotonic medium (5 mM KCl, 140 mM NaCl) – control
- I2- Isotonic medium (control medium, supplemented to 155 mM NaCl)
- I3- Isotonic medium (supplemented to 20.76 mM KCl)
- H1- Hypotonic medium (3.05 mM KCl)
- H2- Hypotonic medium (supplemented to 9.2 mM KCl)
- H3- Hypotonic medium (supplemented to 18.42 mM KCl)
- H4- Hypotonic medium (supplemented to 27.6 mM KCl)
- H5- Hypotonic medium (supplemented to 36.85 mM KCl)
Table 2 shows the results of total OBT (cytoplasmic and nuclear proteins) formed by the various KCl concentrations treatments to hydrogen exchange, revealed as shown in table 2 that the non exchangeable OBT which was produced, was influenced from extracellular changes. The non exchangeable OBT fraction reached high values, of more than 50% as the isotonic control, and this increase occurred due to the hypotonic medium (a 39% effect), and due to the KCl supplementation (-15% in addition to the hypotonic effect). KCl supplementation effect to the isotonic medium growing cells raised the non exchangeable OBT fraction concentration by -30%.

Table 2: The non exchangeable OBT value (ROBT) for various media (see for legend Fig.2) relative to the control (I1) after exposure to 200 Bq/ml of HTO and to different extracellular media. The uncertainties for the given values are about 5%.

<table>
<thead>
<tr>
<th>Medium</th>
<th>I1</th>
<th>I2</th>
<th>I3</th>
<th>H1</th>
<th>H2</th>
<th>H3</th>
<th>H4</th>
<th>H5</th>
</tr>
</thead>
<tbody>
<tr>
<td>ROBT</td>
<td>1.00</td>
<td>0.83</td>
<td>1.30</td>
<td>1.39</td>
<td>1.52</td>
<td>1.52</td>
<td>1.52</td>
<td>1.54</td>
</tr>
</tbody>
</table>

CONCLUSIONS

The intracellular OBT and HTO activities were found to be influenced by the osmolarity of the extracellular area (isotonic and hypotonic mediums). The majority of the intracellular activity (>90%) is found at the OBT fraction. Raising H2O and HTO volumes inside the cells resulted in intracellular total OBT, and non exchangeable OBT increase. Supplementation of ionic concentrations such as K+ caused the non exchangeable OBT values to be increased by 15% (as we found), and therefore exposed the cells to higher amounts (and perhaps to longer periods) of tritium.

REFERENCES


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High Dose Radiation Facility at Rotem Industries Site

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INTRODUCTION
A high dose irradiation facility is located at the Rotem Industries site, near Oron junction. Its function is to check and calibrate mobile radiation monitoring instrument. The facility is equipped with a sealed radioactive source of 120 Ci \textsuperscript{137}Cs in a standard shielding arrangement with special safety features (massive shielding and switches in the radiation facility door). The mobile monitoring radiation instruments to be checked are placed in a cell inside the shield. After the cell door is closed, the high radiation dose facility can be activated.

Figure 1. The scheme of the irradiation facility.

RISK ANALYSIS

\textit{Standard operation}

The main risk is external radiation to the workers present near the facility. A radiation field mapping was carried out and the results are given below:

- at contact with the right wall of the facility: 1 mr/hr.
- at the watching window: 4.6 mr/hr.
- near the computer location: 4.6 mr/hr.
- at the side hole (without a mobile radiation instrument in the cell and without the shield of the hole): 90 mr/hr.
- at the side hole (with a mobile radiation instrument in the cell and without the shield of the hole): 37 mr/hr.
- at the side hole (with a mobile radiation instrument in the cell and with the hole shielded): 0.2 mr/hr.
### Accidents

<table>
<thead>
<tr>
<th>SCENARIO</th>
<th>HAZARD ANALYSIS</th>
<th>PREVENTION</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Opening the radiation cell while the facility is in operation.</td>
<td>Radiation dose in the cell is $1 \times 10^4$ r/hr.</td>
<td>a. opening the cell only when the worker is behind the door. &lt;br&gt;b. closing the door immediately.</td>
</tr>
<tr>
<td>2. Fire</td>
<td>Fire that cause a dispersion of the radioactive source. The committed effective dose will be 120 mrem (100 meter from the source).</td>
<td>a. working according to the safety procedures. &lt;br&gt;b. prevention of fire in system. &lt;br&gt;c. evacuate people from the area. &lt;br&gt;d. consulting a professional laboratory.</td>
</tr>
<tr>
<td>3. Lost/stolen radiation source.</td>
<td>In case that the source is outside the shield, the radiation field due to the bare source is 396 r/hr.</td>
<td>a. guarding procedures. &lt;br&gt;b. alarm system. &lt;br&gt;c. report immediately to the police and to a professional radiation laboratory.</td>
</tr>
</tbody>
</table>

### CONCLUSION AND RECOMMENDATIONS

**Routine operation**

A. Limitation of watching through the window until shielding is placed.

B. The computer has been moved 1.5 meter further from the facility. The dose rate decreased from 4.6 mr/hr to 0.2 mr/hr.

C. Placing of warning signs on the wall of the facility and prohibition of movement in this area until the shielding is placed.

D. Operation of the facility with the side hole open only in special cases and in the presence of safety supervision team.

**Accidents**

A. The cell door must be locked. Its opening can be done only by passing all system interlocks.

B. All flammable materials must be removed from the room. Fire alarm is connected to the guard post. A fire extinguisher is present in the room.

C. Lost/stolen radiation source: installation of a security door and alarm system.
SUMMARY
A high dose radiation facility for checking and calibrating mobile radiation instrument was installed at the Rotem industries site.
A hazard analysis was prepared.
The accident scenario analysis shows that a probability of a failure is very low.

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Boiling in MTR Fuel Element Channels as Cooling Mechanism During Partial Loss-of-Coolant Accident (LOCA) in Pool Type Reactors

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Introduction

Pool research reactors are usually safe when an event like Loss of Coolant Accident (LOCA) or Loss of Flow Accident (LOFA) takes place, due to the large reservoir of water above the reactor core. Some of these reactors use plate-type fuel elements (MTR) because of the large heat transfer surface of the fuel element.

In partial LOCA, as long as the bottom of the fuel is immersed in water, boiling at the bottom part of the fuel elements drifts the water between the plates to a level which is considerable higher than the "nominal" water level. It appears that that mechanism is important to ensure core integrity during the accident.

The case of LOCA in pool type reactors was studied in few works in the past. Specific models and experiments for MTR fuel elements are presented in the works of Dreier and Winkler [1,2], who tried to predict the transient temperature excursions of the fuel plates during a partial uncovering of the MTR fuel elements, caused by LOCA. Another experimental work was presented in the UCRL Safety Analysis Report (SAR) [3] for the Livermore Pool Type Reactor, where the authors present the results for various water levels between 51mm to 130mm. The channel power in all experiments of that work was above 2 kW. They show that they get the melting temperature of Aluminum at a channel power of 2.5kW to 5kW for the water levels of 50mm to 102mm, respectively. At the water level of 130mm, the maximum temperature was lower than 330°C for up to channel power of 5kW.

Aharon and Hochbaum [4] describe experiments performed in a set-up which simulated one channel in the MTR fuel assembly. It was found that for a high channel power, boiling at the bottom of the channel was very intensive and water was drifted above its nominal level by vapor bubbles. For a lower channel power, bubble formation at the bottom of the channel was less intensive and the top of the channel remained dry. In that case, the temperature at the top of the channel began to increase above the saturation temperature and reached levels of about 230°C, at the water level of 300 mm which was the maximum level in that work.

The main objective of the present work is to measure the effective wetted zone above the nominal water level in the fuel element channels in various channel power and nominal water levels. These results serve as the basis for a simple model for calculating the maximum temperature in the real case of an MTR fuel element.

Experimental set-up and procedure

A schematic picture of the experimental apparatus is shown in Fig. 1(a), with more details provided in Fig. 2(b). The apparatus consists of a tank (3) filled with water to the required level, and the test section formed by six electrically heated 2mm thick plates (4), made of stainless steel 304. The plates are 60mm wide and 600mm high, and the gap (1) between them is 2mm. The six plates are fed in parallel by a regulated direct electrical current, in order to achieve the required power, equivalent to the residual heat in the core after shutdown. The back sides of the two peripheral heated plates were insulated by Pertinax boards (2).
Sixteen thermocouples (7) were inserted into the test section heated walls, at various locations along the heated plates. The thermocouples were soldered inside holes drilled in parallel to the plate surface, as shown in Fig. 2, in order to have an accurate measurement of the wall temperature. Two thermocouples were located at the inlet and exit of the central channel, to measure the water temperature at the inlet of the channel, i.e. temperature of the water in the tank, and the outlet temperature of the water/steam at the exit, respectively.

In addition to these fixed thermocouples, a mobile thermocouple was used to measure the height of the wetting front due to boiling at the bottom. That location was defined as the point at which the temperature measured by the mobile thermocouple had a constant value of 100°C, corresponding to the saturation temperature, without any noticeable fluctuation for a period of 60 seconds. It was inserted from the top of the central channel.

![Diagram](image)

Figure 1: The experimental system: (a) front view, (b) top view and TC details

**Results and analysis**

Typical results of the temperature measurement along the channel walls are presented in Figure 2 versus time, at a water level of 300 mm. The power supplied to the channel is presented in the figure as a solid line. It was observed at a power level of about 800W that boiling at the bottom of the channel was very intensive and created slug flow. The water was drifted up to the top of the channel and created a high heat transfer coefficient at the uncovered zone low wall temperature. that channel power, the maximum temperature along the channel wall was the saturation temperature 100 °C cooling mechanism. When the power level decreased, the boiling intensity decreased and the water was drifted lower. Results of the wetted zone length measurement versus the channel power, conducted by the mobile thermocouple, are presented in Figure 3. The wetted length is, for various water levels.
Figure 2: Typical results of the temperature measurement along the channel walls.

Figure 3: Wetted zone length measurement versus the channel power.

The length of the wetted zone above the water level is a parameter which can be used for the calculation of an MTR fuel element in a nuclear reactor in case of a partial Loss of Coolant Accident (Partial LOCA). One-dimensional conduction model with internal heat generation is formulated at the dry zone to get the maximum fuel temperature:

\[ \frac{d^2T}{dx^2} + \frac{\dot{q}}{k} = 0 \]  

(1)

Where the boundary conditions are:

\[ \frac{dT}{dx} = 0 \quad @ \text{Top of the fuel element} \]

\[ T = T_{sat} \quad @ \text{Top of the wetted zone}. \]
The maximum temperature was calculated at the top of the fuel element for various nominal water levels and channel power. The results are presented in Table 1.

Table 1: Calculated maximum fuel temperature.

<table>
<thead>
<tr>
<th>Nominal Water level [mm]</th>
<th>Max. Calculated Temperature [°C]</th>
</tr>
</thead>
<tbody>
<tr>
<td>200</td>
<td>1480</td>
</tr>
<tr>
<td>220</td>
<td>1300</td>
</tr>
<tr>
<td>240</td>
<td>900</td>
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<tr>
<td>260</td>
<td>650</td>
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<td>280</td>
<td>580</td>
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<tr>
<td>300</td>
<td>520</td>
</tr>
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<td>350</td>
<td>270</td>
</tr>
</tbody>
</table>

Conclusions

1. In case of partial LOCA at MTR fuel element, the boiling phenomenon at the immersed part of the fuel produces a cooling mechanism which can cool the exposed part of the fuel element.

2. By using an experimental simulating system, the wetted zone length above the water level was measured for various water levels and channel power. It was found that that value increases with the channel power. The water level has a small affect on that value.

3. By using a conservative one dimension conduction model, the maximum fuel temperature was calculated for various water levels. It was found that the melting temperature of about 650 °C was obtained at a water level of 260 mm at a channel power of about 200 W. At a higher channel power the fuel element temperature decreases because of intensive boiling at the bottom of the channel which causing a better cooling mechanism at the top of the channel.

References


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Core Overheating Event Probability Analysis due Lack of Passive Cooling in a Typical MTR Pool Type Research Reactor by FTA Study

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The maximum allowable cladding surface temperature is a safety limit that is prohibited to exceeded, neither due to forced cooling failure nor as a result of natural convection cooling absence.

Most of the MTR Pool-Type Research Reactors (RR) primary coolant loop are designed to enable downward forced core cooling flow, during operation nominal conditions. However, post normal or emergency shutdown, the cooling pumps are arrested and the residual heat in the core is removed by upward natural convection, which involves a reversal of the flow direction. The flow path enabling the natural convection is formed by opening a FLAPPER valve located at the bottom of the core (Figure 1). The valve consists of a stainless steel semi-closed box with a cover plate, normally closed as the coolant is forced downward through the core and designed to open as the forced core cooling stops.

Post shutdown of the reactor, the cover plate is opened. The opening of the Flapper cover plate enables an upward passive flow between the cold plenum water at the bottom entering the core and the hot fluid exiting the upper plenum of the core. The existence of the undisturbed flow path is essential for safely transferring the residual heat, by Natural Convection to the cold pool, until the steady state temperature of the core is obtained following the shutdown process.

A certain malfunction of the Flapper, obstructing the flow path, and a failure of the backup electrical generator feeding the cooling pumps, is possible to result into an accidental scenario of Loss of Flow Accident, in the MTR. Hence, it is the purpose of the presentation to outline the accidental scenario through a Fault Tree Analysis (FTA) based on the typical design of a Flapper in the bottom of the RR core. A FTA is an analytical technique, whereby an undesired state of system is specified and analyzed in the context of its environment and operation, in order to find all credible ways in which the undesired event can occur.

Through the presentation, the accidental "top event" of core overheating after normal shutdown is analyzed by a FTA diagram down to "primary / basic" events, for which the probability of occurrence is provided. Finally, two main outputs of the analysis will be presented:

a) The minimal “cut set” of the analysis, namely the smallest combination of the component failures with the highest probability of occurrence, involved to create the accidental scenario defined as the “top event”.

b) The analysis result of the overall probability value of occurrence of the "top event", based on typical data of failure rates and/or malfunctions of the investigated "basic events". Furthermore, the result will include a typical sensitivity analysis by changing some input values.

The author believes that the simple analysis presented should be a precious example of how useful is to investigate faults in engineering safety features by a probabilistic analytical tool, in order to manage wisely the maintenance program.
Figure 1: Typical FLAPPER valve at the bottom of the MTR.

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Bottom-Reflooding of PWR Core with Transient Two-Phase Boundaries and Rod Ignition

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INTRODUCTION

The thermal and hydraulic characteristics of a fuel channel in a pressurized water nuclear reactor (PWR) core during a bottom-reflooding phase of a loss of coolant accident (LOCA) scenario is investigated. Numerical results are illustrated for a partially uncovered AP600 type core.

For severe accident mitigation and management in PWRs, quenching by injection of cold water at the bottom of an overheated and partially oxidized core is an important measure. In a partially uncovered core, three major regions may be identified: single-phase liquid at the bottom followed by a two-phase (vapor-liquid) layer between relative heights \( \mu_1 \) and \( \mu_2 \); and an upper most dry region of single-phase vapour with a relative length of \((1-\mu_2)\).

In the present study, the propagation in space and time of the two-phase region's boundaries are established following a step change in the coolant inlet mass flow rate. In addition, tracking the conditions at the dry region allows coupling of the thermal solution with the formation of oxide layer resulting from the chemical interaction between the cladding surface and the surrounding steam. This, in turn, facilitates the calculation of the time at which runaway clad ignition reaction starts in the uncovered fuel metal cladding.

MODEL

The model consists of coolant fluid that flows upward along a vertical channel of total length \( L \), and constant flow cross section area, \( A_f \). Sub-cooled, single phase liquid (water) enters the channel at its bottom. At a certain height, boiling commences and an ensuing two-phase mixture (gas-liquid) region is developed. The first flow region (sub-cooled zone at the bottom) and its time dependent boundary location (lower boiling-boundary, BB), is solved as in a previous work by the authors [1]. A two-phase region develops above BB and extends up to point DB (dry boundary), which defines the beginning of a dry-zone where the two-phase mixture culminates with a superheated vapor region. The mathematical formulation of the problem consists of the two governing equations for the conservation of mass and energy in the fluid.

A heat balance over the fuel domain \((0 \leq z \leq L)\) yields an equation of the type:

\[
\dot{m}\left[C_L(T_s-T_o)+r+C_G(T_V-T_S)\right]=q'_o\int_0^L \sin \frac{\pi z}{L} \, dz + \Delta Q_{CHEMICAL}
\]

(1)

where \( \dot{m} \) is the water inlet mass flow rate (initially equal to the rate of steam production), \( C_L \) and \( C_G \) represent the water specific heats in the liquid and gas phase regions, \( r \) is the latent heat of vaporization, and \( T_o \) and \( T_s \) are water inlet and saturation temperatures. The RHS of eq. (1) consists of two different heat sources: the nuclear heat generated per unit length, \( q' = q'_o \sin \frac{\pi z}{L} \), and the chemical energy, \( \Delta Q_{CHEMICAL} \), due to cladding oxidation by steam in the dry zone at \( \mu_2 L \leq z \leq L \).
In order to calculate the transient heat and mass transfer processes along the fuel channel it is assumed that liquid level within the heated channel is initially constant. This is maintained by a constant initial liquid (water) inlet mass flow rate, \( \dot{m}_0 \), that balances the amount of liquid evaporated within the two-phase region and thus, holding a constant boiling boundary (BB) level at a height \( z_l(t=0^-) = \mu_l(t=0^-) \cdot L \), where \( \mu = z/L \). The transient process is triggered by a step change in the incoming mass flow rate from \( \dot{m} \) to \( \lambda \dot{m} \), where \( \lambda \) is a prescribed factor which may be greater or smaller than one. The initial temperature distribution along the channel is assumed to correspond to the temperature that existed in the core during normal operation.

The mass and energy balance equations in the metal and in the flow channel along the fuel rods are simultaneously solved. A semi-analytical solution method, coupling the transient thermal-hydraulics of the fluid in a vertical heated nuclear fuel channel, with the clad thermal behavior allows solution of the necessary balance equations. In the two-phase fluid the velocity field is analytically determined followed by a numerical solution for the water density. This is an extension to the authors’ previous work [2] in which a constant liquid level \((\mu L)\) was assumed in the core.

The calculated fuel rod thermal parameters include the thickness of the oxide layer, \( \delta \), and the clad surface temperature, \( T_c \), accounting for the steam-zirconium alloy metal reaction. In order to simplify the solution in the dry region, a lumped parameter approach is taken by using the time dependent spatial average value of \( T_c \) over the region to calculate the rate of chemical reaction between zircaloy and steam.

The non-dimensional height of the two-phase region upper boundary (DB) \( \mu_2(t) = z_2(t)/L \) is derived numerically using the solution of the density field, \( \rho(z,t) \). Generally, DB is defined as the point along the channel at which the two-phase density becomes equal to the saturated vapor density \( \rho_g \), i.e.

\[
\rho(z_2,t) = \rho_g \Rightarrow \mu_2(t) = z_2(t)/L \tag{2}
\]

In the first stage of the solution, the velocity field is derived analytically. Then, in the second stage of solution, a single first order PDE is formulated from which the density field \( \rho(z,t) \) is numerically solved. From DB point up to the top of the channel, single-phase gas (superheated) conditions prevail. The flow characteristics of the fluid in the dry-zone are derived using an approximate vapor equation of state.

**RESULTS**

Typical results are illustrated in Figure 1 showing the time dependent special average cladding surface temperature at the dry zone, \( T_c(t) \), for various flow steps \( \lambda \). A partially uncovered AP600 type nuclear reactor core is considered in these results. As indicated before, the parameter \( \lambda \) indicates the step change in the steady state inlet flow rate at the beginning of the transient. Thus, \( \lambda=1 \) represents a special case in which the initial inlet flow was kept unchanged. In that case, the spatial average dry zone temperature increases continuously from its operational level of about 600°C reaching an ignition threshold temperature of about 1500°C at about 140 sec. At that time a rapid oxidation reaction takes place and a sharp temperature escalation is observed. Increasing the flow multiplier, \( \lambda \), up to \( \lambda=2.3 \) generally delays the clad ignition time. For instance when 50% step increase in the reflooding rate (\( \lambda=1.5 \)) is used the ignition time is about 160 sec.
Increasing $\lambda$ above about 2.5 delays the ignition time indefinitely. For example, Fig. 1 shows that increasing the reflooding rate by 300% (or $\lambda=3$) prevents ignition. The rate of temperature increase subsides and its absolute value does not reach ignition threshold.

![Figure 1: Cladding surface temperature histories for various $\lambda$](image)

**CONCLUSIONS**

A theoretical model is developed for estimating the transient thermal hydraulic conditions along a partially uncovered fuel channel subjected to a step change in its inlet flow rate. The solution method presented in this work has been demonstrated as a computational tool for anticipating possible outcomes of different rates of mitigation of severe accidents and as a possible management aid in light water nuclear reactors under LOCA conditions.

**REFERENCES**


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Validation of BGCore System for Burnup Calculations

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INTRODUCTION

BGCore is a software package for comprehensive computer simulation of nuclear reactor systems and their fuel cycles. BGCore interfaces the Monte Carlo particles transport code MCNP4C\(^1\) with a SARAF module – an independently developed code for calculating fuel composition during irradiation and spent fuel emissions following discharge. In BGCore system, depletion coupling methodology is based on the multi-group approach\(^2\) that significantly reduces computation time and allows tracking of large number of nuclides during calculations.

The objective of this study is validation of the BGCore system against well established and verified, state of the art computer codes for thermal and fast spectrum lattices.

BEHCHMARK DESCRIPTION

PWR fuel assembly benchmark

In this part of the benchmark, the BGCore system was validated against BOXER code, which is 2D fuel assembly transport and burnup code\(^3\). The objectives of the benchmark are to compare the burnup dependent k-inf and isotopic composition calculated by BGCore and BOXER codes. The burnup calculations were performed for a typical UO\(_2\) 17\(\times\)17 PWR fuel assembly assuming typical PWR operational conditions.

Full core lead cooled fast reactor benchmark

At this stage, BGCore system was used for the full core burnup calculations of a lead cooled fast reactor (LFR). The design parameters of considered LFR core can be found in Reference 4. The results of the calculations were compared with those performed with MCNP/ORIGEN-2\(^5\) coupling code MCODE\(^6\) developed at MIT.

RESULTS

The results of the calculations are presented in Fig. 1 and 2 for PWR fuel assembly benchmark and in Fig. 3 and 4 for LFR benchmark. The results of neutron multiplication factor are in very good agreement for both cases. The maximum error in k eigenvalue is below 0.8% for thermal lattice and below 0.2% for LFR core. In LFR case, the results of actinide inventory prediction are also in a good agreement (Fig 4). However, larger differences were observed in actinides concentration in PWR assembly benchmark (Fig 2). The possible source of discrepancy is in different neutron cross section libraries utilized by BOXER and BGCore codes (JEF-1 and JEFF-3.1 correspondingly).

To summarize, results of performed benchmark give a confidence that BGCore can be used for the analysis of both thermal and fast reactor systems.
ACKNOWLEDGEMENTS
The MCODE burnup calculations for Lead Cooled Fast Reactor were performed by the third author during his tenure as a Visiting Scientist at MIT in the course of the “Flexible Conversion Ratio Fast Reactor Evaluation” project.

Figure 1. PWR assembly k-inf vs. irradiation time

Figure 2. LFR core k-eff vs. irradiation time
Figure 3. Difference in number densities for important actinides. PWR assembly

Figure 4. Difference in number densities for important actinides. LFR core - representative assembly
REFERENCES


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Thermal-Hydraulic Feedback Module for BGCore System

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INTRODUCTION

The need for accurate fuel management modeling in Generation IV (LFR, HTGR, etc) reactors has motivated the development of a new and comprehensive code (the BGCore system) for core analysis of advanced reactors. This effort is justified since there are currently no high fidelity codes, which are capable of performing for different types of advanced reactors calculations. Reliable modeling of the core performance requires an adequate modeling of a wide range of physical processes, such as fuel depletion and the temperature distribution in the main core components.

The BGCore system uses the Monte Carlo code MCNP for predicting the power distribution in both thermal and fast reactors with complex 3D heterogeneous geometries. Special depletion and decay routines are incorporated in order to update the fuel composition as a function of time. Since cross sections are affected by the fuel temperature (e.g., Doppler effect) and by the neutron spectrum (through moderator density), the BGCore system incorporates a new thermal-hydraulic feedback module (THERMO) that evaluates the fuel temperature and the moderator density at predetermined time intervals. In addition, the THERMO module provides the capability to monitor whether the safety limits of the core are satisfied. This includes several important parameters, such as:

- Maximum fuel temperature, which is limited by the fuel melting point.
- Maximum cladding temperature, which must be below the limit that assures its structural integrity.
- Coolant flow rate and pumping power that may affect the conversion cycle efficiency and in some cases may affect the integrity of structural materials.
- Thermal expansion of core components.

METHODOLOGY

The THERMO module calculates the temperatures distribution in core components by solving the convection/conduction heat balance equations. In addition, the module calculates the coolant flow distribution in the core channels by requiring the pressure drop in all channels to be uniform. These calculations are performed at selected time-intervals during the fuel depletion. In each computation cycle, the THERMO module receives as input the power and the fuel burnup distributions in the core from the neutronic solver. The geometrical arrangements of the fuel assemblies as well as the initial mass flow rate and inlet coolant temperature are defined as input.
The solution procedure is based on the following simplifying assumptions:

- Coolant flows in non communicating sub channels.
- Each sub channel is divided into sub volumes (voxels). The thermal conductivities of the cladding $k_c$ and the gap $k_g$ as well as the physical properties of the coolant (density, viscosity, specific heat etc.) are all assumed constant in each voxel.
- The fuel pellet is also subdivided into a number of radial zones. The fuel thermal conductivity is assumed to be constant in every such zone.
- The heat-transfer coefficient ($h$) between the fuel and coolant is calculated by the use of convection correlations.
- The heat generation in each fuel voxel is uniform.
- No heat is generated in cladding or coolant.

The THERMO module contains an extensive library of thermo-physical properties of most of the common reactor materials as a function of temperature and pressure. Thermo-physical data base currently includes thermal conductivity, density, heat capacity and viscosity for the following core materials:

- Three coolants: Lead, Lead-Bismuth and water.
- Five gap filling materials: Helium, Argon, Krypton, Xenon and Metallic (Lead or Tin alloys).
- Four fuel types: UO$_2$, MOX, UC and Metallic fuel (U-TRU-Zr).

The THERMO module contains a set of correlations predicting the heat transfer coefficient ($h$) and friction factors for three common channel geometries:

- Square array of cylindrical fuel rods.
- Triangular array of cylindrical fuel rods.
- Channeled fuel blocks.

The main outputs of the THERMO module are:

- Average and maximum fuel temperature in each voxel.
- Maximum temperature of the cladding.
- Temperature and density of the coolant in each voxel.
- Velocity and mass flow rate of the coolant in each channel.
- Pressure drop across the core.

**CALCULATION APPROACH**

First, the module reads the initial conditions from the input file containing the inlet pressure, inlet temperature, total mass flow rate of the coolant and the coolant type. In addition the following channel specifications are extracted: geometry parameters, power distribution, grid location and material composition of the fuel rods. An iterative procedure is used in order to calculate the coolant flow rate in each channel. First, a uniform mass flow distribution is assumed and the pressure drop in each channel is calculated. The pressure drop is then used to modify the flow rate for the next iteration. This procedure is repeated until the pressure drop in all channels satisfies the given convergence criteria.
Once the pressure drop is converged, the module calculates the temperature distributions of all core components (coolant, cladding, fuel, etc.) and prepares the output file containing all the relevant data. Typically, convergence is obtained within a few iterations.

**VERIFICATION**

The THERMO module was validated by comparing its results with those of similar computer codes. We present here two of the performed verification cases:

1. Lead cooled fast reactor with square arrangement of cylindrical fuel rods containing metallic fuel (U-TRU-Zr) cladded with HT-9 cladding analyzed by SUBCHAN code [1].
2. PWR core calculated with SILVER code [2].

Typical results of the first case are compared in Figures 1 and 2 and in Table 1. Generally, the coolant temperatures are practically identical. The small differences in the fuel temperature stem from neglecting in BGC:Core the effect of the oxide layer formed on the cladding. Table 1 show that the pressure drop and the mass flow rates are in very good agreement. The differences are up to 0.5%.

![Figure 1. Temperatures in the HOT channel](image1)

![Figure 2. Temperatures in the AVE. channel](image2)

<table>
<thead>
<tr>
<th></th>
<th>SUBCHAN</th>
<th>THERMO</th>
<th>Difference</th>
</tr>
</thead>
<tbody>
<tr>
<td>Flow Rate [kg/s]</td>
<td>1.05</td>
<td>1.12</td>
<td>1.05</td>
</tr>
<tr>
<td>Press. Drop-Total [Pa]</td>
<td>8.51E-05</td>
<td>8.61E-05</td>
<td>8.56E-05</td>
</tr>
<tr>
<td>Friction losses [Pa]</td>
<td>2.89E-05</td>
<td>2.74E-05</td>
<td>2.89E-05</td>
</tr>
<tr>
<td>Acceleration losses [Pa]</td>
<td>3.15E-02</td>
<td>2.21E-02</td>
<td>3.22E-02</td>
</tr>
<tr>
<td>Form losses [Pa]</td>
<td>1.01E-05</td>
<td>1.13E-05</td>
<td>9.71E-04</td>
</tr>
<tr>
<td>Gravity losses [Pa]</td>
<td>4.71E-05</td>
<td>4.73E-05</td>
<td>4.69E-05</td>
</tr>
</tbody>
</table>

Figures 3, 4, 5 and 6 compare the coolant temperature, coolant density and fuel temperatures in a PWR core, calculated by the SILVER code and the THERMO module. Here again, the coolant temperatures evaluated by the two codes are practically identical. The difference in the fuel temperature is a result of the difference in the heat transfer coefficient (0.7% difference between the two codes).
CONCLUSIONS
Integration of the thermal-hydraulic feedback (THERMO) in the BGCore system provides the ability of more accurate neutronic calculation. Moreover, the THERMO model provides the user with computational tools to decide whether the safety limits are satisfied. In comparison to other codes, currently in use, the module gives a very good agreement. However, unlike other codes, which sometimes are limited to specific core designs (constant geometry or one type of coolant), the THERMO module enables accurate simulation of a wide range of reactors designs. Currently, the module is used to study the core behavior with and without the effect of thermal-hydraulic feedback.

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INTRODUCTION

One of the risks in a post core accident at nuclear reactors is the energetic interaction of the fuel melt and the coolant (known as Fuel-Coolant Interaction or FCI). A vapor film is formed between the melt and the coolant and prevents the contact of the coolant with the melt. During film boiling the melt is cooled by radiation and conduction through the film of vapor. When the temperature of the melt is not sufficient to maintain the film of vapor, it collapses and the coolant reestablishes contact with the dry surface of the hot melt. This interaction may lead to very rapid and violent boiling. In this rapid process a large quantity of liquid is vaporized and its expansion can result in destructive mechanical damage to the surrounding structures, hence the name "vapor explosion". A sad demonstration to that kind of explosion took place in the nuclear power plant in Chernobyl.

The temperature in which the film of vapor cannot be maintained anymore has been a subject for many studies worldwide for many decades and in many methods.

The present study deals with the thermal reaction caused by the penetration of a high temperature metal sphere into a liquid reservoir which represent a FCI. The temperature in which the film of vapor cannot be maintained anymore ($T_{\text{MHR}}$) was measured using several types of metals with different size.

In the present experimental study, a criterion for energetic FCI was introduced based on the initial temperature of the fuel and the $T_{\text{MHR}}$.

EXPERIMENTAL APPARATUS AND PROCEDURE

Figure 1 shows the schematic diagram of the experimental apparatus. An electric resistance furnace was used for heating the spheres. The liquid reservoir was mounted on a magnetic stirring device with an internal heater. The homogenous temperature of the distilled water in the reservoir was attained by a stirring magnet. The liquid reservoir was open to the atmosphere. An external heater was used in order to supply the necessary heat to the water. A K type thermocouple was used to measure the temperature of the sphere. The thermocouple was entered to the center of the sphere through a pre-made hole. The difference between the sphere surface temperature and its center temperature was found to be less than 2°C. The thermocouple was connected to a data logger. The data was acquired at frequency of 25 Hz using a PC computer while the cooling process was photographed.
Prior to the experiments the liquid reservoir was filled with distilled water which was saturated for degassing. During that time the water were stirred using a magnetic stirring device in order to promise homogenous temperature. In the meanwhile the examined sphere was heated up to 600°C using the furnace.

The stirring of the water was stopped prior to the conduction of the experiment in order to avoid movement of the water (which affects the saturation curve). Then, the sphere was immersed into the liquid reservoir while its temperature is recorded in the data logger. The recording of the temperature continued until the sphere temperature reached the ambient temperature.

**RESULTS**

Figure 2 shows the sphere's temperature versus time as measured in the experiment with a 12 mm copper made sphere.

![Figure 2. The temperature of a 12 mm copper made sphere versus time](image)
The heat flux from the spheres was calculated from the decrease in the sphere's temperature, the sphere's size, its density and its heat capacity using the following equation:

\[
q_i^u = \frac{dT}{dt_i} \frac{C_p(T) \cdot \rho \cdot V_{sphere}}{A_{sphere}}
\]

Figure 3 shows, as an example, the heat flux from a 12 mm copper made sphere versus its temperature during the cooling process. The temperature where the film of vapor collapses (T_{MIF}) was located by the sharp rise of the heat flux during the cooling process – from values typical for film boiling towards higher values which are typical for nucleate boiling.

![Figure 3. The heat flux from a 12 mm copper made sphere versus its temperature](image_url)
Figure 4 shows $T_{MIF}$ for copper made spheres of different size versus subcooling of the water. $T_{MIF}$ for steel made spheres is shown in Figure 5.

**Figure 4. $T_{MIF}$ for copper spheres**

Three categories of interaction were found based on the initial temperature of the sphere and the $T_{MIF}$ (which is a function of the material, the size and the subcooling of the water). In the first category, the initial temperature of the sphere is lower than $T_{MIF}$. In this category there will be no film boiling at all and hence no collapse of it so in the interaction there won't be any vapor explosion.

In the second category, $T_{MIF}$ is very low compare to the initial temperature of the sphere. In this category there will be a film of vapor which will collapse and so the interaction will be energetic. Because of the low $T_{MIF}$, the energy delivered to the surroundings will be low.

In the last category, $T_{MIF}$ is high but the initial temperature of the sphere is higher. In this category there will also be a film of vapor which will collapse and because of the high $T_{MIF}$ this interaction will be very energetic.

The three categories can be summarized by the following equation:

$$\phi = \eta \cdot \left( T_{\text{initial}} > T_{MIF} \right) 
\cdot T_{MIF}$$  \hspace{1cm} (2)

Where:

$\phi$ - A figure of merit for the FCI magnitude
$\eta$ - A coefficient which is a function of the sphere's size and substance

**CONCLUSIONS**

A criterion for the energetic FCI was introduced. The criterion is based on the initial temperature of the fuel and the $T_{MIF}$ (which is a function of the fuel material, the size of it and the subcooling of the coolant). The most energetic interaction will take place if the $T_{MIF}$ is high and the initial temperature of the fuel is higher.

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Noise Reduction of an Ultrasonic System for the IRR2 Research Reactor Tank

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As part of the IRR2 reactor preventive maintenance, an ultrasonic scanning system was developed for inspection of the reactor tank wall. The design validation was carried out in a full-sized mock-up. An important issue tested in the mock-up experiments was the ultrasonic scanning system sensitivity to electromagnetic noise. Indeed, the system was approved only after the noise levels were found to be tolerable. However, when the Ultrasonic Inspection System was first operated in the IRR2 reactor, unacceptable noise-levels were measured, i.e., system immunity to electromagnetic noise demonstrated in the mock-up testing proved to be irrelevant for the IRR2 reactor itself. Various methods of noise reduction were attempted; consequently, noise interference was reduced to an acceptable level.

SYSTEM DESIGN

The scanning system consists of: a computer controlled manipulator carrying the ultrasonic search-unit, a folding manipulator arm carrying a search unit of 16 ultrasonic transducers, a motion control unit, a drive motor, computerized ultrasonic instrumentation and an array of computers designed for storage and processing of the acquired data. A scheme of the inspection system is presented in Figure 1.

The ultrasonic inspection system is based on a 16-channel industrial ultrasonic scanner[1]. The electrical pulses, needed for ultrasound pulse generation and initial amplification of the RF signals coming from the transducers, are implemented by four Remote Pulse Preamplifiers (RPP). These RPPs rotate with the manipulator, therefore, the connection between them and the main ultrasonic system is through slip-rings.
Figure 1: A schematic drawing of the manipulator and the ultrasonic system

THE PROBLEM
Electromagnetic noise is known to be layout dependent and must be often handled on each location exclusively. Frequently, noise can be a consequence of improper grounding, power supply, electronics devices, magnetic machines at near and far surroundings, illumination, and the like. In the mock-up testing, noise was successfully suppressed down to 2-3% of the signal, which is a better result than the required. However, in the final installation at the IRR2, noise appeared as bursts of up to 15% of the signal amplitude, which is an unacceptable noise level (see Figure 2).

Figure 2: Signal and noise burst on A-scan. The shaded area is the time interval used to produce the C-scan as shown on Figure 6.
SOLUTION

By increasing the transmitted signal, the SNR improved but the signal saturation decreased the dynamic range. This implied that the noise was additive in the path from the transducer to the ADC. Further investigation revealed that the noise burst source origins from the motor control required for the manipulator motion. The noise was radiated by the shielded cable connected to the motors control and penetrated to the system through the slip rings. Noise attenuation improvement was achieved by double shielding the cable and grounding it only on the controller side. The noise frequency produced by the controller was below 10 MHz. For those frequencies, ferrite chokes for eliminating common mode noises are useless; therefore, a toroidal shape choke made of a nano-crystal material was used, but turned out to be ineffective for noise reduction as well. Analog noise filtering of the signal could not be performed since both spectral distributions were similar, as seen in Figure 3.

![Figure 3: Signal and noise frequency distribution](image)

Considered digital filters were not effective, and anyway, the enormous amount of data (over a Tbyte) did not allow digital filtering with the existing hardware. Various methods of noise reduction were attempted and considered. The chosen solution is demonstrated in Figure 4.

![Figure 4: Basic scheme of noise reduction](image)
Amplifying the signal before the slip rings by factor A improves the SNR by A. By attenuating the signal, and the noise, by the same opposite factor, the signal returns to the required level, and the noise is reduced by factor A.

![Amplifier and Divider](image)

Figure 5: Amplifier and divider schemes and circuits

**RESULT**

A 50Ω matched load x3 amplifier and 1/3 voltage divider were designed, see Figure 5. Consequently, the noise was reduced by a factor of 3 and interference reached an acceptable level of 5% of the scale. Statistical measurements were performed to validate the noise reduction. Figure 6b demonstrates a C-scan of a representative region in the tank, where at each point in the scan, the amplitude was recorded at a time interval with no real signals (see Figure 2). That is, figure 6 is a pseudo-colour representation of the noise level. Without noise, it should have been completely black (see color bar on the right). As can be seen, without reduction, the noise level reaches up to 15%. Figure 6a demonstrates the same area scanned after noise reduction. As can be seen, the noise level does not exceed an acceptable level of 5%, as required.
Figure 6: Noise level as it appears on a C-scan image.
   a. With noise reduction; b. Without noise reduction

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[1] ASME Boiler and Pressure Vessel Code, Section XI.

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An Ultrasonic NDE System for the IRR2 Reactor Tank Wall

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ABSTRACT
The development and the validation of a new ultrasonic NDE system in-service inspection of a research reactor closed tank are surveyed. In addition, preliminary results from the inspection of the IRR2 reactor tank are presented. The inspection has shown that the aluminium tank is in good physical condition with negligible corrosion and no significant structural defects.

1. INTRODUCTION
Reactor vessel being a critical component, receives special attention. The condition of the IRR2 reactor tank is constantly monitored in order to ensure its safety. Several methods of estimating the condition of the tank were employed over the last four decades. These methods include an inspection of irradiated specimens that are made of the same material as the reactor tank, post-service inspected of irradiated parts and a micro-hardness inspection of the tank itself. It remains to check if the tank integrity is not compromised by some local, macroscopic flaws that might have evolved during its long life.

Ultrasonic testing is one of the established tools for routine in-service inspection of reactor tanks. In case of nuclear power plants pressure vessels, these periodic non-destructive inspections are mandated by various codes. However, for research reactors, there is less standardization. Significant differences between the construction of power-reactors and research reactors complicate the transfer of inspection technology from the former to the later reactors. For example, the inspection of power reactors generally requires the removal of the vessel top cover, an extremely difficult task in some research reactors. The outer diameter surface of the vessel is inaccessible in many research reactors, unlike modern BWR. In addition, reactors built according to standards require periodic inspections. In aging research reactors, not built according to the aforementioned standards, periodic inspections are not performed. Thus, in the interpretation of results of inspections in research reactor vessels one usually cannot rely on comparison with past inspections.

With these difficulties in mind, the NRCN decided to develop a new ultrasonic NDE system, suitable for the inspection of a research reactor vessel from the inside, without removal of the vessel top cover.
2. SYSTEM DESIGN
The chosen inspection technique is immersion ultrasonic scanning. The inspection is being done without direct contact with the wall. The system consists of a computer-controlled manipulator that carries the ultrasonic search-unit, motion control unit, drive motors, ultrasonic instrumentation and a system of computers designed for data storage and processing. The inspection system scheme is presented in Figure 1.

The ultrasonic search-unit is carried by a custom-designed manipulator with three motion axes: the elevation $Z$, the rotations $\Phi$ along the perimeter and rotation $\phi$ of the transducers head, see schematic description on Figure 1.
At first the manipulator is inserted in parallel to the tank central axes. Then, the elbow folds up to a 90 degree position in order to place the transducers-head perpendicular to the tank wall. The folded arm length approximately 1.25 m is dictated by the geometry of the IRR2 tank, but it could be matched to other tank geometry by connecting different segments between the joints.

The manipulator is stirred by a three-motor drive governed by a computerized controller. The position of the manipulator is monitored by a relative encoder that is attached to its motor, an absolute encoder and a home switch. The Z axis also has a two limit switches that are designed to prevent the manipulator from going out of the designed range of motion.

A small motor, located inside a sealed can at the front-end of the manipulator, enables the rotation of the search-unit around its central axis. This motor is controlled by an electromagnetic resolver, a limit switch and a homing switch.

Two compact search-units were developed. One, designed for detection of defects in all the orientations, is composed of two 5 MHz straight transducers and six pairs of 2.25 MHz angular transducers, operated in both Pulse-Echo and Pitch-Catch modes. The other, designed for evaluation of suspect areas, contains a 10 MHz straight transducer and three pairs of 2.25 – 10 MHz angular transducers. These search units provide full coverage of the 8 mm-thick aluminium tank of the IRR2 reactor.

All the scanned data is stored (full A-scan spatial-temporal record) for further evaluation and comparison with successive scanning data. For processing this huge amount of data a system comprising of three computers working in parallel and a two TByte RAID storage device was set up. A dedicated software package was developed that enables the full off-line analysis of the acquired data.

The design validation was carried out in two stages. First, the ultrasonic technique was calibrated on specimens with known flaws, using a laboratory-scale ultrasonic scanner. Some of these specimens were intentionally covered with stone by boiling them in tap-water, in order to simulate the sediment layer in the tank. In the second stage, the whole inspection system was tested in a full-sized mock-up, containing a full-scale segment of a tank wall. The mock-up tank wall contained flaws that were unknown to the inspector. The successful detection of these flaws, with no false alarms, was an important condition for the final approval of the design).
A typical result of the validation experiment is presented in figure 2. There, both void-like and crack-like flaws of various size and orientation are clearly seen in a C-Scan image produced by the angular transducers, operating in the pitch-catch mode.

![C-Scan Image](image)

Figure 2. Detection of various artificial flaws in the validation experiment. The 4 mm X 1 mm C-scan image is produced by 6 pairs of angular 2.25 MHz transducers operating in the pitch-catch mode. The colors in the image represent transmitted amplitude, according to the color-bar on the left.

3. PRELIMINARY RESULTS
The examination of the tank wall showed no sign of serious defects. The tank wall thickness varied between 7.8 mm to 8.2 mm. The difference in thickness between the several plates from which the tank was assembled was seen clearly. Hence, these thickness variations are not a result of degradation in service but rather, a feature of the initial construction.

Several indications of small (~1 mm) inclusions, located inside the wall were seen. These kinds of indications are typical to the manufacturing process, and have been there ever since. In Figure 3 an example of a tiny void of sphere geometry (~1 mm) inside the weld is shown. It can be seen that there is only one bubble in this (typical) part of the weld.
As can be expected after forty years of operation, there were several indications of mild corrosion. Typically, corrosion areas were of concave shape, with maximal depth of up to 0.6 mm and about 20 mm diameter. There are no indications of corrosion defects at the outer reactor surface.
Also, we may conclude that scanning the whole tank wall, the ultrasonic inspection showed no signs of service induced cracks.

Figure 3. Indication from a void in a weld, detected using straight 5 MHz transducer.

4. CONCLUSIONS
A special apparatus was developed for the ultrasonic scanning of a research reactor tank wall, without removal of the vessel top cover. The ultrasonic NDE system has been designed to scan the whole tank wall, store the whole acquired data (full A-scan) and analyse it.
The operation of the newly developed system was practiced using a full-scale mock-up. The inspection technique was validated using a variety of flaws that were unknown to the operators.
The ultrasonic inspection has shown that the aluminum tank is in good physical condition. The welds were found to be free of significant defects. The negligible corrosion that has evolved over forty years of operation can be attributed to the high chemical quality of the coolant/moderator water.

REFERENCES

3 ASME Boiler and Pressure Vessel Code, Section XI.

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Investigation of the Ionization Density Dependence of the Glow Curve Characteristics of LiF:Mg,Ti

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INTRODUCTION

There is continuing interest in the possible use of LiF:Mg,Ti (TLD-100) thermoluminescent detectors in mixed-field neutron-gamma ray dosimetry, and as ionization density discriminators (average linear-energy-transfer (LET) estimators) in various heavy charged particle and neutron-gamma radiation fields in space and air-craft dosimetry, heavy ion medical radiation dosimetry and other applications.

In this work we are studying the dependence of the shape of composite glow peak 5\textsuperscript{(1)} and the high temperature thermoluminescence (HTTL)\textsuperscript{(2)} in the glow curve of TLD-100 on ionization density\textsuperscript{(3)}. The shape of composite glow peak 5 (specifically the ratio of peak 5a/5) has been suggested as an ionization-density dependent nanodosimeter\textsuperscript{(4)} due to the localized nature of the electron-hole recombination giving rise to peak 5a and various authors\textsuperscript{(5)} have promoted the use of the HTTL as an estimator of average LET in space radiation fields due to the enhanced relative intensity of the HTTL at high ionization density.

EXPERIMENTAL MATERIALS AND METHODS

The thermoluminescent detectors (TLDs) employed were 3 mm x 3 mm x 0.89 mm LiF: Mg, Ti (TLD-100) Harshaw chips. Two batches (CNS-4715 and CNS-4945 purchased in June 2002 and November 2006 respectively) were studied in order to investigate the possibility of batch effects. Two pre-irradiation annealing procedures were used: (i) 400 °C for one hour followed by a rapid non-linear cool down in ~15 minutes to room temperature achieved by removing the TLDs from the annealing oven (referred to as standard cooling – SC) and (ii) a programmer- controlled linear slow-cooling rate (SLC) of 100°C hr\textsuperscript{-1} in the oven. A post-irradiation anneal on the reader planchet of 155°C for 6s was also applied in order to remove the low temperature peaks 2 and 3 from the glow curve. A slow and linear glow curve heating rate of 100 °C minute\textsuperscript{-1} from 50°C to 400°C was employed to enhance glow peak resolution and allow the investigation of the behaviour of individual glow peaks.

The following heavy ion irradiations at fluence levels of 10\textsuperscript{9} cm\textsuperscript{-2} were carried out at the Radiological Research Accelerator Facility (RARAF) of Columbia University on two batches of “standard-cooled” TLD-100: (i) 2 MeV and 7.5 MeV alpha particles (ii) 4.5 MeV protons, (iii) 4.5 MeV and 1.7 MeV deuterons (iv) 0.2 MeV and 14 MeV neutrons at dose levels of ~1 Gy on both bare and covered (3 mm plastic) detectors. An additional series of heavy ion and neutron irradiations were carried out on the slow-cooled materials\textsuperscript{(6)} – a procedure which enhances the relative intensity of peak 5a as well as the HTTL. In addition, X-ray irradiations (60 kVp and 250 kVp) were carried out at two different dose-levels of 0.1 Gy and ~250 Gy on both the standard-cooled and slow-cooled materials.
Irradiations at Ben Gurion University using a $^{90}\text{Sr}^{90}\text{Y}$ source were also carried out to serve as a reference-radiation glow curve at low ionization density.

**RESULTS**

Typical glow curves following $^{90}\text{Sr}^{90}\text{Y}$, proton, alpha particle and neutron irradiation are shown in Figs. 1-4.

![Graph 1](image1.png)

Figure 1. TL glow curve of LiF:Mg,Ti following irradiation by beta rays at a dose level of 1Gy. The dominant glow peak (referred to as peak 5) occurs at a $T_{\text{max}}$ of ~205°C. Note the relatively low intensity of the high temperature thermoluminescence (HTTL) between 250-350°C.

![Graph 2](image2.png)

Figure 2. TL glow curve of LiF:Mg,Ti following proton irradiation at a fluence level of $10^9$ cm$^{-2}$ of energy 4.5 MeV. Note the increased intensity of the HTTL. The dominant HTTL glow peak is referred to as peak 7 and occurs at a $T_{\text{max}}$ of ~265°C.
Figure 3. TL glow curve of LiF:Mg,Ti following irradiation by 7.5 MeV alpha particles at a fluence level of $10^9$ cm$^{-2}$. The HTTL is now very intense and is comprised of at least 4 glow peaks. Peak 5 is considerably broader due to an enhanced intensity of peak 5a.

Figure 4. TL glow curve of LiF:Mg,Ti following irradiation by 0.2 MeV neutrons at a dose level of 1Gy. The dominant HTTL glow peak has shifted to higher temperatures (~ 320°C) and is referred to as peak 8.

Initial glow curve analysis based on peak-height and glow peak width has revealed a rich degree of intriguing and previously unobserved characteristics. These will be future-analysed using computerized glow curve deconvolution to study the ionization density behaviour of the components of composite peak 5 (peaks 5a and 5b) and the HTTL (peaks 6,7,8). Some immediate/obvious highlights of the results are outlined below:

1. Very significant differences in the relative intensity and shape (relative intensities of peaks 6, 7 and 8) of the HTTL are observed for the two batches.

2. In agreement with previous studies, the behaviour of the high temperature ratio (HTR) for peak 7 [the HTR is defined as the ratio of the peak height of the HTTL divided by the peak height of peak 5 following heavy ion/neutron irradiation normalized to the same ratio following gamma or electron irradiation] is strongly correlated with ionization density and increases in a well-behaved manner from a factor 10 following 4.5 MeV proton irradiation (3 keV/$\mu$) to a value of 30 following 1.7 MeV deuteron irradiation (30 keV/$\mu$). At higher ionization density, however, the HTR levels off and even decreases from 37.5 following 7.5 MeV alpha particle irradiation (67 keV/$\mu$) to a value of 35 following 2 MeV alpha particle irradiation (150 keV/$\mu$).
The HTR (peak 7) for the low energy neutrons is lower than expected. The leveling off the HTR at high ionization density is an unexpected result which has not been previously discussed in the literature.

1. The HTR for peak 8 behaves in a very erratic manner and is not correlatable with LET. The implication is, therefore, that the use of the HTR in LET estimation should be based on peak 7 alone.

2. The enhanced presence of peak 5a (the low temperature component of composite peak 5) is clearly observed in the alpha particle irradiations and is evidenced by the increased width of composite glow peak 5 and a shift of $T_{\text{max}}$ of composite peak 5 to lower temperatures. This behaviour, however, is not observed following proton or neutron irradiation. This suggests that the combined use of the shape of peak 5 and the HTR (peak 7 only) may have significantly increased potential in mixed-field radiation dosimetry.

3. The presence of peak 5a and the HTTL is enhanced in the slow-cooled materials and suggests that materials for improved ionization density discrimination may be developed by further investigation and optimal usage of highly controlled annealing protocols.

CONCLUSIONS

Initial glow curve analysis based on peak-height and glow peak width has revealed a surprisingly rich degree of intriguing and previously unobserved characteristics. These will be future-analyzed using computerized glow curve deconvolution to study the ionization density behaviour of the components of composite peak 5 and the HTTL (peaks 6, 7, 8). These studies may lead to improved ionization density discrimination using LiF:Mg,Ti TLDs.

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Applying the High-Temperature TL in LiF:Mg,Ti to Mixed Thermal Neutron-Gamma Dosimetry – a Review

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INTRODUCTION

In a recent review article on the subject of High Temperature TL (HTTL) characteristics, Horowitz, Oster and Datz¹ criticized the attempts to use the two peak method in mixed field neutron-gamma ray dosimetry, promoting their firm conviction, as also published in an earlier survey by Horowitz et al.², that this issue is unworthy to be investigated. Although it should have been a review, the article concentrates in the corresponding section only on one of our publications on the subject³, trying to prove their view through our "highly problematic results", as they define it. The "problematics" will be dealt with in specific future publications, but in order to present a realistic and balanced range of views on this issue, a comprehensive review of the results published on this topic is to be given. Many works deal with mixed field dosimetric application of the two-peak method, but for the sake of clarity we will concentrate only on the applications of ⁶LiF for thermal neutron fields, as presented in our publications³,⁴.

SURVEY OF PUBLICATIONS

The idea to use the different LET response of the main dosimetry peaks (defined as 4 and 5) and the High Temperature peaks (mostly denominated as 6 and 7) is not new. Almost from the beginning of studying the thermoluminescence phenomenon and its application to dosimetry, it was recognized that that LiF:Mg,Ti, the most widespread dosimetric material, responds differently when irradiated by low LET (as gamma rays) and high LET (as α particles) radiation.

Already in 1970, Busuoli at al.⁵ presented results on evaluating the light output of the low and high temperature peaks by a dual heating cycle of about 38 s. each. In the first cycle the maximum temperature was limited for readout of only the low temperature peak (at about 210°C), while for the second cycle the maximum temperature was increased for complete readout. No details are given on the experimental system and heating rate. Based on their results, the authors conclude that the method is valid in practice for mixed fields of gamma ray and slow neutrons, for doses of gamma rays ranging from a few tens of rads to 10 rads and for neutron doses ranging from 1 rad to 10⁴ rads, being reliable for normal personal dosimetry.

Marshall et al.⁶ published in 1977 similar results based on dual temperatures readout in one cycle. They conclude that the two peaks method can be used with sufficient precision for albedo dosimetry in personal neutron monitoring. Also Nash and Johnson⁷ published in 1977 results on experiments to determine if the different LET response of the ~200°C and 250°C peaks in ⁶LiF(TLD-600) can be used for mixed gamma-ray and thermal neutron dosimetry. They used 3.2mm x 3.2mm x 0.9mm crystals read by a Harshaw 2271 automatic reader, applying a heating rate of 5°C/s up to 330°C with a constant temperature thereafter. The response of the two peak regions was defined by integration time of the different regions (for 18 s, and from 20 s to 25 s). By analyzing the results obtained for gamma and neutron doses of up to 500 mR, the authors conclude that the accuracy of the method is reasonable and comparable to that of the pair (⁶Li+⁷Li) method. This readout schedule was introduced for routine application in 1976 at Naval Research Labs./USA.
Uray\(^{(8)}\) performed experiments with similar Harshaw crystals, and published their results in 1986. He used annealing, and a slow heating rate of \(3^\circ\text{C/s}\). The light intensity was measured in two separate regions: from \(100^\circ\text{C}\) to \(225^\circ\text{C}\) and from \(225^\circ\text{C}\) to \(400^\circ\text{C}\). He defined empirical constants for the two regions for gamma and neutron radiations, which can be used for mathematical evaluation of the doses in a mixed field. He found evidence of a supralinear component in the HTTL region also for the lower dose range, which seems not to be significant due to the statistical uncertainty. Additional experiments to resolve High Temperature and Low Temperature peaks information in a mixed field are presented by the same author\(^{(9)}\), this time by using an adaptation of an unfolding program used in gamma spectroscopy.

Horiuchi and Sato\(^{(10)}\) published in 1992 a work on simultaneous evaluation of neutron and gamma dose with a single \(^6\text{LiF}\) TLD by deconvoluting the glow curves into 7 peaks, with shapes approximated by Gaussian functions. A commercial TLD reader using a heating plate with a \(5^\circ\text{C/s}\) heating rate was employed. The different peak area ratios were studied for combinations of gamma and neutron doses, and the method was found satisfactory for mixed field dosimetry. The authors recommend using specific lot response factors in order to reduce the dependence on the composition of ingredients in the TLD material.

Gambarini and Roy\(^{(11)}\) studied the glow curves obtained from \(^6\text{LiF}\) chips after exposure to high doses of thermal neutrons. The readout was performed using a heating rate of \(6^\circ\text{C}\) up to \(330^\circ\text{C}\), and a glow curve deconvolution into Gaussian peaks was applied. In their publication in 1997, they present ratios of peaks \((4 + 5)\) to peaks \((6 + 7)\) areas as a function of fluence, and recommend to use dosimeters of the same batch as used for the reader calibration.

The most detailed work on the subject was published by a group from Stanford and Oak Ridge/USA. In a first published work in 1991 by Liu and Sims\(^{(12)}\), the high temperature peak characteristics of reader annealed TLD-600 were analyzed. Although using relatively thick TLD chips (0.9 mm), a high heating rate of \(25^\circ\text{C/s}\) from \(50^\circ\text{C}\) to \(300^\circ\text{C}\) and a hold time of 6.7 s at \(300^\circ\text{C}\) were applied, by using an automated Harshaw 8800 hot gas reader. Using deconvolution to separate between the contributions in the high and low temperature peaks regions was not satisfactory, therefore the Region of Interest (ROI) method was applied. The authors conclude that the high temperature peak methodology can be applicable to most radiation protection dosimetry situations. Based on successful tests with mixed doses of gamma and thermal neutron radiations, a practical four-element neutron-photon-beta TL dosimeter was developed\(^{(13)}\). It consisted of a combination of a Harshaw beta-gamma TLD and an albedo neutron dosimeter, based on using the high temperature peak methodology with a 0.38 mm thick TLD-600 chip and a filtration algorithm. The neutron-photon signal separation was found adequate for routine application of the dosimeter. In a later article\(^{(14)}\), additional tests are presented, including comparison with an algorithm which did not utilize the high temperature peaks. The algorithm based on the HT peaks analysis passed all tests of mixed field exposures, with a smaller standard deviation relative to the conventional algorithm. It deserves to be mentioned that this mixed field radiation dosimeter, based on the high temperature peak characteristics of \(^6\text{LiF}\) detectors, was registered as US Patent nr. 5,340,985 in 1994.

Among the most recent articles, Delgado et al.\(^{(15)}\) present in 2007 a modified procedure for separate estimation of neutron and gamma doses in mixed radiation fields with one TLD-600 element, but not relaying exclusively on the HT region. A peak resolving glow curve analysis code was employed to define the characteristic glow curve structure for pure neutron and gamma irradiations. However, the difference in the HT region plays the major role, and is the key to the procedure for determination of the relative portion of the neutron and gamma responses. Linear heating rates at \(5^\circ\text{C/s}\) up to \(350^\circ\text{C}\) were employed. Recently, Triolo et al.\(^{(16)}\) studied the glow
curves of TLD-600 (among other TLD materials) exposed to thermal neutrons, obtained with a very low heating rate. In their recent publication (2007) the view is expressed, that it is possible to obtain information on the mixed field components by deconvolution of the glow curves by using general order kinetics.

DISCUSSION

As seen above, in most of the works published, a slow heating rate was applied, as only a small lag of the crystal temperature is present in this case, and the shape of the glow curve is more distinct, enabling visual observation of its various features. The results presented in our works\(^3\),\(^4\) were intended to investigate the possibility of practical application of the HTTL information in a specific dose range in routine dosimetry, where a fast heating rate (25°C/s) is applied and the gas heating method replaced the contact heating. Our results and conclusions are to be viewed in the general complex of efforts presented here to investigate the feasibility of applying HTTL to mixed thermal neutron-gamma dosimetry, which indicate a continuous research effort in the field, from the beginning of use of the TLD up to the present days.

The main reasoning of Horowitz et al.\(^1\),\(^2\) against using the two-peak method with one LiF:Mg,Ti crystal are the non-constancy of the crystals impurity content, the non-linearity of the peak 7 response to gamma radiation, and the lack of knowledge on precise mechanisms of the TL process. They mention a list of experimental parameters that may affect HTTL dose–response, consisting of the influence of the annealing procedure and of the heating rate, the concentration of Mg, Ti, OH, O impurities that can change the spectral emission properties, the batch characteristics, the spectral response of the photocathode and other components, errors in the estimation of sample temperature, storage conditions and the method of background subtraction. However, all these factors are present not only when dealing with HTTL, but also for the TL dosimetry in general. TLD is a relative method, and the crystals must be carefully calibrated, using proper protocols. In spite of the shortcomings mentioned above, which were much more severe and critical in the early days of TLD, this method was introduced in practice and has become the most widespread and accepted. The two peaks methodology was and is investigated, the results are far from being erratic, and even routine applications were applied. A method should be considered feasible for practical applications, if stringent QC procedures can ensure stable and reproducible results for a known parameters range. We have no abundant choice of neutron dosimetry methods, and in our opinion every effort in this field is positive and is to be encouraged.

REFERENCES


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Reassessing TLD100 doses by the Residual dose – a Review

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INTRODUCTION
In operational dosimetry, the residual response of the TL material is taken as a negligible interference to future readout results. The consent that this quantity can by neglected is based on the assumption that it has almost no effect on future dose measurements, being small compared to the original dose. The residual dose is defined as the measured dose by an extra readout of a TL material after it was read and reset.

Theoretically, the heating of the TL material (as a part of its reading process) resets all dose information stored in it. However, in routine operation a high heating rate is applied (25°C/second) up to the temperature of 300°C (chosen as not to damage the Teflon coating in the TLD cards). This reading process does not completely depopulate the deep traps of the TL material and some charge carriers are left trapped. Previous works showed that the measured residual information left in a TL chip is proportional to the chip original. Therefore, by using equation 1, it is possible to set a reassessment factor \( R_f \) for the TL chip, using an original, known dose \( D \) and its measured residual dose \( D_R \).

\[
R_f = \frac{D_R}{D}
\]

The reassessment factor, by taking into consideration the low yield of the residual glow curve and other operational characteristics (background readings and electronic noise of the reader) will determine the LOD (limit of detection) of the reassessment process. The only practical way to lower the LOD of the reassessment process is to increase the yield of the residual readout. This work presents some methods to reassess doses by using the residual glow curve.

MATERIALS AND METHODS
The measurements were performed using standard Harshaw/Bicron manufactured TLD100 cards containing 3\( \times \)3\( \times \)0.38 mm\(^3\) chips. The readouts were performed by an automatic 6600 Harshaw/Bicron reader and the temperature profile consisted of a preheat to 50°C, then heating at 25°C/second up to 300°C keeping this temperature to the end of the reading cycle and then cooling to room temperature naturally. The total reading time of a standard readout is 13.3 seconds and this parameter was increased to 30 seconds for the "extended readout period" to increase the depopulation of deep traps in the residual glow curve reading. Irradiations were carried out using the built in \(^{90}\)Sr-\(^{80}\)Y source of the Harshaw/Bicron 6600 automatic reader and u.v. irradiation was performed by a 15W UV lamp (\( \lambda = 254 \)nm).
THE GLOW CURVE OF THE RESIDUAL READOUT
A standard glow curve of a TLD100 chip irradiated to 30 mSv is shown in figure 1. A second, following readout of the same chip using the same reading parameters is shown in figure 2. It can be seen that the glow curve of the residual reading is significantly different from the original one. By integrating the residual glow curve area, we obtained about 0.2% relative to the area of the original glow curve, in accordance to results obtained in a previous work for the same exposure value\(^2\). Because of the low yield of this reassessment process, the statistical errors are very high for the exposure range chosen and the results are of no practical use.

![Glow Curve Diagram](image1)

Figure 1: A typical standard glow curve from the reading of a TL chip that was irradiated to 30 mSv, for readout of 13.3 seconds \((1)\).

By analyzing the shape of the residual readout glow curve, it can be observed that it is possible to increase the yield of the residual readout by increasing the readout time. The buildup towards the end of the glow curve is not finished and by extending the readout time the area under the residual glow curve will increase, therefore the reassessment factor will grow and the LOD could be lowered.

![Residual Glow Curve Diagram](image2)

Figure 2: A typical residual reading glow curve of a TL chip that was irradiated to 30 mSv, for readout of 13.3 seconds \((1)\).
INCREASING THE REASSESSMENT FACTOR
"Extended period" method
Figure 3 shows the residual glow curve of a TLD100 chip irradiated to 30 mSv read with a standard readout (as in figure 1) and then re-read again for an extended reading period of 30 seconds. By increasing the readout time to 30 seconds, we get a complete glow curve, without cutoff. The main peak starts after the region where the dosimetry peaks are present (after ≈11 seconds) and may indicate transitions from the deeper traps which are possible due to the long heating.
Analyzing the glow curves shows that on both sides of the main peak there are interferences. A small peak before the main peak and a tail after it till the end of the reading process.

![Residual glow curve](image)

Figure 3: A typical residual reading glow curve of a TL chip that was irradiated to 30 mSv, for readout of 30 seconds.

The nature of the lower additional peak was investigated and was found to be the result of the Teflon response to heat (in TLD cards used in the current work, the TL chips are held between two Teflon layer). The tail response is probably a result of IR emission or contributions from the deeper traps. As shown in table 1, both effects have a negligible affect on the reassessment factor.

Table 1: The percentages of the residual glow curve areas from the original readings, obtained by the 30 seconds reading after a previous 13.3 seconds reading (5).

<table>
<thead>
<tr>
<th>Dose [mSv]</th>
<th>Full glow curve</th>
<th>Main peak Region</th>
</tr>
</thead>
<tbody>
<tr>
<td>10</td>
<td>0.37±0.08</td>
<td>0.41±0.06</td>
</tr>
<tr>
<td>20</td>
<td>0.56±0.12</td>
<td>0.48±0.07</td>
</tr>
<tr>
<td>30</td>
<td>0.55±0.10</td>
<td>0.48±0.07</td>
</tr>
<tr>
<td>50</td>
<td>0.52±0.09</td>
<td>0.47±0.07</td>
</tr>
<tr>
<td>100</td>
<td>0.58±0.10</td>
<td>0.51±0.07</td>
</tr>
<tr>
<td>Average</td>
<td>0.51±0.09</td>
<td>0.47±0.04</td>
</tr>
</tbody>
</table>

As shown in table 2, the average residual dose for doses in the range of 10 to 100 mSv is about 0.5% with a standard deviation of about 10% and with a repeatability factor of 93.8% (4,5).
The PTTL Method

Another possible way to increase the yield of the residual glow curve is to irradiate the TL chip for 30 minutes to UV light and then to read it again for 30 seconds. The PTTL glow curve is a result of deep traps that are not depopulated by the standard readout.

A typical 30 seconds PTTL glow curve for a TL chip that was irradiated to 30 mSv is shown in figure 4. The shape of the PTTL glow curve is almost similar to the shape of the original first reading (2). It can be seen that all information is released in up to about 13 seconds, which indicates that there is no need for a longer heating period but due to the fact that the rest of the glow curve is very low, there is no significant difference between the reassessment factors of both regions (1).

![Graph showing PTTL glow curve](image)

Figure 3: A typical PTTL glow curve of a TL chip that was irradiated to 30 mSv. The residual dose was read for 30 seconds after a first normal reading (13.3 seconds) and an additional irradiation to u.v. for 30 minutes (1).

The "zero reading" of the PTTL process is the area under the residual glow curve of a TL chip that was not irradiated (0 mSv) and then irradiated with u.v. light for 30 minutes and read a residual readout. The "zero reading" value was measured and found to be equivalent to about 3.7% of the original reading after 10 mSv exposure. This measured value is significantly greater than for a residual reading without UV irradiation. The high background was mentioned also in other works (2), and is one of the main shortcomings of the PTTL method. As shown in table 2, the average values of the PTTL residual readings are about 1.5% from the original dose, with a standard deviation of about 10%.
Table 2: The percentage of the residual glow curve areas from the original glow curve area (reassessment factors), obtained by the "extended period" reading (30 seconds), after a previous 13.3 seconds reading and after PTTL (irradiated with a UV light source)\(^{(1)}\)

<table>
<thead>
<tr>
<th>Dose [mSv]</th>
<th>&quot;extended period&quot;</th>
<th>PTTL</th>
</tr>
</thead>
<tbody>
<tr>
<td>10</td>
<td>0.37 ± 0.08</td>
<td>1.43 ± 0.97</td>
</tr>
<tr>
<td>20</td>
<td>0.56 ± 0.12</td>
<td>1.68 ± 0.94</td>
</tr>
<tr>
<td>30</td>
<td>0.55 ± 0.10</td>
<td>1.77 ± 0.81</td>
</tr>
<tr>
<td>50</td>
<td>0.52 ± 0.09</td>
<td>1.53 ± 0.46</td>
</tr>
<tr>
<td>100</td>
<td>0.58 ± 0.10</td>
<td>1.57 ± 0.33</td>
</tr>
<tr>
<td>average</td>
<td>0.51 ± 0.09</td>
<td>1.57 ± 0.15</td>
</tr>
</tbody>
</table>

CONCLUSIONS
Reassessing a single dose by using extra readouts of TLD100 chips at any time after the original readout is possible, while taking into consideration the effects of background doses and fading of the TLD glow curve. Both "extended period" and PTTL reassessment methods are repeatable and accurate within an uncertainty range of about 20%.

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The PTTL Effect in Different TL Materials

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Radiation Protection and Safety, NRC-Negev, Beer-Sheva, Israel

INTRODUCTION

It's well known that one of the main drawbacks of the TL dosimetry, especially for routine personnel dosimetry, is the emptying of the traps during readout, resetting the dose information. However, some information is left as a residual dose. There are two ways to reassess the initial dose (especially in case of high doses): by measuring the residual dose through the second reading of the dosimeter and by photo transferred thermoluminescence (PTTL), which is thermoluminescence due to the UV exposure in a phosphor which had been earlier exposed to ionizing radiation and then partially annealed and readout, leaving some TL as a residual. There is another effect called PITL - the thermoluminescence induced by direct exposure to light (usually UV light), which defines the background.

LiF:Mg,Ti (TLD-100 Harshaw/STI) consists of LiF doped with approximately 170 mol ppm Mg and 10 mol ppm Ti. It has survived competition from other materials of higher sensitivity and superior signal-to-noise due to its seniority and to its many superior TL characteristics. LiF:Mg,Cu,P (marketed as TLD-100H, LiF:MCP and GR-200), is a thermoluminescent (TL) material having additional dosimetric characteristics. Its 60Co sensitivity is 20 - 40 times greater than that of LiF:Mg,Ti. This enables ultra-low dose measurements in the microgray dose range, far beyond the usual capability of TLD-100. The problematic over-response of TLD-100 does not exist in LiF:Mg,Cu,P. This phosphor is more "tissue equivalent" to low energy photons (<150 keV) than TLD-100. One main drawback of LiF:Mg,Cu,P is that readout above ~240°C leads to an irreversible loss in sensitivity and changes in the glow curve structure. There are several other TLD-s used in personnel dosimetry (as Panasonic manufactured Li2B4O7:Cu) and in environmental dosimetry (CaF2:Dy, CaF2:Tm, CaSO4:Tm, etc.).

One of the advantages of the TLD-100 phosphor is the possibility to reassess high and intermediate doses by the PTTL technique. There are controversial results concerning the PTTL of LiF:Mg,Cu,P. Some works claim a very low PTTL (about three orders of magnitude lower than for TLD-100), while others found a value higher than for TLD-100. According to Osval{9}, the PTTL of the LiF:Mg,Cu,P chips (both the GR-200 and MCP-N) is very low, but the PTTL measured with the GR-200 chips is higher than the PTTL for the TLD-100, the minimum reassessable dose being about 3 mGy. No detectable PTTL was measured in the MCP-N dosemeters. On the other hand, Driscoll{11} claims that the PTTL of the GR-200 chips is lower than for TLD-100.

In the present work, we refer to the UV sensitivity and PTTL of some TLD-s used in personnel and environmental dosimetry, to highlight the possibility of high doses reassessment.
THE PITL AND PTTL OF DIFFERENT TL PHOSPHORS

The PITL and PTTL of Harshaw chips and GR-200

The measurements of LiF: Mg,Ti, CaF₂:Dy and CaF₂:Tm dosimeters were performed with TLD-100, TLD-200 and TLD-300 hot pressed chips, annealed in air at 400°C/1 h, followed by a fast cooling to room temperature before irradiation and readout. The chips were irradiated by a UV lamp (λ = 254 nm) after gamma irradiation and readout. The optimal time of exposure to the UV lamp was found to be 15 minutes, similar to the results reported by Mason.

In table 1, the results of the PITL - intrinsic UV sensitivity (obtained after exposing un-irradiated chips to the UV lamp during 15 minutes) and the PTTL (obtained after irradiation to a gamma dose of 80 mGy, evaluation, exposing to UV during 15 minutes and reading again) are presented for several TLD materials.

Table 1: The mean UV sensitivity and the PTTL for several Harshaw/STI chips.

<table>
<thead>
<tr>
<th>Dosimeter</th>
<th>PITL (mGy)</th>
<th>PTTL (mGy)</th>
</tr>
</thead>
<tbody>
<tr>
<td>TLD-100</td>
<td>0.17 ± 0.05</td>
<td>4.00 ± 0.40</td>
</tr>
<tr>
<td>TLD-200</td>
<td>0.11 ± 0.03</td>
<td>0.12 ± 0.05</td>
</tr>
<tr>
<td>TLD-300</td>
<td>0.06 ± 0.01</td>
<td>0.08 ± 0.02</td>
</tr>
<tr>
<td>GR-200</td>
<td>No measurable signal</td>
<td>No measurable signal</td>
</tr>
</tbody>
</table>

A very low intrinsic UV sensitivity for TLD-200 and TLD-300 chips was observed, and similar results were measured for the PTTL. On the other hand, TLD-100 has a low intrinsic UV sensitivity too, but the PTTL is about 20 times greater (for a previous gamma irradiation of 80 mGy) – about 5% from the initial dose. The PTTL in TLD-100 is relative to the original dose and is indeed used for reassessing doses in personnel and environmental dosimetry, being greater than the residual dose by about one order of magnitude.

The irradiation of several GR-200 chips was performed to different doses of gamma rays in the range of 1 mGy to 100 mGy. The PTTL was measured after evaluation and exposure to an UV lamp (λ = 254 nm) during 15 minutes. No detectable PTTL was measured when using the routine heating profile and therefore the reassessment of high doses is questionable. The glow curve of the residual dose (second reading) and of the PTTL are presented in figure 1. Although the shape seems to be somewhat different, no measurable quantitative difference could be detected.
The glow curve of the residual dose (A) and the PTTL (B) for GR-200.

The PITL and PTTL of Panasonic phosphors
The measurements of the intrinsic UV sensitivity and the PTTL were performed using the Panasonic UD-802 badges, each of them containing two Li$_2$B$_4$O$_7$:Cu and two CaSO$_4$:Tm chips\(^{(5)}\). The cards were exposed first to a UV lamp (\( \lambda = 254 \text{ nm} \)) for different periods of time (to measure the intrinsic UV sensitivity), then irradiated with gamma rays, evaluated, exposed to the UV lamp for 10 minutes and evaluated again (to measure the PTTL). The PITL for both materials is presented in table 2, for different exposure times. The PTTL of CaSO$_4$:Tm is presented in table 3 for different photon doses in the range from 0 mGy to 100 mGy.

Table 2: The intrinsic UV sensitivity (PITL) of the two chips vs. UV exposure.

<table>
<thead>
<tr>
<th>UV Irradiation time (min)</th>
<th>Li$_2$B$_4$O$_7$:Cu (mGy)</th>
<th>CaSO$_4$:Tm (mGy)</th>
</tr>
</thead>
<tbody>
<tr>
<td>5</td>
<td>6.28 ± 0.43</td>
<td>0.03 ± 0.01</td>
</tr>
<tr>
<td>10</td>
<td>10.46 ± 0.53</td>
<td>0.05 ± 0.01</td>
</tr>
<tr>
<td>20</td>
<td>17.41 ± 0.99</td>
<td>0.10 ± 0.01</td>
</tr>
<tr>
<td>40</td>
<td>27.23 ± 1.70</td>
<td>0.13 ± 0.02</td>
</tr>
<tr>
<td>60</td>
<td>34.91 ± 1.76</td>
<td>0.16 ± 0.02</td>
</tr>
<tr>
<td>120</td>
<td>40.99 ± 4.04</td>
<td>0.25 ± 0.04</td>
</tr>
<tr>
<td>300</td>
<td>61.56 ± 4.19</td>
<td>0.49 ± 0.08</td>
</tr>
</tbody>
</table>
A high intrinsic UV sensitivity of Li$_2$B$_4$O$_7$:Cu was observed, in agreement with the results of Pradhan$^{(8)}$. The results for PTTL are within the standard deviation range of the PITL values, thus no measurable PTTL contribution could be detected. The intrinsic UV sensitivity of CaSO$_4$:Tm is much lower than for Li$_2$B$_4$O$_7$:Cu, but its PTTL is higher and has an almost constant value for the different gamma doses in the range up to 100 mGy, as seen in table 3, thus no use for reassessment is possible.

Table 3: The PTTL of CaSO$_4$:Tm for different photon doses.

<table>
<thead>
<tr>
<th>Gamma dose (mGy)</th>
<th>PTTL (mGy)</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>0.05 ± 0.01</td>
</tr>
<tr>
<td>5</td>
<td>0.43 ± 0.04</td>
</tr>
<tr>
<td>20</td>
<td>0.46 ± 0.03</td>
</tr>
<tr>
<td>50</td>
<td>0.51 ± 0.01</td>
</tr>
<tr>
<td>100</td>
<td>0.57 ± 0.03</td>
</tr>
</tbody>
</table>

CONCLUSIONS

A comparison of the UV sensitivity (PITL) and the PTTL of different thermoluminescent dosemeters used in routine personnel and environmental measurements, is presented. Most of the chips are not sensitive to the UV light, except the Li$_2$B$_4$O$_7$:Cu. The PTTL in TLD-100 is relative to the original dose and is useful for reassessment of high and intermediate doses, while the PTTL in CaSO$_4$:Tm was found to be almost constant for the range of gamma doses between 5 and 100 mGy. No measurable PTTL was detected for the GR-200 chips.

Further research is to be performed to try to reassess high doses in LiF:Mg,Cu,P by changing some physical parameters for the PTTL (e.g. enhanced temperature, different wavelength of the UV lamp, etc.) or by using other techniques like optical bleach or extended time readout.
REFERENCES


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Fast Neutron Dosimetry Using the Glow Peak Separation Method in CaF$_2$:Tm

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INTRODUCTION

ICRP publication 60\(^1\) recommends a decrease of the yearly limit for occupational exposure from 50 to 20 mSv/yr. The 20 mSv limit will not create difficulties in individual photon dosimetry if thermoluminescent or glass detectors are applied, because photon doses can easily be measured with sufficient accuracy already at the level of 20 - 30 μSv. If occupational doses are evaluated every month (which is obligatory in many countries), the individual neutron dosimeter will have to measure neutron doses in the range of about 100 μSv. Thus, the personnel neutron dosimetry is continuing to be one of the unsolved problems. No single detector or detection mechanism have the combination of energy response, sensitivity, orientation dependence characteristics and the range necessary to meet fully the needs of a personnel neutron dosimeter.

Several attempts have been made to measure the doses of fast neutrons in a mixed field of fast neutrons and gamma rays, by employing the method of glow peaks separation. Busuoli\(^2\), Endres\(^3\) and others tried to separate the fast neutron and the gamma dose using the height ratio of different glow peaks in LiF:Mg,Ti. The wide acceptance of thermoluminescence dosemeters (TLD's) for personnel dosimetry of photons and beta rays suggests that further efforts are expected in this direction.

THE TWO PEAKS METHOD USING CaF$_2$:Tm

The TL two peaks method for neutron-gamma discrimination using CaF$_2$:Tm\(^4,5\) has long been considered of significant potential in neutron-gamma dosimetry, and indeed it is an accepted and useful technique at the high dose levels encountered in clinical and accident dosimetry. The main glow peaks of the CaF$_2$:Tm are peak no. 2, peak no. 3 and peak no. 5. The first one is fading after a few days.

It's well known that the CaF$_2$:Tm is not tissue equivalent. In one of our previous research we measured the energy dependence of the CaF$_2$:Tm for low energy photons\(^6\). We have suggested to cover the CaF$_2$:Tm chips with a metallic filter of 0.2 mm Ta and 0.15 mm Pb, to flatten the energy response for low energy photons, according to the requirements of the American Standard for Environmental Dosimetry\(^7\).

The different LET sensitivity of peaks 3 and 5 enables in principle to extract the gamma and neutron doses in a mixed field. Loncol et al.\(^8,9\) analyzed the response of the TLD-300 dosimeters in heavy particle beams. In vivo dosimetry is recommended for treatment verification in radiation therapy, using neutron and 85 MeV proton beams, to analyze the effect of different radiation qualities on the dosimetric behavior of the detector irradiated in phantom. The ratio of heights of peak 3 and peak 5 decreases by 70% from the Co-60 reference radiation to the C-12 heavy-ion beam. This parameter was strongly correlated with the change of radiation quality.
Hoffman\textsuperscript{(10)} highlighted that the proper peak height analysis of the two main peaks of CaF\textsubscript{2}:Tm glow curves can separate the high and low LET content of a mixed radiation field. Therefore the distribution of the biological equivalent dose can be determined in a single radiation. TLD-300 dosemeters have been successfully employed in high LET radio therapeutic fields generated by fast neutron, negative pion and heavy ion beams. The method was extended to a pre-therapeutic proton field of 175MeV maximum energy.

The aim of this work is to check the possibility of using the two peaks method in CaF\textsubscript{2}:Tm chips, to evaluate the separate exposures of fast neutrons and gamma rays in a mixed field of these radiations, for intermediate and low doses.

**MATERIALS AND METHODS**

We used standard CaF\textsubscript{2}:Tm hot pressed chips (3mm X 3mm X 0.38mm). A pre-irradiation annealing of 400\textdegree{}C/1h was performed. The heating profile consists of a preheat to 50\textdegree{}C and a linear heating rate of 5\textdegree{}C/s up to 400\textdegree{}C, for a total time of about 77 s. Gamma irradiations were carried out using a $^{137}$Cs source and fast neutron irradiation were performed by using an AmBe source. The TLD chips were read by a manual 4500 Bicron/Harshaw reader.

**EXPERIMENTAL RESULTS**

**Determination of the optimal time between irradiation and reading.**

Four CaF\textsubscript{2}:Tm chips were irradiated by a $^{137}$Cs source to a dose of about 10 mSv (1 rem) and read after 5 different period of time (from 6.5h to 90h). The same four CaF\textsubscript{2}:Tm chips were irradiated by fast neutrons to a dose of about 7.5 mSv (0.75 rem) and read after 4 different periods of time (from 17h to 70h).

In figure 1 we present the ratio of the main peak heights (peak 3 / peak 5) irradiated by gamma rays and by fast neutrons, and read after different times after irradiation. The peak heights ratio remains constant as a function of time. The optimal reading time to be chosen is after the fading of peak 2 (about 3 days). The average result of the peak ratios for gamma rays is 2.36 ± 0.15 and for fast neutrons is 1.55 ± 0.07.
Figure 1. The ratio of the main peak heights irradiated by gamma rays and fast neutrons, as a function of the time between irradiation and readout.

In figure 2 we present the glow curves of the CaF$_2$:Tm irradiated by gamma rays and fast neutrons. It can be seen that the sensitivity of peak 3 is similar for the both irradiations and peak 5 is more sensitive for fast neutrons.

Figure 2. The glow curves of CaF$_2$:Tm irradiated by gamma rays and fast neutrons.
The peak height ratios for different doses of neutrons.
The four CaF$_2$:Tm chips were irradiated to neutrons to several different doses, in order to check the ratio of peak 3 to peak 5. In figure 3 we present the ratios peak3/peak 5 as a function of the fast neutron doses. It can be seen that the ratios of peak3/peak5 are constant for the dose range checked of up to about 8mSv.

![Graph showing ratio of peak3/peak5 as a function of gamma and fast neutron dose](image)

**Figure 3.** Irradiation of the CaF$_2$:Tm chips to different doses of fast neutrons.
CONCLUSIONS

The ratio between the two main glow peaks of the CaF$_2$:Tm chips was measured for gamma rays and for fast neutrons. There is a significant difference in the sensitivity of peaks 5 for the two kinds of irradiation (figure 2), while the sensitivity for peak 3 is similar.

Because of the overlapping between peak 2 and peak 3, the optimal time of reading is about 3 days after the irradiation, after the fading of peak 2.

We have checked the ratios of peak 3/peak 5 for different low and intermediate doses (between 0.3 and 7.5 mSv) and the results are constant. These results prove that the CaF$_2$:Tm dosimeter can be used for separate measurements of fast neutrons and gamma rays for intermediate and low doses.

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Assessment of the Uncertainty in Creatinine Normalization to 24-hour Urine in Uranium Measurement

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INTRODUCTION

Indirect monitoring of radionuclide intake is based on the determination of activity concentrations in body excreta, especially urine. The International Commission on Radiological Protection (ICRP) has developed biokinetic models for urine excretion after intake of radionuclides, which are based on 24-hour urine excretion volume(¹). Therefore, 24-hour urine samples provide the best basis for assessing intakes. However, there are circumstances where 24-hour urine samples cannot be obtained and urine samples are taken as spot samples. In such a case, a normalization procedure for the spot samples to simulated 24-hour should be carried out. There are several normalization procedures with various uncertainties. Studies have indicated that the uncertainty can be expressed in terms of the geometric standard deviation of the distribution (termed scattering factor SF) assuming that the distribution of the results can be approximated by a log-normal distribution. One method that is widely used to normalize spot samples to simulated 24-hour samples is by the 24-hours urine volume, which is taken as 1.6 liters per day for males, according to the ICRP Publication on "Reference Man"(³). However, since real 24-hours urine volumes have a broad distribution of values, the uncertainty of this method is large and its SF was estimated as 1.9 - 2.0⁽²⁾. A more accurate method for 24-hour normalization is by the creatinine content of the sample⁽⁵⁾. The simulated 24-hour sample is estimated by the following equation:

\[
U(ng/24h) = \frac{U_{spot}(ng/L) \times CRE(gr/24h)}{CRE_{spot}(g/L)}
\]

Where, CRE (the 24-hours creatinine excretion in the urine) is taken as 1.7 g/day for males according to the ICRP Publication on "Reference Man"(³). All the parameters in this expression have uncertainties associated with their value. The uncertainty of the concentration of the uranium is \(U_{spot}(ng/L)= 5\%\) and of the creatinine is \(CRE_{spot}(g/L)=2\%\) and is determined by the measurement process. The uncertainty of the value for the "standard" male creatinine excretion \(CRE(g/24h)\) is based on the assumption that the distribution of the content of creatinine in urine can be approximated by a log-normal distribution. The geometric standard of the deviation of the distribution (scattering factor SF) expresses this uncertainty. Not all of the components will make significant contribution to the combined uncertainty. Components that are less than one third of the largest value need not to be taken into account⁽⁶⁾. This paper presents an assessment of the uncertainty of standard men creatinine extraction, based on urine measurements taken from male workers at the NRCN. As the results show, the uncertainty of the standard males is the most significant value (SF=1.3) and this value determines the uncertainty of creatinine normalization to 24-hour urine in uranium measurement.
EXPERIMENTAL

Sample collection: Two or three 24-hours urine samples were collected from 133 male workers at the NRCN, giving a total of 319 samples. Three collection campaigns were undertaken, separated by several months (in winter, summer and autumn).

The volunteers were given clean 2-L sterile plastic vessels and were requested to collect all the urine excreted during a 24-hour period. The filled plastic vessels were returned to the laboratory. In cases where the urine sample from a single person was contained in more than one vessel the content of the vessels was mixed in a large beaker. The total volume of each sample was recorded and a sample was drawn for analysis of the creatinine concentration.

In additional measurements, two volunteers were provided with several plastic vessels and were asked to collect the urine from each voiding in a separate vessel and mark the time of the voiding. The urine volume of each voiding was recorded, the uranium and creatinine concentration were determined and the total amounts of urine, uranium and creatinine excreted during the 24-hours were summarized. 24-hr urine normalization by creatinine of each urine sample was performed.

Creatinine measurements: The creatinine concentration was determined by a kinetic–colorimetric method with a Cobas-Mira instrument (F. Hoffmann-LaRoche, Switzerland). The method is based on the fact that the initial rate of color formation after reaction with picric acid in sodium hydroxide, determined through absorption at 510 nm, is proportional to the creatinine level in the sample. Creatinine results are normally reported in units of mg/dL, and acceptable values are in the range of 50–300 mg/dL (0.5 – 3.0 g/L).

Uranium measurements: The uranium concentration was determined directly, without sample pretreatment, by flow injection inductively coupled plasma mass spectrometry (FIAS-ICPMS), performed on a Perkin-Elmer/Sciex (Canada) instrument, at the Geological Survey of Israel, as described in detail previously\(^7\). The results for uranium concentration are reported in units of ng/L.
RESULTS
The experimental data fit generally to a log normal distribution. The results for 119 samples of 24-hour urine collection in the third (autumn) campaign are shown in figure 1. The geometric mean and geometric standard deviation (GSD) of the creatinine contents in 24-hour collections in all campaigns are summarized in Table 1. As seen in Table 1, the uncertainty, in terms of the scattering factor, is 1.3 and is smaller than the 24-hour normalization by volume, which was estimated by a SF of 2.0.

![Graph](image)

Figure 1. The cumulative and frequency distribution of creatinine content on the 24-h collection

Table 1: The geometric mean and geometric standard deviation in the first, second and third 24-hour collection campaign.

<table>
<thead>
<tr>
<th></th>
<th>First Winter</th>
<th>Second Summer</th>
<th>Third Autumn</th>
</tr>
</thead>
<tbody>
<tr>
<td>Creatinine geometric mean (g/day)</td>
<td>1.43</td>
<td>1.41</td>
<td>1.3</td>
</tr>
<tr>
<td>Creatinine GSD</td>
<td>1.3</td>
<td>1.3</td>
<td>1.3</td>
</tr>
</tbody>
</table>

For one of the volunteers who provided spot samples (subject #1) the total daily diurnal creatinine excreted in 11 voiding was 1.98 g/day. The total amount of uranium in the pooled spot samples, representing a true 24-hour urine sample, was 92.3 ng. When each spot sample was normalized by creatinine, the deviation from the true value ranged between 5% and 31%. Two samples containing less than 0.50 g/L creatinine were omitted, according to recommendation in the literature. The detailed results for this case are presented in Table 2.
For subject #2, the total amount of uranium in the pooled spot samples, representing a true 24-hour urine sample was 1017 ng/24h. The total creatinine value was 1.86 g/day. Each voiding was normalized to creatinine. The deviation from the "true" value ranged between 2% and 30%. The results when the uranium content is normalized to the creatinine concentration indicate a good agreement with 24-hour uranium content (within 30%).

Table 2: The analysis results for each voiding during 24-hours of subject #1.

<table>
<thead>
<tr>
<th>Deviation from &quot;true&quot; value (92.3 ng/day) [%]</th>
<th>Uranium normalized to creatinine [ng/day]</th>
<th>Urine volume [ml]</th>
<th>Uranium Conc. [ng/l]</th>
<th>Time of voiding</th>
</tr>
</thead>
<tbody>
<tr>
<td>-27</td>
<td>67.4</td>
<td>125</td>
<td>84</td>
<td>23:00</td>
</tr>
<tr>
<td>-16</td>
<td>77.8</td>
<td>550</td>
<td>33.4</td>
<td>6:00</td>
</tr>
<tr>
<td>-31</td>
<td>63.8</td>
<td>150</td>
<td>53.7</td>
<td>8:00</td>
</tr>
<tr>
<td>-29</td>
<td>65.2</td>
<td>140</td>
<td>39.1</td>
<td>10:00</td>
</tr>
<tr>
<td>-10</td>
<td>83.4</td>
<td>365</td>
<td>25.5</td>
<td>13:00</td>
</tr>
<tr>
<td>-5</td>
<td>-</td>
<td>260</td>
<td>17.8</td>
<td>13:45</td>
</tr>
<tr>
<td>-12</td>
<td>81.3</td>
<td>190</td>
<td>41.6</td>
<td>17:30</td>
</tr>
<tr>
<td>-8</td>
<td>85.0</td>
<td>180</td>
<td>46</td>
<td>19:30</td>
</tr>
<tr>
<td>-14</td>
<td>79.5</td>
<td>180</td>
<td>26.2</td>
<td>21:00</td>
</tr>
<tr>
<td>-</td>
<td>-</td>
<td>300</td>
<td>18.4</td>
<td>22:00</td>
</tr>
</tbody>
</table>

CONCLUSION

The uncertainty in the total creatinine excretion of the standard male is the most significant term. This uncertainty can be given in terms of the geometric standard deviation of the measurements distribution (termed scattering factor SF) and was assessed in this paper to have a value of 1.3, that is much smaller that the SF value for volume normalization which was assessed in the literature, having a value of 2.0. It is recommended that normalization to creatinine should be performed when spot urine samples are taken, in order to reduce the uncertainty.
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The Distribution of Committed Effective Dose over Fifty Years after Acute Intake of Uranium

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INTRODUCTION
The International Basic Safety Standards for Protection against Ionizing Radiation\(^1\), approved by the International Atomic Energy Agency (IAEA) in 1996, has set dose limits to occupational exposure of workers. The dose limits were based on the recommended values set by the International Commission on Radiological Protection (ICRP) in Publication No. 60\(^2\). The limits apply to the sum of the doses from external exposure in the specified period and the 50-year committed dose from intakes in the same period.

This paper presents the distribution of the committed doses from intakes of uranium compounds over the 50 years after an acute intake. Dose distribution of the three default absorption types according to ICRP-66\(^3\) model (F, M and S) will be presented. These types are defined according to their absorption rates to the blood as:

- **Type F** - 100% absorption with a half-time of 10 minutes.
- **Type M** - 10% absorbed with a half-time of 10 minutes and 90% with a half-time of 140 days.
- **Type S** - 0.1% absorbed with a half-time of 10 minutes and 99.9% with a half-time of 7000 days.

Additionally, 50-year dose distribution of uranium absorption type that was measured at the Nuclear Research Center Negev (NRCN) will also be presented. The absorption type determines the committed effective dose due to its different retention time in the body.

RESULTS
Solubility tests of uranium aerosols samples taken from the metal production and machining facilities at the Nuclear Research Center Negev (NRCN) indicated that the sampled aerosols at the NRCN should be assigned to absorption type between M and S types. Figure 1 presents a comparison of the lung retention (as a fraction of the intake) between ICRP default types M and S and the NRCN absorption type. Type F was omitted from the figure since it is cleared practically immediately from the lungs.

Figures 2, 3, 4 and 5 present the 50-year distributions of the effective doses to the lungs, bone surface and red marrow (tissues that receive high doses from uranium inhalation) and the total (whole-body) effective dose for uranium absorption types F, M, S and the NRCN solubility type, respectively.
DISCUSSION

Uranium type F - Less than 14% of the total committed effective dose is received during the first year after intake. Even 50 years after intake there is some amount of dose received by the body (about 1% of the total effective dose).

This is explained by the fact that most of the inhaled type F uranium is quickly absorbed into the blood and transferred to various tissues (like bone surface and red marrow) where it is deposited for many years.

Uranium type M - More than 90% of the total committed effective dose is received during the first year. This is explained by the fact that most of the inhaled type M uranium is transported to the gastrointestinal tract from where it is excreted from the body. Therefore most of the committed dose is received by the lungs during the first year before the uranium is cleared by particle transport to the GI tract.

Uranium type S - Near 50% of the total committed effective dose is received during the first year. This is explained by the fact that absorption to blood is negligible and basically all the inhaled type S uranium is transported to the gastrointestinal tract from where it is excreted from the body. Most of the committed dose is received by the lungs but since the clearance from the lungs is slow, the dose is more distributed in time than type M.

Uranium NRCN type - More than 60% of the total committed effective dose is received during the first year. Nearly all the total committed effective dose is received during the first 5-6 years after intake. Most of the committed dose is received by the lungs.
Figure 2. 50-year distributions of the effective doses to the lungs, bone surface and red marrow and the total (whole-body) effective dose, for uranium absorption type F.

Figure 3. 50-year distributions of the effective doses to the lungs, bone surface and red marrow and the total (whole-body) effective dose, for uranium absorption type M.
Figure 4. 50-year distributions of the effective doses to the lungs, bone surface and red marrow and the total (whole-body) effective dose, for uranium absorption type S.

Figure 5. 50-year distributions of the effective doses to the lungs, bone surface and red marrow and the total (whole-body) effective dose, for uranium NRCN absorption type.
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Adaptation of the Scinti SPEC Multi-Channel Analyzer for Spectrometry of CZT Detectors

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INTRODUCTION
The use of CdZnTe (CZT) detector is becoming common for low-energy gamma spectroscopy at room temperature. The common electronics used for such detectors are solid-state detectors electronics such as for HPGe, with over-memory for large amount of channels. The CZT operating voltage is around 500 -1000 volts only, efficient from a few tens of keV up to hundreds of keV spectrometry only. The CZT 5 x 5 x 5 mm$^3$ detector energy resolution is 4% FWHM at 122 keV [1], hence the FWHM is of about 4.9 keV, and therefore a gain of 1.3 keV per channel would be sufficient for most of the applications. In order to have the maximum energy range, up to 665 keV, it will be adequate to use up to 512 channels only.

The ScintiSPEC [2], manufactured by ICX-Radiation™ (formerly Target), has been developed in order to provide "all in one" system for typical gamma radiation scintillators, such as NaI(Tl), BGO or CSI(Tl). The ScintiSPEC is designed to be connected using a USB input from a computer to the detector with a 14 pin diheptal socket suitable for the photomultiplier tube base. The ScintiSPEC provides a 0 – 1200 V high voltage power to the detector. A PHA software, winTMCApro-lab, is included with the ScintiSPEC, which controls the voltage on the detector, and provides the user with digital setup tools to set a variety of spectrum reading parameters, for example the channel group size, calibration and more. The software has the basic capabilities to handle, analyze and to save the scored spectra.

The aim of this study is to develop a direct connection device, which will make the ScintiSPEC system, or similar systems, operational for a CZT detector. The ScintiSPEC is planned to supply the high voltage power, and to receive the detector signal in order to score spectral data from the CZT detector.

RESULTS
The CZT internal preamplifier was connected to a +12 V DC from the back side of an ORTEC spectroscopy amplifier, type 673 [3]. Since a high gain is commonly necessary in order to read the signal, the signal output BNC from the CZT detector was connected to the amplifier with the following setup: coarse-gain = 100, and shaping time = 1 µsec. The amplifier output is connected to the ScintiSPEC through the newly developed connection device, as shown in Figure 1.

The high voltage bias to the detector was set up to +1000 V by using the digital dial in the software winTMCApro-lab.

Two calibration sources of Am-241 and Cs-137 were placed 10 cm from the CZT detector. The detector was operated and tallied a spectrum showing its typical response (shown in Figure 2).
Figure 1. The electronic device chart and its connections from detector to the scintiSPEC [2] socket.

Figure 2. The CZT spectrum collected from Am-241 and Cs-137 gamma sources.
CONCLUSIONS
The use of the newly developed adapter provided us with the flexibility to use the same MCA system for dual purpose, scintillators or CZT detectors.
The spectral results showed that the ScintiSPEC system connected using our adapter facilitated high performance of operating, reading and analyzing of the CZT detector.

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Continuous Tracking of RFID Tagged Radioactive Sources

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INTRODUCTION

The prevention of radiation hazards due to radioisotopes is one of the concerns of the Atomic Energy Agency (IAEA). In a series of international conferences held in the last five years\textsuperscript{(1)} this issue was discussed thoroughly. One of the conclusions was that strict management of radioactive sources is essential. The management of radioactive sources would help to prevent transference of radioactive materials to unauthorized personal. For this purpose, states should make a concerted effort to follow the principles of the Code of Conduct on the Security of Radioactive Sources\textsuperscript{(2)}. In this context, the identification of roles and responsibilities of governments, licensees and international organizations is vital\textsuperscript{(3)}. The referred activities are primarily related to control over radioactive sources and enhance the tracking ability of radiation sources\textsuperscript{(4)}.

In this paper, a proposed Radioactive Sources Tracking System is presented. This system facilitates real time monitoring capability of fixed and mobile radiation sources. The system provides the location of the source and indication whether the source is inside or outside the shielding container. The information about the sources location and condition can be used to coordinate a fast response in case of any attempt to steal or tamper with a source.

These goals are achieved by using GPS (Global Positioning System), RFID (Radio Frequency IDentification) and control and management software.

RFID TECHNOLOGY

In the recent years, automatic identification procedures (Auto-ID) have become popular in many service industries, manufacturing companies, purchasing, distribution and logistics. Automatic identification procedures exist to provide information about people, animals, goods and products in transit.

Barcode labels are inadequate in an increasing number of cases especially because a line-of-sight between the barcode and the reader is essential and it is not always the case with radioactive sources.

Smart cards, the most common form of electronic data-carrying device in everyday use are read via a contact field (telephone smart card, bank cards). It is far more flexible when there is no contact between the data-carrying device and its reader.

Radio Frequency IDentification utilizes data stored on an electronic chip. The information exchange between the data-carrying device and the reader is achieved without galvanic contacts, by using magnetic or electromagnetic fields instead. Due to the numerous advantages compared with other identification systems, the RFID systems beginning to conquer new markets. One example is the use of RFID smart cards as tickets for short-distance public transport.
DESCRIPTION OF THE PROPOSED SYSTEM

The traditional procedure, currently used, for tracking of radiation sources is based on written documentation and annually inspection. This procedure registers the source type, location and activity, without capability of real time tracking of the source location or the radiation levels. This system may be satisfactory for heavy shielded sources permanently located in hospitals and industrial irradiation facilities, but it is not sufficient for small or medium size mobile sources, usually transferred in ordinary vehicles.

The novel aspect of the proposed tracking system is development of continuous location monitoring of the radiation sources. The tracking is based on individual tagging of each source. Most of the basic components needed for the implementation of the system are already available in the market. The challenge is to develop a unique detection and transmission system, integrating all the components into an operational system.

Schematic description of the system is presented in Figure 1.

![Diagram of the proposed system](image)

Figure 1. The proposed system
The Radioactive Sources Tracking System is composed of three layers:
1. Detection and location layer: Radiation detector, RFID tag and GPS receiver.
2. Communication Layer: WIFI or Cellular communication device.
3. Control layer: Software and hardware for real time tracking of radiation sources.

The RFID tag, GPS and radiation monitor should be attached to the radiation source container. The GPS provides continuous information about the container's location. As long as the radiation levels, read by the monitor, are above a preset threshold level, the status is regarded as "OK". The status is toggled to "not OK" whenever the radiation source is removed from its container. A status signal is sent continuously to the control layer which displays the information in the control room. In the control room there is a GIS (Geographical Information System) based software that presents the location and status of each radiation source.

The status of the source is indicated by a green, yellow or red sign.
Green indicates that the source is in location.
Yellow indicates that the source is removed from its container by authorized personal.
Red indicates that the source is removed from its container without authorization.
An authorized removal should be coordinated with the staff at the control room that will update the control software.

**CONCLUSION**
Prevention of radiation hazards due to radioisotopes can be achieved by using the tracking system presented in this paper. The presented system provides a comprehensive solution to the management and tracking of radioactive sources.

**REFERENCES**

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Innovative Alpha Measurement Array for Radon Mobility Research

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INTRODUCTION

The design goal was to develop a high efficiency radon detection instrumentation to support a research on radon mobility through different type of soil as functions of density, porosity, soil moisture content and depths.

Radon is a colorless, odorless, tasteless, radioactive noble gas that generally lacks activity toward other chemical agents. Radon is a health hazard since exposure can cause lung cancer. Radon is the sixth frequent cause of cancer death overall.$^1$ The dominant source for indoor radon is soil gas emanations from soils and rocks.$^2$

Radon is formed by the decay of radium. Its most stable isotope is $^{222}\text{Rn}$ which has a half life of 3.8 days. Because radon is a gas, it can readily travel through several meters of permeable soils before decaying. The primary limiting factor for $^{222}\text{Rn}$ migration in the soil is its natural decay.

The decay of $^{222}\text{Rn}$ is followed by emission of alpha and gamma radiation. Due to the expected low radon concentrations during the future experiments, a detection array with a large area and a high efficiency was needed. The developed research equipment includes a large area alpha detector and a 2" * 2" NaI(Tl) Scintillator detector for the gamma radiation.

METHODS

Simplified schematic of the test array is shown in figure 1. A pipe is filled with soil, which its radon mobility properties are under test. Radon is released from the radon source in the bottom of the pipe. A detection head in the top of the pipe is used to measure the amount of the incoming radon.

For the alpha measurement array, in order to minimize cost, our preference was to use commercially available silicon solar cells. The theoretical efficiency of the solar cells is nearly 100% response to alpha particles that hit the cells surface. The alpha detection array consists of 100 silicon solar cells with an active area of 1cm$^2$ each. Every 5 solar cells are connected in parallel and installed on a separate Printed Circuit Board (PCB) - "strip". 20 solar cells PCB strips are attached to the detection head (see Figure 2). The detection head will be attached to the radon mobility measurement system.

A dedicated amplifier module was developed. The amplifier module consists of four amplification channels. Each channel includes a charge amplifier$^3$, a second amplification stage and a level discriminator with a TTL output. The module also utilize an OR gate that combines the output pulses of the four channels to a single TTL output.
MEASUREMENTS and RESULTS
In order to evaluate the ability of the detection array to count alpha particles, an alpha emission source was placed in front of the solar cells strip. The solar cells strip was attached to the dedicated amplifier. The outputs of the amplifier were measured by an oscilloscope to demonstrate the wave forms and a multi-channel analyzer to measure the overall counts and the statistics of signal amplitude and noise.

The measurements were carried with one to five cells connected in parallel on the strip and with different bias voltages applied to the cells. In each measurement, only one cell was exposed to the alpha particles.

Table 1 shows the wave forms and the energy spectrums measured with one and five cells connected in parallel on the strip and with a bias voltage of 3.5V applied to the cells.

As seen in the table, there is a great increase in noise while the number of cell changes from one cell to five cells.

The change from one cell to five cells also caused a 10% decrease in the average pulse amplitude.

<table>
<thead>
<tr>
<th>Number of Solar Cells per Amp</th>
<th>Wave Form</th>
<th>Energy Spectrum</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 Cell</td>
<td><img src="image1" alt="Graph" /></td>
<td><img src="image2" alt="Graph" /></td>
</tr>
<tr>
<td>Upper Trace – Analog output, 200 mV/div</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Lower Trace – TTL output, 5 V/div</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Time base - 10μs/div</td>
<td></td>
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</tbody>
</table>

| 5 Cells                      | ![Graph](image3) | ![Graph](image4) |
| Upper Trace – Analog output, 200 mV/div |
| Lower Trace – TTL output, 5 V/div |
| Time base - 10μs/div |
CONCLUSIONS
A high efficiency, large area and low cost radon measurement array was developed. The measurement array uses commercially available silicon solar cells. The array was tested successfully for detection and counting of alpha particles. The array is ready for tests with radon and then to the incoming radon mobility research.

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Cyclotron Target Monitoring During Bombardment for PET Isotope Production

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INTRODUCTION

The increasing use of radio-labeled pharmaceuticals in medicine has generated the need for radioisotope availability on a routine commercial basis. Cyclotrons are commonly used to produce short-lived positron-emitting isotopes. A growing number of small cyclotrons have been installed in hospitals. The materials produced in cyclotrons are used in PET (Positron Emitting Tomography) studies as research tools for observing physiological mechanisms and diagnostic procedures for numerous medical problems. Due to the short half-lives of positron-emitting isotopes, they must be produced as rapidly as possible and be available in the required quantity. Sometimes, due to problems in the cyclotron operation, the production run fails. Physicians and patients expect the radioisotopes to be available at a particular time. A failed run, in addition to delaying or even canceling the scheduled medical examination, is also extremely costly. Hence the need for automated control sensors to monitor the expensive production process arises. The need is even greater in PET commercial, production and distribution centers, and in high energy cyclotron centers that produce and distribute isotopes for Single Photon Emission Computed Tomography (SPECT), since numerous hospital clinics and patients rely on the prompt availability of the radiopharmaceuticals. Our goal was to develop an optimum radiation detection system to be used as a diagnostic tool for startup, maintenance and operational needs of the cyclotron facility.

METHOD

The existing procedure for investigating the bombardment during PET isotope production is measurement of the accumulated beam current. The particle energy, the beam particle rate (beam current), and the nuclear reaction cross-section determine the quantity of the radionuclide that can be produced at a given period of time. Although the production rate is linear to the beam current, measuring the beam current yields, at best, a crude estimate of the radionuclide production rate in the target where the radionuclides are produced. This estimation is obtained by extrapolation of the bombardment time and the beam current but ignores many actual factors that influence the production rate. Some of these parameters can be measured, but others are unknown such as: target condition (full or empty), percentage of $^{18}$O water enrichment (in the case of $^{18}$F production), changes in the beam profile and density of air bubbles in the target liquid. Therefore, the accumulated beam current is a useful tool to predict the produced activity in the target only so long as there are no malfunctions. For instance, the measured current value is the same whether a full or an empty target is bombarded; a run will proceed without any recognition whether the desired radionuclide and its required activity are satisfactorily produced. Understanding the present limitations of using the accumulated beam current as a means of evaluating bombardment, a computerized radiation detection system that measures the cyclotron target can offset these limitations. We have designed, constructed, evaluated and utilized such a system.
The computerized detection system monitored the radiation levels adjacent to the cyclotron target, during and after isotope bombardment, to show that a particular cyclotron run might be faulty. Of necessity, the system had to be situated adjacent to the cyclotron target where high radiation fields, gamma photons and neutrons, increase the system noise. The system utilized a BGO scintillation detector focused on measuring photons in the 511 keV energy window and Geiger Muller (GM) detectors. The system monitored radiation levels during the production run, as well as the isotope activity before and after its transfer procedure. The detector read-outs were collected on-line and results were transferred to the software. Those results were used by the cyclotron operator as part of the bombardment run overall diagnostics and quality control procedures. The designed detector configurations were divided into two main categories: a) Sensitive detectors aimed at monitoring the target with a high signal to noise ratio after the bombardment and b) Low sensitivity detectors aimed at monitoring the target with a wide measuring range during and after the bombardment, but with a low signal to noise ratio during the bombardment.

RESULTS

The system development required several experimental steps. Preliminary experiments were carried out to evaluate whether monitoring of the gamma radiation level would provide an indication of the run quality (see Figure 1). From these experiments, a measuring system previously designed for radiation safety monitoring was adjusted to suit the special needs of target monitoring. Interference with the operation system components, caused by the high radiation level, required separation of the main detector electronics from the sensor, to extend the detector life-time and at the same time to create a rugged detector for high radiation dose effects.

The data acquired by the measuring system were compared to both the accumulated beam current and the isotope measured activity, which was performed in the chemical laboratory upon its transfer. Results were evaluated with consideration to predicting the quantity of activity that would be produced and indicating any defects that might exist in a “bad” run during an early stage of the bombardment. The measurements were used to define the radiation curve profile during the bombardment. Upon testing the profile, a measurable difference (>20%) between the gamma radiation levels during a “normal run” and a “malfunctioning run” was detected (see Figure 3).
Malfunctions such as running on an empty target or an incorrect buildup of the target after cyclotron maintenance activity, were also detectable. Different radiation levels were also measured for different percentages of $^{16}$O enrichment of the target water used for $^{18}$F production. A highly reliable conversion factor was found between the measured radiation before delivery and the produced activity. Thus, the operator can be alerted about cyclotron malfunctions before the radiopharmaceutical would be transferred from the target to the chemical modules for further processing (see Figure 2).

![GM detectors accumulated dose vs. accumulated beam current](image1)

![Detector readings vs. Produced activity](image2)

**Fig 1** – Detector reading vs. beam current  
**Fig 2** – Detector reading vs. produced activity

![Scintillation Detector](image3)

**Fig. 3.** BGO detector reading vs. accumulated beam current results, while the detector measurements were taken 1 minute after the beam was turned off.
CONCLUSIONS

A measuring system for the PET isotope production process, consisting of a gamma radiation detector, was designed and successfully tested as a tool for measuring the production rate of radionuclides in the cyclotron target. The system provided a tool for early detection of target malfunction, which had not been previously available. This detector system increased the production process reliability by providing on-line alert to a variety of abnormal cyclotron working conditions, thus serving as an important new maintenance tool for cyclotron operations.

The results obtained next to the cyclotron target indicated that on-line measurement can provide an accurate estimation of the amount of activity produced in the target, thus becoming an important indicator of the process quality control. The quantity of isotope activity in the target can be displayed by appropriate calibration of the detector readings.

The results indicated that such measurements can serve as powerful tools for sensing various target conditions, which can be translated to early awareness of isotope production malfunctions and interventional correcting steps can be taken by the cyclotron operator. The following conclusions concerning the advantages of our design as a measuring system for maintenance and operational procedures were reached:

Maintenance

- The system can be used to evaluate the cyclotron performance after routine target maintenance, and to distinguish between productions with different water enrichment levels.
- The system provided a tool for identifying a malfunction source during the cyclotron run, either in the production itself or in the activity delivery stage.
- The target monitoring system, when integrated into the cyclotron control system, is an extremely useful tool, especially for the long production runs ($^{18}$F, $^{125}$I, $^{103}$Pd).

Operational

- The system can serve as a powerful tool for sensing various target conditions, and by giving early notification to the cyclotron operator that a particular run is malfunctioning, permits corrective measures to be taken.
- The ability to measure the activity prior to its transfer into the chemistry module enables verification of the amount of activity produced so that an educated decision can be made, such as whether additional bombardment is required.

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INTRODUCTION

There is growing demand for sensitive, small sized radiation detectors. A new Silicon Photomultiplier technology, often called "SiPM" or "silicon in Geiger mode" in the literature, was reviewed for its capability to be used as the light sensor in radiation detectors. The current radiation detectors, based on scintillation crystals, generally consist of photomultiplier tubes (PMT) or silicon photodiodes (PD) as the light sensor. The sensor accumulates the light produced during the crystal scintillations caused by the interacting radiation and produces an electrical signal relative to the energy deposited. This work investigates the new technology properties to be used as replacement for the current technology, and presents the experimental results obtained with the radiation detector consisting of a CsI(Tl) scintillation crystal coupled to a SiPM. These results are compared to the international regulations requirements.

The SiPM was inventoried by Dr. Dolgoshein (MePhI, Russia) in the beginning of this century and has been ranking a lot of interest for the past three years. Today this technology is being manufactured by many companies like Pulsar (Moscow), SensL (Ireland), Photonique (Switzerland), Hamamatsu (Japan), INFN (Italy) and Zecotek (Singapore). The concept of this technology is based on multi pixel semiconductor avalanche photodiodes, where pixels are joined together as a matrix on a common silicon substrate; and therefore combines the PMT high gain advantage together with the dimensions and low power consumption advantages of the silicon photodiode.

METHOD

The Silicon Photomultiplier (SiPM) is a multi pixel semiconductor photodiode (see Figure 1). Each pixel operates in limited Geiger mode with a gain of approximately $10^6$, with a reverse bias voltage of 10 to 20% higher more than the diode breakdown voltage, so each carrier is generated by light photons (signal) or thermally (noise), gives rise to a Geiger-type discharge. Each pixel dimension can vary from 20 to 100 μm, depending on the used mask, with a density of up to 1000 pixels per mm$^2$. The SiPM output signal is a sum of the signals from a number of pixels activated by the photons.

Figure 1 – A) Single pixel structure B) Schematic description C) SensL12x12mm diode
In applications where SiPMs are used as the light sensor (e.g. when coupled to scintillators), the obtained light output signal is directly proportional to the number of light photons, as long as their number is less than the number of SiPM pixels (dynamic rage). The pixels are connected in parallel, via integrated resistors, to a common load. Supply voltage (Vb) depends on the junction type and varies from 25V up to 70V. This voltage is 30 to 60 times lower than that required by a traditional photomultiplier tube (PMT). The pixel output signal is independent to the number of electrons which initiated the Geiger breakdown. Due to the parallel connection of all the pixels, the SiPM output signal is the sum signal of all the pixels that underwent a Geiger breakdown. However, although each pixel works in a digital mode (as a binary device), the SiPM is an analog device because all the pixels are read in parallel, generating signals with a dynamic range from a single photon to a maximum number of pixels.

The Geiger discharge is stopped when the voltage goes down below the breakdown value due to the resistor R on each pixel (typical R value is about 100-200 kΩ). This resistor also serves as a decoupling element between the individual pixels. Single pixel gain is determined by the charge accumulated in the pixel capacity (C\(_{\text{pixel}}\)), see Equation 1:

\[ Q_{\text{pixel}} = C_{\text{pixel}} \times (V_{\text{bias}} - V_{\text{breakdown}}) \]

Typically, \( C_{\text{pixel}} \) is approximately 50 fF, and \((V_{\text{bias}} - V_{\text{breakdown}})\) is only a few volts; therefore, each pixel charge \( Q_{\text{pixel}} \) is within the range of 0.1 pC. The SiPM gain is calculated according to Equation 2:

\[ \text{Gain} = Q_{\text{pixel}} / e, \]

Where \( e \) is the electron charge \( 1.6 \times 10^{-19} \), therefore the gain is about \( 10^6 \), i.e., the same order as in the PMT. The SiPM gain depends on the over voltage \( \Delta V = V_{\text{bias}} - V_{\text{breakdown}} \) and temperature. The typical values for gain variation are:

\[ dG/G \approx 7 \times dV_{\text{bias}} / V_{\text{bias}}, \text{ which gives } dG/G > 3\% \text{ for } dV_{\text{bias}}=0.1 V \]

\[ dG/G \approx 1.3 \times dT / T, \text{ which gives } dG/G=0.5\% \text{ for } dT=1^\circ C \text{ and } T=20^\circ C. \]


RESULTS

The tested SiPM has a 35 x 35 μm pixel size, with a total number of 3640 pixels in an area of 9 mm². The supply voltage (Vb) was 29.5 V according to the manufacturer recommendation in order to optimize the signal to noise ratio (SNR). A 3x3x15 mm CsI(Tl) scintillation crystal was coupled to the SiPM and the following parameters have been investigated:

1. Pulse shape and decay time noise vs. signal (Figure 2).
2. Detector linear reading for 137Cs source (Figure 3A).
3. Noise level vs. temperature (Figure 3B).
4. Detector resolution (Figure 4).

![Figure 2 - A) SiPM diode self noise and B) 137Cs source signal](image)

![Figure 3 - A) Linearity test measurements of 137Cs source and B) Noise level vs. temperature](image)

![Figure 4 - Detector resolution A) 137Cs source 7.5% and B) 241Am source 23%](image)
CONCLUSIONS
A new technology for the radiation detector light sensor was tested and found to be a promising approach. The SiPM main features and typical specifications can be summarized as:

1. Total quantum efficiency about 20%, similar to a traditional PMT.
2. High gain (G) similar to a PMT (~10^6) and high temperature and voltage stability.
3. Behavior independent from magnetic fields.
4. Low bias voltage.
5. Photon counting capability.
6. Dimensions enabling an extremely compact, light and robust mechanical design.
7. Low power consumption (≤50 μW/mm²).
8. Simple and relative low cost electronic circuitry.

The main difficulty in using PIN diodes as the light sensor was its high noise level of approximately 80 keV at 25°C. However, the noise threshold level measured with the detector consisting of 3x3mm SiPM diode coupled to 3x3x15mm CsI(Tl) scintillator, was only about 25 keV at room temperature and approximately 40 keV at 50°C. This low noise level enabled detection of 241Am in the wide energy range. The current SiMP limitations are its small size and the photon detection efficiency. For a detector consisting of a 3x3mm SiPM diode coupled to 10x10x30mm CsI(Tl) scintillator, where the diode covered only 10% of the crystal surface, the noise level increased to approximately 45 keV at room temperature. However, new large size SiPM diodes are under development and expected to be available in the market within few months. Another approach for the technical solution that are still under development is the increasing of the pixel size (50x50μm) and improving the Photon Detection Efficiency. The resolution measured for a detector consisting of a 3x3mm SiPM diode coupled to 3x3x15mm CsI(Tl) scintillator, was approximately 7.5 % for 137Cs and 24% for 241Am. This resolution is similar to the one measured with a detector consisting a PMT coupled to a CsI(Tl) scintillator.

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Radioactive Material Detection System (RMDS) Based on Advanced Spectroscopic Portal Technology

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INTRODUCTION
Since the attacks of September 11, 2001, combating terrorism has been one of the world’s highest priorities. As part of that effort, preventing radioactive material from being smuggled through the national borders—may be used by terrorists in a nuclear weapon or in a radiological dispersal device (a “dirty bomb”)—has become a key national security objective in many countries. During the last years (1991-2007) more than 20 incidents of Special Nuclear Material (SNM) seizures worldwide were registered and published in the media.¹⁻³ In order to supply an appropriate solution for the challenge of preventing such illicit nuclear trafficking, Totem Plus has developed the Radioactive Material Detection System (RMDS) in cooperation with Rotem Ind. which provides the testing according the standard requirements and the marketing services.

The RMDS is based on the so-called Advanced Spectroscopic Portal (ASP)⁴ concept which was mainly driven by the US Domestic Nuclear Detection Office (DNDO) in the last few years.⁵ This kind of Portal Monitor is capable of the isotopic identification of radioactive material within a vehicle or a container for the purpose of identifying vehicles carrying nonthreatening radioactive sources: Naturally Occurring Radioactive Material (NORM), radioisotopes utilized for medical and industrial purposes and those carrying the threatening Special Nuclear Materials (SNM). The characteristics of the RMDS have been tested with compliance to international standards⁶, and are described in this report.

Nowadays, the RMDS is being utilized as part of the Megaport Initiative (MPI) in the Haifa Port, Israel. The MPI is a worldwide effort to prevent nuclear smuggling in sea containers and is motivated mainly by the US Department of Energy (DOE). The performance of the system and the collective experience from other installation sites is being studied and compared to other methods, like hand-held instruments which are used for the same purpose.

METHODS
System description: The RMDS is designed to passively detect γ and neutron radiation from moving containers, trucks and cars, without disturbing the cargo flow. It is based on 3"×3" NaI(Tl) scintillation γ-detectors which are arranged inside each pillar in an optimized geometry for high sensitivity and efficiency (see Figure 1). The installed system at Haifa Port consists of 32 such NaI(Tl) γ-detectors (16 in each pillar) and a 2m long ⁴He tube with polypropylene moderator neutron detector in each pillar. Several other combinations of the number of NaI(Tl) γ-detectors can be used depending on the application (pedestrian, luggage or vehicle screening) and the requirements (static or dynamic screening).

A very important, part of the system is the state-of-the-art algorithm for isotope identification which is implemented in the RMDS user-friendly software package. Through this algorithm it is manageable to unite spectra from multiple NaI(Tl) detectors and calibrate them on the energy axis (Figure 2). A unique methodology is used to perform gain compensation due to the drift caused by ambient temperature variations.
The Data Acquisition electronics combined with the RMDS software emulates the performance of the Multi Channel Analyzer (MCA), which is essential for spectrum analyses. The data from all the 32 detectors is collected and processed by a PC equipped with a fast Analog to Digital input card.

Figure 2. Simultaneous measurements of $^{133}$Ba, $^{137}$Cs and $^{60}$Co at 47 °C measured for 5 sec. Stabilized combined spectra of 32 detectors.

**Characteristics and Performance:** Extensive testing of the RMDS has been performed in the last few months both internally and by American officials from several national laboratories operated by the DOE. These tests were performed in order to characterize the capabilities and the features of this advanced equipment. Partially, the tests were performed according to the ANSI N42.38.\(^{(6)}\)

Among the tests that were performed was ambient temperature test, relative humidity test, mechanical performance (vibration resistance) test, static and dynamic radionuclide identification including masking effect by Naturally Occurring Radioactive Material (NORM) and simultaneous identification of several isotopes, determination of false alarm rate, of full-width-half-maximum (FWHM) of combined spectra and determination of source location within the vehicle.
**RMDS Applications:** The RMDS has been designed as a flexible modular component for spectroscopic portal monitors. The modularity of the RMDS provides easy tailored solutions to the specific requirements of any facility and for the particular purposes it is intended to be used. As for now, the RMDS has been already installed in various detector combinations in the following sites: Haifa Port, Israel, as part of the Megaport Initiative; Valbruna, Italy, scrap yard facility; Akko, Israel, scrap yard facility; Ramat Hovav, Israel, hazardous material waste site; Sidney, Australia for the Australian customs (under evaluation).

The current Megaport concept worldwide is based upon Portal Monitors which are used for dynamic primary screening of vehicles and containers. These Portal Monitors are equipped with 4 $^3$He tubes for neutron detection and 4 plastic scintillators for gamma detection which do not give information about the nature of the alarm. In case of alarm in the portals, the vehicle/container is directed to a secondary station where hand-held Radiisotope Identifiers (RIIDs) are applied for secondary screening to localize and identify the radioactive material. This laborious and time consuming procedure is critical for the port’s vehicle and cargo flow, in order to identify vehicles causing false alarms, innocent alarms caused by NORM, Medical or Industrial radionuclides from those carrying threatening SNM.

The RMDS enables not only faster vehicle screening than using hand-held RIIDs but also more precise identification, as comparison shows. Another advantageous aspect of the RMDS is the determination of the location of the radioisotopes inside the vehicle without the need of scanning with hand-held equipment.

Figure 3. The main screen of the user-interface in the alarm mode.
Results
The various tests show the following characteristics of the RMDS: the FWHM for $^{137}$Cs has been measured to be 7.9% (combined for 32 detectors); the operational temperature range is from -30°C to +50°C (Table 1); relative humidity operational range is up to 93%RH; the false alarm rate stands around 1.6000.

<table>
<thead>
<tr>
<th>Temperature</th>
<th>30°C</th>
<th>20°C</th>
<th>0°C</th>
<th>5°C</th>
<th>22°C</th>
<th>40°C</th>
<th>50°C</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of identifications out of 10 trials</td>
<td>10</td>
<td>10</td>
<td>10</td>
<td>10</td>
<td>10</td>
<td>10</td>
<td>10</td>
</tr>
</tbody>
</table>

Table 1. $^{137}$Cs source identification over the temperature range.

The source location is determined with an accuracy of approximately ±70 cm.

Other detection and identification probability rates are presented in Table 2, including masking effects and simultaneous radionuclide identification.

Acceptance tests results for Megaports are in progress and will be available soon.

Table 2. Results of static and dynamic radionuclide identification tests.

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Simultaneous Radionuclide Identification Test Mode</th>
<th>Results</th>
<th>Detection</th>
<th>Identification Isotope</th>
<th>NORM</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fertilizer</td>
<td>Stationary (600s)</td>
<td>100%</td>
<td>n/a</td>
<td>100%</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Dynamic (container speed 8 km/h)</td>
<td>100%</td>
<td>n/a</td>
<td>100%</td>
<td></td>
</tr>
<tr>
<td>$^{57}$Co (20μCi)+ Fertilizer</td>
<td>Stationary (600s)</td>
<td>100%</td>
<td>100%</td>
<td>100%</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Dynamic (8km/h)</td>
<td>100%</td>
<td>90%</td>
<td>80%</td>
<td></td>
</tr>
<tr>
<td>$^{241}$Am (50μCi) + Fertilizer</td>
<td>Stationary (600s)</td>
<td>100%</td>
<td>100%</td>
<td>100%</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Dynamic (8km/h)</td>
<td>100%</td>
<td>90%</td>
<td>100%</td>
<td></td>
</tr>
<tr>
<td>Neutron Detection – $^{252}$ Cf (12,000 N/s) + Fertilizer</td>
<td>Dynamic (8km/h)</td>
<td>100%</td>
<td>n/a</td>
<td>90%</td>
<td></td>
</tr>
<tr>
<td>$^{137}$Cs (15μCi) + Fertilizer</td>
<td>Stationary (600s)</td>
<td>100%</td>
<td>100%</td>
<td>100%</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Dynamic (8km/h, with 3 cm steel shielding)</td>
<td>100%</td>
<td>80%</td>
<td>80%</td>
<td></td>
</tr>
<tr>
<td>Simultaneous Radionuclide Identification – $^{137}$Cs (15μCi)+ $^{57}$Co (20μCi)+ $^{241}$Am (50μCi)+ Fertilizer</td>
<td>Stationary (600s)</td>
<td>100%</td>
<td>100%</td>
<td>n/a</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Dynamic (8km/h)</td>
<td>100%</td>
<td>90%</td>
<td>90%</td>
<td></td>
</tr>
<tr>
<td>DU (4.5 Kg)</td>
<td>Stationary (600s)</td>
<td>100%</td>
<td>100%</td>
<td>n/a</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Dynamic (8km/h)</td>
<td>100%</td>
<td>100%</td>
<td>n/a</td>
<td></td>
</tr>
</tbody>
</table>
CONCLUSIONS
In conclusion, the technological advantages of the RMDS over the other methods have been shown. Although the statistics from the Haifa Port have not been released yet, it can be qualitatively stated that preliminary results show that the RMDS gives more precise identifications than the hand-held RIIDs. Its outstanding environmental stability, detection and identification abilities are definitely vital for this kind of equipment.
From the economical aspect, it should be emphasized that using the RMDS enables to lower false and innocent alarm rates and thus to reduce the intervention with the traffic flow at the site and reduce number of operators.

This results from the combination of the highly sensitive, with high granularity, NaI(Tl) scintillators (3” diameter and 3” length) used for gamma radiation, advanced neutron detection system (“He detectors), sophisticated isotope identification algorithms and gain stabilization. The system has also a very user-friendly software package which is easy to operate and to initialize. All this makes the RMDS the preferred solution for Radiological Security applications.

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Radionuclides in Air Particulates at Soreq Nuclear Research Centre, Yavne

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INTRODUCTION
Fallout measurements have been conducted worldwide since the early 1960's to track natural and anthropogenic radionuclides in the air, which were released from atmospheric nuclear weapons tests and reactors. As part of the safety monitoring, performed at Soreq Nuclear Research Centre in Yavne, the radionuclides concentration in air particulates are sampled and monitored. The concentration of natural and environmental (\(^{137}\)Cs) radionuclides is governed mainly by atmospheric conditions, such as temperature, wind and climate conditions \(^1,2\). The concentration of the natural and environmental radionuclides during the period of 2003 to 2006 is presented. Their correlation with climate conditions was investigated. No anthropogenic radionuclides were found, apart from environmental levels of \(^{137}\)Cs released at the Chernobyl accident.

MATERIALS AND METHODS
Monitoring of the air particulates is performed continuously since 2003 using a glass fiber filter type CAMFIL CS 5.0 \(^3\). This filter was found suitable because of its collecting efficiency after 3-4 days, its good performance at relatively high humidity conditions and relatively low natural radionuclide activity concentration. The filter media, of size 39.5 x 33 cm\(^2\), is mounted on a centrifugal pump, model PB 11/435, with a nominal rate of 1000 m\(^3\)/hr. The monitoring regime consist of: sampling during 3-4 days, waiting period of 24 hours for the decay of total activity in the filter, and a 24 hour measurement period by gamma spectrometry.

The gamma spectrometry system consists of a shielded High Purity Germanium detector (HPGe) with a relative efficiency of 35% (Canberra, USA) and a resolution of 2.05 keV, both at 1332.5 keV. The filter media is compressed by a hydraulic compressor applying a force of 10 tons to a homogeneous disc of 70mm diameter and 5.5mm height.

RESULTS
Cosmogenic gamma emitting radionuclides, produced by the interaction of cosmic radiation with the atmosphere, were measured. \(^{7}\)Be is produced in the upper troposphere and lower stratosphere as a result of spallation reactions of secondary cosmic ray neutrons on the nuclei of nitrogen and oxygen. The production rate of \(^{7}\)Be is thus a function of the solar activity or solar cycle (an 11 years cycle). A difference of up to 40% in \(^{7}\)Be concentration can be detected from maximum to minimum solar activity. The primary cosmic protons are deflected by the earth's magnetic field, causing the secondary neutron flux and the \(^{7}\)Be concentration to vary according to the geomagnetic latitude \(^4\). The cosmogenic \(^{7}\)Be was found to be normally distributed along the years 2003-2005, with an average concentration of 5 mBq/m\(^3\) [1-10 mBq/m\(^3\)], this is similar to other locations worldwide \(^1.3.6\). Traces of the low yield \(^{22}\)Na were sporadically recorded at a concentration lower than \(^{7}\)Be by five orders of magnitude.
The most common natural radionuclides found belong to the $^{224}$Ra decay chain; especially $^{212}$Pb and its decay products $^{212}$Bi and $^{208}$Tl. The $^{212}$Pb concentrations were found to be log-normal distributed with a mean activity concentration of 160 mBq/m$^3$ [$10^{-4}$-10$^4$ mBq/m$^3$]. Lower values of $^{210}$Pb and $^{40}$K were measured with a mean (log-normal) concentration of 0.46 mBq/m$^3$ [$10^{-3}$-10 mBq/m$^3$] and 20 μBq/m$^3$ [$10^{-1}$-$10^2$ μBq/m$^3$] respectively. The variations in time of cosmogenic and natural radionuclides are presented in figure 1.

The only anthropogenic radionuclide found was $^{137}$Cs having a log-normal distribution with an average of $10^{-3}$ mBq/m$^3$ [$10^{-4}$-$10^{-3}$ mBq/m$^3$].

![Graphs showing cosmogenic and natural radionuclide concentrations in the air at Soreq NRC](image)

**Figure 1.** Cosmogenic (left) and natural (right) radionuclide concentrations in the air at Soreq NRC

**DISCUSSION**

**The Correlation between $^{137}$Cs and $^{40}$K**

A strong correlation was found between $^{137}$Cs and $^{40}$K concentrations in surface air as seen in figure 2. This correlation indicates that a common transport mechanism, such as resuspension of soil, exists for both radionuclides. This correlation and the absence of irregularly high $^{137}$Cs concentrations during these years, emphasizes that the cause of the concentration variation of anthropogenic radionuclides is environmental.
**Figure 2.** $^{40}$K-$^{137}$Cs correlation (left) and the variation in time measured at Soreq NRC (right)

**Meteorological factors affecting radionuclides concentration in surface air**

The mean monthly atmospheric $^7$Be concentration varies by a factor of 2 during the year, with the highest values observed during the summer (6-8 mBq/m$^3$) and the lowest during the winter (3-5 mBq/m$^3$). The increase in concentration during the warm summer months is due to the increased rate of vertical transport of air within the troposphere as can be seen in figure 3. The seasonal variation in the tropopause height is the cause for the variation of $^7$Be concentration in the surface air as a function of the temperature, a phenomenon known as "spring leakage" which is noticeable in the measurements conducted at Soreq NRC$^{[7]}$.

**Figure 3.** $^7$Be concentration in surface air as a function of the average temperature
Relatively high $^{40}$K concentrations in surface air were measured at high temperature occurrences during periods when the wind direction was east to south (winds were coming from the east to the south region). Very high concentrations of $^{210}$Pb were measured during periods when the wind direction was mostly southerly (wind coming from the south) and high temperatures (around 36°C) were recorded. Very high temperatures combined with low wind speed caused a larger temperature gradient between the ground and the surface air, resulting in increased diffusion of $^{220}$Rn from the top layer of the soil. The hot, dry, and dusty south-wind originating from the deserts of Sinai and North Africa caused an increased amount of transported particles from distances of up to 600 km.

The $^{137}$Cs measured in Soreq NRC is from resuspended soil particles which originated in the Chernobyl accident. A qualitative correlation can be seen in figure 4 between the wind speed and the $^{137}$Cs concentration. As the wind speed increases, the resuspension process causes more soil particles to rise into the surface air. During spring (March-May) and autumn (September-November), seasons characterized by relatively strong winds, the concentrations of $^{137}$Cs were the highest.

![Figure 4. Average $^{137}$Cs concentration in surface air and wind speed during 2004.](image-url)
CONCLUSIONS
Radionuclides in air particulates are monitored at Soreq NRC. Natural and cosmogenic radionuclides and $^{137}\text{Cs}$ were found at environmental concentration levels similar to those found at several locations worldwide. No abnormal concentrations of anthropogenic radionuclides were found.
The linear correlation found between $^{40}\text{K}$ and $^{137}\text{Cs}$ concentrations in the measured years, indicates that a common transport mechanism exists for both radionuclides.
Atmospheric factors like temperature, wind, and climate conditions were found to strongly affect the concentration of the radionuclides in the surface air.

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1. INTRODUCTION

Naturally occurring radioactive elements, mainly radium isotopes at relatively high concentration were found in groundwater in the Negev and the Arava valley in the southern part of Israel [1,2,3]. Radium and uranium isotopes are transferred from the host rock into the aquifer by geochemical processes and they are found in the groundwater in four radium isotopes: $^{228}$Ra ($T_{1/2} = 5.75 \text{ y}$), $^{226}$Ra ($T_{1/2} = 1600 \text{ y}$), $^{222}$Ra ($T_{1/2} = 3.66 \text{ d}$) and $^{223}$Ra ($T_{1/2} = 11.435 \text{ d}$) and three uranium isotopes: $^{238}$U ($T_{1/2} = 4.47 \times 10^9 \text{ y}$), $^{234}$U ($T_{1/2} = 2.46 \times 10^3 \text{ y}$) and $^{235}$U ($T_{1/2} = 7.04 \times 10^8 \text{ y}$) [4].

High radium and uranium concentration may play a key role in the potential exploitation and utilization of groundwater for drinking and irrigation from the two main aquifers in the Negev and the Arava valley: The Nubian sandstone aquifer in the Negev and Arava Valley is part of the Lower Cretaceous of the Kurnub Group [5] and the carbonate aquifer in the Negev is part of the Upper Cretaceous Judea Group.

Groundwater samples from both aquifers were analyzed for radium and uranium isotopes and radon using independent methods. The radiological consequence of the use of the groundwater as drinking water is evaluated. A comparison of the radium and uranium measurement procedures used in the present survey is presented.

2. MATERIALS AND METHODS

Ninety groundwater samples from wells in the Negev and the Arava valley were sampled for radium, uranium and radon analysis. The samples were collected from active pumping wells in the Kurnub and Judea aquifers, initially preserved in a cold box and later transferred to a refrigerator in the laboratory.

Radium isotopes measurements were performed by: pre-concentration by evaporation followed by gamma spectrometry for $^{226}$Ra and $^{228}$Ra at Soreq, adsorption onto MnO$_2$ coated fibers followed by $^{222}$Rn measurement with the RAD7 system for $^{226}$Ra [6] at Ben Gurion University, $^{223}$Ra and $^{224}$Ra determination by delayed coincidence counting of the alpha particles emitted by $^{210}$Rn and $^{222}$Rn and their decay products [7] at Stanford University and by gamma spectrometry for simultaneous measurement of all four radium isotopes at Soreq. $^{234}$U and $^{238}$U in groundwater samples were measured by direct injection in a multi-collector inductively coupled plasma mass spectrometry (MC-ICPMS, Nu Instruments, Wrexham, UK) [8] at the Israel Geological Survey, and by selective extraction liquid scintillation counting (SE-LSC) at Soreq.

Samples were split into two: 1) 10 liters of water were acidified with HNO$_3$ to pH = 2 and sent to laboratory for radium and uranium analysis. 2) Large volume samples of 40-50 liters of groundwater were quantitatively transferred in-situ through 15 grams of acryl fibers coated with MnO$_2$ packed in columns at a low flow rate of 1.5 L/min. The MnO$_2$ fibers were further analysed in the laboratory for radium determination.
3. RESULTS
This Sandstone (Kurnub) aquifer is characterized by relatively high concentration of all radium isotopes especially those from the $^{232}$Th decay series. The results are log-normal distributed with an aquifer mean activity concentration were 0.29 Bq/l (<0.03 - 1.27 Bq/l) and 0.39 Bq/l (<0.02 - 4.61 Bq/l) for $^{226}$Ra and $^{224}$Ra respectively. The $^{224}$Ra/$^{226}$Ra ratio was found at disequilibrium (>1) in more than 70% of the wells in the aquifer (mean 1.57).

Isotopes from the $^{238}$U decay series were found at lower activities. The $^{226}$Ra log-normal mean activity concentration was 0.19 Bq/l (<0.03 - 0.86 Bq/l). The homologous $^{223}$Ra in the water was found also in excess to the expected natural $^{223}$Ra/$^{226}$Ra in solids.

The carbonate (Judea) aquifer is characterized by relatively high concentration of isotopes from the $^{238}$U decay series. The log-normal mean for $^{226}$Ra activity concentration resulted similar to the Kurnub aquifer being 0.24 Bq/l (<0.02 - 3.4 Bq/l). Large $^{222}$Rn/$^{226}$Ra ratios were measured.

Almost no activity concentration of radionuclides from the $^{232}$Th decay series were found in the carbonate aquifer.

Histograms of the $^{226}$Ra and $^{228}$Ra in both aquifers are presented in figure 1a and 1b.

Fig 1: Groundwater radium concentration in both aquifers a) $^{226}$Ra, b) $^{228}$Ra
The activity and concentration of uranium isotopes were determined and measured. In the Kurnub aquifer relatively low and varying concentration of $^{238}$U were measured in a wide range from 3 to 8835 ng/l. $^{234}$U was found in excess to the activity equilibrium ratio to $^{238}$U by a mean factor of 2.7 (1.1-5.25). The combined $^{234}$U+$^{238}$U log-normal mean activity concentration is 2.8 mBq/l.

In the Judea aquifer the range of $^{238}$U concentration in the aquifer groundwater is from 6 to 20885 ng/l with an activity $^{234}$U/$^{238}$U ratio exceeding the equilibrium value by a mean factor of 1.4 (1-3). The combined $^{234}$U+$^{238}$U log-normal mean activity concentration is 38 mBq/l.

Histograms of the activity concentration of the combined $^{234}$U + $^{238}$U and the $^{234}$U/$^{238}$U ratio at both aquifers are presented in figure 2a and 2b.

Fig 2: Combined uranium activity in groundwater from both aquifers a) Kurnub, b) Judea

4. DISCUSSION

4.1 Comparison of the Analytical Method

Four analytical methods for radium measurements were used throughout the survey: $^{226}$Ra and $^{226}$Ra by gamma spectrometry of a concentrated liquid solution ($\gamma$-spec), $^{226}$Ra adsorbed into MnO$_2$ coated fiber and measured by $^{222}$Rn monitoring (RAD7), gamma spectrometry of the MnO$_2$ coated fibers ($\gamma$-MnO$_2$) and measurement of $^{223}$Ra and $^{224}$Ra by delayed coincidence counting of radon isotopes (DCC).

A good correlation for $^{226}$Ra results was obtained by $\gamma$-spec and RAD7 techniques as well as for $^{226}$Ra and $^{228}$Ra between $\gamma$-spec and $\gamma$-MnO$_2$ techniques as can be seen in Fig. 3a and 3b respectively.

The low activity of $^{223}$Ra in the samples, the interference of gamma ray energies from $^{224}$Ra and $^{223}$Ra decay products and the short half-life of both radium isotopes are the main reasons for the weak correlation found between DCC and $\gamma$-MnO$_2$ techniques.
Fig 3: Method comparison, a) \(^{226}\text{Ra}\) with RAD7 and \(\gamma\)-spectrometry, b) \(^{226}\text{Ra}\&^{228}\text{Ra}\) measured by \(\gamma\)-spectrometry and \(\gamma\)-spectrometry of MnO\(_2\) fibers.

### 4.2 Radiological Consequences

Compliance of the groundwater with the updated Israeli drinking water regulations \(^{[9]}\) which are based on WHO recommendations of limiting the effective dose from drinking water to 0.1 mSv per year is calculated by the following expression:

\[
\frac{^{226}\text{Ra}}{0.5} + \frac{^{228}\text{Ra}}{0.2} + \frac{^{224}\text{Ra}}{2.1} + \frac{^{234}\text{U}}{2.8} + \frac{^{228}\text{U}}{3.0}
\]

Annual effective dose from drinking water in the studied area is presented as histograms in figures 4a and 4b. More than 80% of the groundwater samples from the Kurnub aquifer and 23% from the Judea aquifer do not comply with the regulations mainly due to high concentration of \(^{228}\text{Ra}\). Uranium isotopes do not represent a radiological issue, however 10% of the groundwater samples from the Judea aquifer are in excess of the WHO limit for toxicity to the kidneys (15 \(\mu\)g/l).

Fig 3: Potential effective dose from drinking water, a) Kurnub aquifer, b) Judea aquifer.
5. CONCLUSIONS
The first comprehensive survey of natural radioactivity in the groundwater of the southern part of Israel has been conducted. Radium and uranium isotopes were measured in groundwater samples from wells in the Nubian Sandstone (Kurnub) and the lower Cretaceous (Judea group) aquifers in the Negev and Arava valley.

The Judea aquifer is characterized by high $^{226}$Ra, $^{238}$U and $^{234}$U activity concentration while the Kurnub aquifer by high $^{228}$Ra, $^{224}$Ra and moderate $^{226}$Ra activity concentration.

A comparison of the measurement techniques used for radium isotopes was performed. Good correlation was found for $^{226}$Ra and $^{228}$Ra measurements by different procedures used in this survey. Lower limits of detection were achieved using the techniques based on radium adsorption in MnO$_2$ fibers. Good correlation was achieved for uranium isotopes using MC-ICPMS and SE-LSC.

More than 80% of the groundwater samples from the Kurnub aquifer do not comply with the drinking water regulations while 23% of the groundwater at the Judea aquifer do not.

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Sustainable use of Natural Water Sources Containing Ra Activity

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INTRODUCTION

Relatively elevated concentrations of naturally occurring radium isotopes are found in groundwater in the southern part of Israel in two main aquifers of the Negev and the Arava Valley: the Nubian Sandstone aquifer (Kurnub Group) and the Lower Cretaceous aquifer (Judea Group). Radium is transferred from the host rock into the aquifer by geochemical processes and it is commonly found in the groundwater as three isotopes: $^{228}$Ra ($T_{1/2}=5.75$y), $^{226}$Ra ($T_{1/2}=1600$y) and $^{224}$Ra ($T_{1/2}=3.66$d). The water in some of the wells in the Southern Arava contains radium at concentrations of 1-2 Bq·l$^{-1}$. These activity concentrations exceed the limits in drinking water, set by the Ministry of Health at 0.5 Bq·l$^{-1}$ and 0.2 Bq·l$^{-1}$ for $^{226}$Ra and $^{228}$Ra, respectively. These wells, which were planned to produce about 500 Mm$^{3}$·y$^{-1}$, are consequently not in use, causing a considerable loss, and experiments are under way to test the feasibility of using the water for irrigation\textsuperscript{(1)}. It was shown that Ra uptake by plants is mainly controlled by environmental conditions: soil solution activity, water availability and potential evapotranspiration\textsuperscript{(2)}. Radium distribution between the soil solution and the soil particles, as well as its mobility, is governed by the soil physical and chemical characteristics (i.e., mineralogical composition, particle distribution, organic matter content, etc.). Predicting Ra uptake by plants and environmental implications of irrigation with water containing relatively elevated Ra concentrations requires the development of a comprehensive model, describing the coupled processes of transport and sorption of radium in soil.

MATERIALS AND METHODS

$^{226}$Ra adsorption isotherms

Increasing concentrations of $^{226}$Ra (2 Bq/l to 2×10$^{5}$ Bq/l) were added to 30 g of oven-dried Arava sandy-loam soil in a total volume of 100 ml. Solutions were adjusted to a pH of 7.5 using 0.1 M NaHCO$_3$. The test tubes were shaken at 25°C for 24 hours or for 30 days. They were then centrifuged at 3,000 rpm for 1/2 hour and the supernatant was filtered. The equilibrium concentrations of Ra in the solution and in the solid phase were determined by direct gamma ray spectrometry. The low yield 186 keV gamma peak of $^{226}$Ra was used for its identification and quantification. The gamma spectrometry system consists of a shielded High Purity Germanium (HPGe) detector with a relative efficiency of 60% (Ortec, USA) and a resolution of 1.9 keV, both at 1332.5 keV. The adsorption isotherms were constructed by plotting the sorbed $^{226}$Ra activity (Bq·100g$^{-1}$) against its activity in the equilibrium solution (Bq·l$^{-1}$).
1038.4 g of oven-dried Arava sandy loam soil were packed in a PVC column (diameter and length of 5 and 20 cm, respectively). The bulk density was measured and the porosity was calculated accordingly, giving values of 1.71 g·cm⁻³ and 0.35, respectively. Before the displacement experiment, the soil column was slowly saturated by feeding from the bottom a background solution of distilled water adjusted to a pH of 7.5 by means of 0.1 M NaHCO₃. After an initial equilibration period of more than 10 pore volumes (PV), a concentration of 4545 Bq·l⁻¹ in the background solution (0.1 M NaHCO₃, pH 7.5) was injected into the soil column at an average pore-water velocity of 6.1 cm·h⁻¹. Effluent samples were collected from the bottom of the column and measured by alpha spectrometry Liquid Scintillation Counting (LSC). The identification and quantification of the ²²⁶Ra were done with its 4.78 MeV alpha particle. The automatic QUANTULUS 1220 LSC system (Wallac Oy, Finland) was used. The breakthrough curve (BTC) was then constructed by plotting relative concentration (effluent concentration divided by influent concentration) vs. dimensionless time (PV).

²²⁶Ra transport modelling

The BTC was analyzed by using a linear sorption non-equilibrium transport model, expressed as:

\[ \theta R \frac{\partial C}{\partial t} = \theta D \frac{\partial^2 C}{\partial x^2} - \theta v \frac{\partial C}{\partial x} - \rho_b \frac{\partial S}{\partial t} \]

\[ \frac{\partial S}{\partial t} = \alpha [K_d C - S] \]

\[ R = 1 + \frac{\rho_b K_d}{\theta} \]

Where \( \theta \) is the water content (dimensionless), \( R \) is the retardation factor (dimensionless), \( C \) is the solution concentration (Bq·cm⁻³), \( t \) is the time (h), \( D \) is the hydrodynamic dispersion coefficient (cm²·h⁻¹), \( x \) is the distance (cm), \( v \) is the average pore-water velocity (cm·h⁻¹), \( \rho_b \) is the bulk density (g·cm⁻³), \( S \) is the sorbed concentration (Bq·100 g⁻¹), \( \alpha \) is a first-order rate coefficient (h⁻¹) and \( K_d \) is the sorption distribution coefficient [cm³·g⁻¹]). The change of Ra sorption with time (\( \partial S/\partial t \)) was calculated assuming a linear isotherm.

Values of \( D, R \) and \( \alpha \) were estimated by using the STANMOD software. The kinetics of Ra distribution in the soil and in the drainage water was modeled by means of the HYDRUS 1-D software package. Spring and autumn growing of tomatoes was simulated for 15 years, under the southern Arava typical climatic conditions and irrigation water quality (1.6 Bq·l⁻¹ ²²⁶Ra) and with the fitted sorption and transport parameters. The soil hydraulic parameters were taken from another study. The seasonal irrigation and evapotranspiration amount to 610 and 530 mm, respectively. The soil is leached with 100 mm of water prior to every growing season.
RESULTS

Figure 1 illustrates the $^{226}$Ra sorption isotherms obtained for the two equilibration times. The $K_d$ value (24.2 l·100g$^{-1}$) obtained by equilibrating the soil with $^{226}$Ra solution for 30 days is significantly higher than its value when equilibrated for only 1 day (8.37 l·100g$^{-1}$), indicating that Ra sorption to soil particles is time dependent. Another study reports that the $K_d$ of $^{226}$Ra sorbed on bentonite and smectite is in the range of 10·10$^3$ l·100g$^{-1}$ and its sorption mechanism is dominated by ion exchange phenomena$^6$. The clay content of the soil used in the current work (11%) is much lower than in the other study and therefore a lower $K_d$ is expected.

![Figure 1. Linear sorption isotherms of $^{226}$Ra onto Arava sandy loam soil. $^{226}$Radium was equilibrated with $^{226}$Ra BTC is shown in Figure 2. Ra activity was detected in the effluent approximately after half a PV was collected. The relative activity of the effluent ($C/C_0$) rose to 5% at 1 PV and from there on Ra activity increased at a constant rate. A relative activity of 32% was reached after 9.5 PV. The fitted curve that was calculated with the STANMOD software, assuming linear sorption non-equilibrium miscible displacement, is in good agreement with the measured results.

![Figure 2. $^{226}$Ra breakthrough curve: Experimental data and optimized simulation for $^{226}$Ra transport in Arava sandy loam soil at an input concentration of 4545 Bq·l$^{-1}$.]
The values of the Ra BTC parameters calculated with STANMOD are presented in Table 1. The calculated $K_d$ is smaller than the value obtained from the two batch experiments. This can be explained by the short time period of the BTC measurements (approximately 6 h), compared with the two batch experiments. The fitted value of $D$ is small, suggesting that sorption phenomena tend to decrease both Ra hydrodynamic dispersion and diffusion. The fitted values of $D$, $R$ and $\alpha$ are also in good agreement with results obtained elsewhere for Cd sorption and miscible displacement measurements: 2.7-5.71 cm$^2$·h$^{-1}$, 21-33 and 0.015-0.048 h$^{-1}$, respectively.

Table 1. Parameter estimation obtained by the application of the linear non-equilibrium model to the $^{226}$Ra miscible displacement experiment at an input concentration of 4545 Bq·l$^{-1}$.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Units</th>
<th>Value</th>
<th>Lower</th>
<th>Upper</th>
</tr>
</thead>
<tbody>
<tr>
<td>$D$</td>
<td>cm$^2$·h$^{-1}$</td>
<td>2.09</td>
<td>-4.39</td>
<td>8.57</td>
</tr>
<tr>
<td>$R$</td>
<td>-</td>
<td>20.2</td>
<td>13.3</td>
<td>27.1</td>
</tr>
<tr>
<td>$K_d$</td>
<td>l·100gr$^{-1}$</td>
<td>3.61</td>
<td>2.12</td>
<td>5</td>
</tr>
<tr>
<td>$\alpha$</td>
<td>h$^{-1}$</td>
<td>0.05</td>
<td>0.015</td>
<td>0.085</td>
</tr>
</tbody>
</table>

Results of the simulation of 15 years of crop irrigation with water from the Shizafon well (1.6 Bq·l$^{-1}$) under the Southern Arava climatic conditions are presented in Figure 3. The Ra activity concentration in the root zone reaches a steady-state level of 1.65 Bq·l$^{-1}$ after 6 y, while its activity in the drainage water (below 100 cm) reaches a value of 1.5 Bq·l$^{-1}$ after 15 y. This simulation suggests that water uptake by plants irrigated with Shizafon water may enhance Ra retardation and hinder its mobility towards the water table.

Figure 3. HYDRUS 1-D simulation of $^{226}$Ra activity concentration in the rhizosphere solution (continuous curve) and in the drainage water at a depth of 100 cm (dashed curve) as a function of consecutive days (DAP).
CONCLUSIONS
The $^{226}$Ra mobility in a typical Arava sandy loam soil is hindered by its sorption to soil particles. Radium sorption was found to be time dependent. Therefore, we suggest that Ra has two sinks. The surface of the clay minerals is the first and the immediate one. The second sink is yet to be investigated.

Simulation of 15 years of crop irrigation with water from the Shizafon well yields low Ra activity concentration in the root zone, which may cause a minor, if not negligible, accumulation in edible tissues.

Further research should be focused on Ra-Ca competitive adsorption, Ra sorption under high ionic strength and lysimeter miscible displacement studies under non-saturated water content conditions.

REFERENCES


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Radiological Considerations in the Production of Lightweight Concrete Based on Coal Ashes

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INTRODUCTION

The common disadvantage of using high volumes of coal ashes (both bottom ash and fly ash) in lightweight concrete production is the presence in the ashes of trace amounts of heavy metals and natural radionuclides such as $^{226}$Ra, $^{232}$Th and $^{40}$K. Therefore, coal ashes are classified in many cases as a material with enhanced radioactivity. Accordingly, the use of coal ashes in lightweight concrete must comply with requirements of international recommendations and national legislations.

Over the past decade, a series of studies were performed at the Ariel University Center of Samaria (Ariel, Israel) in search for optimum solutions for the use of coal ashes (both fly and bottom ashes) in the lightweight concrete technology. These studies included investigations of scientific and engineering issues related to the radiological and engineering aspects of the utilization of coal ashes in the building industry, as described in this paper. Special attention was paid to a problem of producing ecologically-friendly concrete based on coal ashes.

Our product contributes significantly to the advancement of material recirculation by using in addition to fly ash, also bottom ash. This significant by product of coal combustion is used at present predominantly for construction beddings, structural fills, etc.

Utilization of fly ash for concrete production contributes to the reduction of the potential damage to the environment that can be caused by the accumulation of coal ashes in piles and ponds near power stations. Bonding the ash particles (in safe quantities) with the cement in concrete articles and structures reduces the potential exposure of humans to internal radiation by avoiding the penetration of fine ash particles to human internal organs.

In order to lower the radioactivity of concrete resulting from the use of coal ashes, we added to the concrete mixture calculated amounts of unprocessed crushed sand (UCS). This material is denser and is very low in radioactivity. UCS is a by-product of crushing limestone or dolomite in the process of crushed stone production at stone quarries.

The Ariel University Center of Samaria (Ariel, Israel) performed during the last decade a series of studies, searching for optimum ways for the utilization of coal ashes in the production of economically friendly lightweight concrete. These studies were related to a variety of scientific and engineering issues as described in this presentation.

EXPERIMENTAL METHODS

Studied fly ashes met the main requirements of Israel's Standard IS 1209. The sum of $\text{SiO}_2 + \text{Al}_2\text{O}_3 + \text{Fe}_2\text{O}_3$ was higher than 80% (the standard requirement is min. 70 %). The value of $\text{SO}_3$ was of 1.70 to 3.46% (the standard requirement is max. 5%). Loss of ignition of ashes from sources 1-3 was about 3%, from sources 4 and 5 about 9% (the standard requirement is max. 8%). Samples of fly ash had the following characteristics: relative specific gravity of fly ash - 2.25, Blain - 385 m²/kg.

The activity concentrations of the radionuclides $^{226}$Ra, $^{232}$Th and $^{40}$K measured in fly ash samples are presented in Table 1 (the radiological evaluation was carried out at the Soreq Nuclear Research Center).
The radioactivity of fly ash is different for the various sources of coal. However, the activity concentrations of the radionuclides and the range of their variations are similar to the data published in the technical literature. For example, we compared the data of Table 1 with the data of the activity concentration of $^{226}\text{Ra}$ in fly ash from sources in 10 European countries \(^5\). According to the statistical analysis of the data of this reference, the activity concentration of $^{226}\text{Ra}$ varied in the range of 158±62 Bq/kg (95% chance of occurrence).

Studied bottom ashes produced at Israeli power stations were characterized by low loose bulk density (of 560 to 730 kg/m\(^3\)). The grading of these bottom ashes complied usually to a lightweight aggregate.

The activity concentrations of the radionuclides $^{226}\text{Ra}$, $^{232}\text{Th}$ and $^{40}\text{K}$ measured in bottom ash samples are presented in Table 2. The radioactivity concentration values in bottom ash samples were also different for various sources of the coal used. Data of the radioactivity of bottom ash at the technical literature from different countries is deficient to compare obtained results with published earlier. Therefore, the selection of bottom ash used in the experiments was based only on the results of the radiological measurements.

Table 1. Activity concentrations of the radionuclides $^{226}\text{Ra}$, $^{232}\text{Th}$ and $^{40}\text{K}$ in fly ash samples

<table>
<thead>
<tr>
<th>Source of coal</th>
<th>Activity concentrations (Bq/kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>$^{226}\text{Ra}$</td>
</tr>
<tr>
<td>No 1 (South Africa)</td>
<td>151±15</td>
</tr>
<tr>
<td>No 2 (South Africa)</td>
<td>213±21</td>
</tr>
<tr>
<td>No 3 (South Africa)</td>
<td>200±17</td>
</tr>
<tr>
<td>No 4 (Columbia)</td>
<td>107±10</td>
</tr>
<tr>
<td>No 5 (Indonesia)</td>
<td>149±16</td>
</tr>
<tr>
<td>Average</td>
<td>164±16</td>
</tr>
</tbody>
</table>

Table 2. Activity concentrations of the radionuclides $^{226}\text{Ra}$, $^{232}\text{Th}$ and $^{40}\text{K}$ in bottom ash samples

<table>
<thead>
<tr>
<th>Source of coal</th>
<th>Activity concentrations (Bq/kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>$^{226}\text{Ra}$</td>
</tr>
<tr>
<td>No 1 (South Africa)</td>
<td>241±26</td>
</tr>
<tr>
<td>No 2 (South Africa)</td>
<td>103±14</td>
</tr>
<tr>
<td>No 3 (South Africa)</td>
<td>160±21</td>
</tr>
<tr>
<td>No 4 (Columbia)</td>
<td>39±6</td>
</tr>
<tr>
<td>No 5 (Indonesia)</td>
<td>204±23</td>
</tr>
<tr>
<td>Average</td>
<td>150±17</td>
</tr>
</tbody>
</table>

The samples of unprocessed crushed sand (UCS) were taken from an Israeli quarry exploiting dolomite deposits. The physical properties of UCS were as follows: relative specific gravity – 2.8; relative particle density (including pores) – 2.72; porosity – 3%; water absorption – 1.7%; loose bulk density – 1490 kg/m\(^3\); volume of voids (interspaces between particles) – 45.2%. The radiological characteristics of the UCS used were as follows (Bq/kg): $^{226}\text{Ra}$ - 17±1, $^{232}\text{Th}$ - 1±0.2 and $^{40}\text{K}$ - 12±4. The low activity concentrations of the radionuclides in UCS allow considering it as a diluting material.

Portland cement close to Israel's Standard IS 1 was used in all experiments.
The physical properties of the cement were as follows: relative specific gravity - 3.15; Blain - 370 m$^2$/kg; 28 day compressive strength of standard prism - 41.5 MPa. The radiological characteristic of the cement are follows: $^{226}$Ra - 46±4, $^{232}$Th - 19±0.1 and $^{40}$K - 94±11. The relatively high activity concentration of $^{226}$Ra in the cement is explained partly due to the use of fly ash as an additive during the cement production (5%), but main reason is the use of radioactive components (clay, gypsum, etc) in the clinker composition. Similar average results are also obtained by analysis of the data from reference (5) concerning the radioactivity of cement produced in 12 European countries ($^{226}$Ra - 51±39 Bq/kg). Hence, the activity of the cement contributes to a total radioactivity of the concrete and must be taken in account as an additional source of activity to the coal ashes.

The main object of the current experiments was to evaluate the influence of concrete constituents on the strength and the density, as well as the radiological characteristic, of the obtained lightweight concrete. A value of 330 kg/m$^3$ was selected for the cement content according to ASTM C330. This value is accepted by this standard for evaluating the given porous material (in our case bottom ash) as aggregate for lightweight concrete. The relation between the fly ash (FA) and the cement (C) was used according to our earlier study (6) as suitable for the conducted experiment: FA/(FA+C) = 0.5. The varying parameter of the relation between UCS and bottom ash (BA): UCS/(BA+UCS) was used. The effect of the ratio UCS/(BA+UCS) on the concrete properties was studied in a broad range: from 0 to 1.

RESULTS AND DISCUSSION

Fig. 1 compares the curves of concrete strength – versus concrete density in our experiments and plotted according to the requirements of ASTM C330. This standard allows evaluating the fitness of the given lightweight concrete to the requirements accepted for structural concrete. The analysis of the curves shows that the obtained concrete strength exceeds essentially the requirements of ASTM C330 for accepted values of the density (1680, 1760 and 1840 kg/m$^3$).

The activity concentrations of the radionuclides $^{226}$Ra, $^{232}$Th and $^{40}$K in the concrete with different ratios of UCS/(BA+UCS) were calculated. Values of the activity concentrations of the radionuclides $^{226}$Ra, $^{232}$Th and $^{40}$K were in range: 43-97 Bq/kg, 38-92 Bq/kg and 46-78 Bq/kg, respectively. The radiological evaluation of the concrete elements for the above-mentioned versions of the concrete proportions was conducted according to limitations set by international recommendations (7) and Israel's Standard IS 5098 (Nov. 2002). The following example of the field experimental production of thermal hollow masonry units based on the ternary lightweight concrete is considered below. For manufacturing the concrete mixture, bottom ash and fly ash from the source No 2 mentioned above were used.
The total content of coal ashes in the hardened concrete was 40 to 42%. UCS from the above-mentioned source was used as a diluting material. The technical characteristics of the obtained masonry units were as follows: dimensions 200x200x400 mm; density 845 to 875 kg/m³; strength (gross area) 3.4 to 4.6 MPa; radon emanation 0.5 to 0.7%; activity concentration index i 0.87 to 0.88 (on data of the Soreq Nuclear Research Center). Thus, the masonry units comply with the present building and radiological requirements.

CONCLUSION
A novel technology for the utilization of large-scale volumes of bottom ash and fly ash for manufacturing lightweight concrete was developed. This technology allows using both bottom ash and fly ash containing enhanced concentrations of the natural radionuclides $^{226}$Ra, $^{232}$Th and $^{40}$K, as well as bottom ash consisting of the weak particles. A ternary lightweight concrete based on bottom ash, fly ash and by-product of crushed stone production (unprocessed crushed sand) complies with the standards on lightweight concretes, as well as with standards and international radiation protection recommendations.

The experimental study of the suggested technology revealed the following potential of its application in practice:
1. Broadening the possibilities of the utilization of bottom ash and fly ash for building material production.
2. Advantages of the combined use of coal ashes with an additive of unprocessed crushed sand for the production of a broad range of thermal insulating/structural and structural concretes.
3. Reducing the potential damage to the environment caused by piles of unused/non-disposed coal ashes being accumulated near power stations and saving the natural resources for the production of building materials.

REFERENCES

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Determination of Natural Radioactivity and Gamma Radiation dose in Ceramic Materials Containing Fly Ash Developed in Israel

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INTRODUCTION

The ionizing radiation dose from building materials containing fly ash (FA) has been studied. Ceramic products, plates, blocks and bricks containing fly ash have been studied and compared to other construction materials for both natural occurring radioactive materials (NORM) content, radon exhalation rate and gamma radiation dose[1,2,3]. The utilization of fly ash (FA) as resources was studied for many years in different areas such as, ceramic products, in agriculture, paints, plastic industry, etc.[5].

Usually, ceramic blocks or blocks containing a mixture of clay and fly ash, are fired in a metallurgical furnace at selected temperatures of 750° C - 950° C. In order to preserve the environment, from large quantities of FA and BA (bottom ash) produced during the combustion of coal for energy production, the application of this pollutant (FA) was experimentally investigated by utilizing it in manufacturing of ceramic products recently developed in Israel as building materials.

During the last years there has been an increased awareness of the radiological health hazards due to ionizing radiation emitted by building materials including those containing fly ash. Much research has been carried out in order to determine the correlation between the concentration of radioactive materials in building materials and the external radiation doses to which the general public is exposed[6].

In order to limit the exposure of the public to natural sources of radiation there is a requirement to limit the concentration of NORM in building materials. International recommendations are described in publication 115 of the safety series issued by the IAEA[7] and the ICRP[8]. In Israel an approach based on considering the NORM and radon from building materials as an intervention issue was implemented. Accordingly, the total annual dose from building materials was set to 0.7 mSv. This is implemented through the proposed Israeli standard 5098 issued in 2007 [9].

The aims of this work were:

a. To determine the radon exhalation rate from ceramic blocks containing FA and BA and to compare it to that of other common construction materials.

b. To determine the radionuclides content in these ceramic products.

c. To measure and estimate the total annual dose equivalent external gamma radiation dose from ceramic products.

d. To check whether the ceramic block complies with IS 5098.
MATERIALS AND METHODS
Determination of radionuclides concentration in the samples was performed by high-resolution gamma-ray spectrometry. The measured material was grounded and milled to 14 mesh (1400 μm), overnight dried and packed in 1 L Marinelly beakers measured on an HPGe detector with 42% relative efficiency at the 1332 keV gamma-ray of $^{60}$Co. The measurement of the gamma radiation dose was performed using a plastic scintillator model Szintomat (Automess Ltd.) For determination of radon exhalation rate, both alpha and gamma ray spectrometry were applied. The method of alpha spectrometry developed by Keller et al. is based on measurements of the electrostatic deposition of positively charged ions (produced by stripping effect) of radon decay products ($^{216,218}$Po) onto a metallic surface barrier detector. The gamma spectrometry method is based on inserting the sample together with a calibrated charcoal canister, which collects the emanated radon, into a hermetically sealed chamber (with a volume of 175 L).

CALCULATION OF RADIATION INDEX IN CONSTRUCTION MATERIALS
The total radiation dose from building materials due to NORM elements are usually represented by the radiation index- $I$. The definition of the radiation index in the European community is $^{[11]}$

$$I = \frac{[^{40}K]}{3500} + \frac{[^{226}Ra]}{150} + \frac{[^{232}Th]}{185}$$

where: $^{40}K$, $^{226}Ra$ and $^{232}Th$ represent the activity concentration in Bq/kg, respectively. The European Community demands that the radiation index for building materials should be less than 1.

In the proposed Israeli approach (IS 5098) issued in 2007, the radiation index is calculated via similar equations who take into consideration the dimensions and density of the building material and the radon emanation coefficient. The radiation index – $I$, which in order to comply with the IS 5098 must be lower than 1, is calculated by the following equation:

$$I = \left( \frac{[^{40}K]}{A([^{40}K])} \right) + \left( 1 - e \right) \frac{[^{226}Ra]}{A([^{226}Ra])} + \frac{[^{232}Th]}{A([^{232}Th])} + e \frac{[^{226}Ra]}{A([^{222}Rn])}$$

where: $[^{40}K]$, $[^{226}Ra]$ and $[^{232}Th]$ are the activity concentration (Bq/kg) of the respective radionuclide in the building material. $A([^{40}K])$, $A([^{226}Ra])$ and $A([^{232}Th])$ are the radiation index coefficients of the respective radionuclide taken from the table amended to the IS 5098 and $e$ represent the radon emanation coefficient of the building material.

MEASUREMENT OF GAMMA-RADIATION DOSE DUE TO CERAMIC BLOCKS
Various ceramic blocks containing fly ash were measured with respect to their radioactivity. The effect of a whole wall was studied and a wall ($2.6 \times 2.3$ m$^2$) of ceramic blocks was built for this measurement. The ceramic material contains 45% FA and each block was of dimensions 25.5 x 20.5 x 10 cm$^3$. The density of these blocks was 1180 kg/m$^3$. The wall was built on the floor of an open space (200m$^2$) on the sixth floor. Before constructing this wall the total background radiation in this open space (due to cosmic and terrestrial radiation) was measured. 150 measurements were carried out, leading to an average background radiation dose of 47±6 nGy/h.

The gamma radiation dose from this wall measured for various distances from the wall is shown in Fig. 1.
Fig. 1. Radiation dose as a function of the distance from the wall of ceramic blocks

RESULTS AND DISCUSSION
Table 1, summarizes the results of the radionuclide concentration and exhalation rate for various building materials.

<table>
<thead>
<tr>
<th>Block (Type)</th>
<th>Bulk density (kg/m³)</th>
<th>$^{226}$Ra [Bq/kg]</th>
<th>$^{232}$Th [Bq/kg]</th>
<th>$^{40}$K [Bq/kg]</th>
<th>$^{222}$Rn exhalation rate [mBq/m² sec]</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ceramic with FA</td>
<td>1200</td>
<td>78.5±3.9</td>
<td>87.2±4.4</td>
<td>365.4±25.6</td>
<td>*0.013±0.001</td>
</tr>
<tr>
<td>Ceramic with FA</td>
<td>1200</td>
<td>78.5±3.9</td>
<td>87.2±4.4</td>
<td>365.4±25.6</td>
<td>**0.018±0.001</td>
</tr>
<tr>
<td>Ceramic with FA</td>
<td>800</td>
<td>78.5±3.9</td>
<td>87.2±4.4</td>
<td>365.4±25.6</td>
<td>**0.016±0.017</td>
</tr>
<tr>
<td>Light-weight concrete</td>
<td>1100</td>
<td>31.2±1.6</td>
<td>3.3±0.2</td>
<td>49.9±3.5</td>
<td>**0.09±0.008</td>
</tr>
<tr>
<td>Normal-weight concrete</td>
<td>2400</td>
<td>53.4±4.1</td>
<td>15.1±0.9</td>
<td>72.3±9.3</td>
<td>@0.24±0.01</td>
</tr>
<tr>
<td>Light-weight aggregate</td>
<td>660</td>
<td>52.8±2.6</td>
<td>44.8±2.2</td>
<td>842.0±58.9</td>
<td>**0.056±0.005</td>
</tr>
<tr>
<td>Ceramic with FA</td>
<td>800</td>
<td>78.5±3.9</td>
<td>87.2±4.4</td>
<td>365.4±25.6</td>
<td>#e = 1.3±0.2%</td>
</tr>
</tbody>
</table>

* Measured by alpha spectrometry
** Measured by gamma-ray spectrometry
@ Measured by continuous radon monitoring
# e - emanation coefficient measured according to IS 5098
FA-Fly ash
The radon emanation coefficient of the ceramic block was found to be \( 1.3 \pm 0.2\% \). The radiation index of the building material with density of 800 kg/m\(^3\) is calculated according to the IS 5098 procedure:

\[
I = \left( \frac{40K}{3974} \right) + (1 - 0.013) \left( \frac{226Ra}{377} \right) + \left( \frac{232Th}{278} \right) + 0.013 \left( \frac{226Ra}{13} \right) = 0.69
\]

The maximum annual effective dose in the center of a standard room (3 x 3 x 2.7 m\(^3\)) built with 6 sides of the ceramic blocks (25.5 x 20.5 x 10 cm\(^3\)) is calculated to be 0.48 mSv.

Since ceramic blocks are intended to be used only for exterior walls in buildings, only two walls will be built in a standard room and therefore, the annual effective dose expected in the center of the room will be only a third from the total effective dose leading to 0.16 mSv.

The measured annual radiation dose in the center of the standard room due to gamma radiation is estimated to be about 350 µGy/y (effective dose of 0.35 mSv/y), which is in good agreement to the calculated according to the IS 5098 approach.

Ceramic blocks with fly ash has been found to comply with IS 5098 and contribute to the effective dose similarly like other common building materials as concrete, light weight aggregate blocks, etc..

REFERENCES


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