Health Physics Society Midyear Meeting

Radiation Measurements

2011 Topical Meeting of:
Health Physics Society
(The Forty-Fourth Midyear Topical Meeting of the Health Physics Society)
American Academy of Health Physics

Sunday, 6 February -
Wednesday, 9 February 2011

Final Program

Charleston, South Carolina
Charleston Convention Center

Health Physics Society Committee Meetings

All Committee Meetings are in the Convention Center unless noted with (ES) for Embassy Suites

Saturday, February 5, 2011

FINANCE COMMITTEE
8:00 - 10:30 am *Executive Board Room (ES)*

ABHP PART II PANEL WORKSHOP
8:00 am - 5:00 pm *Club North*

WEB OPERATIONS COMMITTEE
9:00 am - Noon *Edisto (ES)*

HPS EXECUTIVE COMMITTEE
Noon - 5:00 pm *Executive Board Room (ES)*

Sunday, February 6, 2011

AAHP EXECUTIVE COMMITTEE
8:00 am - 5:00 pm *Meeting Room 13*

ABHP PART II PANEL WORKSHOP
8:00 am - 5:00 pm *Club North*

HPS BOARD OF DIRECTORS
8:00 am - 5:00 pm *Meeting Room 12*

Monday, February 7, 2011

GOVERNMENT & SOCIETY RELATIONS COMMITTEE
Noon - 1:30 pm *Meeting Room 1*

INTERNATIONAL COLLABORATION COMMITTEE
Noon - 1:30 pm *Meeting Room 3*

N13.3 DOSIMETRY FOR CRITICALITY ACCIDENTS
1:00 - 5:00 pm *Meeting Room 4*

Tuesday, February 8, 2011

SCIENTIFIC AND PUBLIC ISSUES COMMITTEE
9:00 - 11:00 am *Meeting Room 4*

HISTORY COMMITTEE MEETING
11:30 am - 1:00 pm *Meeting Room 1*

ANSI 32.3
1:00 - 5:00 pm *Meeting Room 4*

HOMELAND SECURITY COMMITTEE
4:30 - 6:00 pm *Meeting Room 1*

Wednesday, February 9, 2011

ANSI/HPS N42.54
1:00 - 5:00 pm *Meeting Room 1*

LAPC
10:00 am - 12:30 pm *Meeting Room 3*

LAAC
11:30 am - 2:00 pm *Meeting Room 3*

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Registration Hours
Exhibit Hall Foyer

Sunday, 6 February	3:30-6:30 PM
Monday, 7 February	7:30 AM-3:00 PM
Tuesday, 8 February	8:00 AM-3:00 PM
Wednesday, 9 February	8:00 AM-Noon

Exhibit Hours
Hall A

Monday	5:00-6:00 PM	Opening Reception
Tuesday	9:30 AM-5:00 PM	Exhibits Open
	9:45-10:30 AM	Refreshment Breaks
	Noon	Lunch-Exhibit Hall
	3:15-3:45 PM	Refreshment Breaks
Wednesday	9:30 AM-Noon	Exhibits Open
	10:00-10:30 AM	Refreshment Breaks

Speaker Ready Room
Meeting Room 5

Sunday	1:00-5:00 PM
Monday & Tuesday	8:00 AM-Noon; 1:15-5:00 PM
Wednesday	8:00-11:00 AM

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SHUTTLE SCHEDULE

Sunday, 6 February – Tuesday, 8 February
2- 54 Person Passenger Coaches

Convention Ctr to Downtown	Downtown to Convention Ctr
6:00 PM	6:20 PM
6:45 PM	7:05 PM
7:30 PM	7:50 PM
8:30 PM	8:50 PM
9:15 PM	9:35 PM
10:00 PM	10:20 PM

Tours…..Events….. Tours…..Events…..

Sunday 6 February
Welcome Reception
Ballroom C1/C2, 6:00-8:00 pm

Plan on stopping in for the HPS Welcome Reception. There will be an opportunity to meet friends and to start your evening in Charleston. Cash bar and light snacks will be available.

Monday 7 February
Exhibitor Opening Reception
Hall A, 5:00-6:00 pm

Join the Exhibitors for food, a cash bar, and the latest in Health Physics equipment.

Tuesday 8 February
Complimentary Lunch in Exhibit Hall
Hall A, Noon-1:00 pm

Technical Tour of GEL Laboratories, LLC
5:30-8:30 PM, Includes Dinner
Onsite $25

GEL Laboratories, LLC (GEL) is conveniently located less than 5 miles from the North Charleston Convention Center. The tour will start with an overview of our full service mixed waste laboratory. The tour routes along the sample flow process beginning at sample receiving. The tour includes radiobioassay, analytical chemistry, radiochemistry laboratories and waste management facilities. Experts from various sections of the laboratory will be available to answer specific questions during the tour.

This outing will be valuable to Health Physics Members who rely on analytical data for making decisions on human health and the environment. A greater appreciation of the laboratory process will enhance their future interactions with radiological laboratories. There is a 100 person limit for this tour.

Job Openings/Resumes

Post your printed job opening or resume on the "Job Boards" in the 100 aisle of the Exhibit Hall

CHARLESTON, SOUTH CAROLINA ON YOUR OWN

Fort Sumter - This brick fort at the entrance to Charleston's harbor was the site of the opening shots of the American Civil War. Rebels fired upon the vastly outnumbered Union garrison for over 40 hours when they were forced to surrender due to raging fires. Union attacks throughout the war battered the fort, but were never able to force its surrender. Tour boats take visitors out to the fort from either a dock in downtown Charleston or from Patriots Point. Cannonballs can still be seen in the brick walls of the fort.

Fort Moultrie - This brick fort on Sullivan's Island is also preserved by the National Park Service. The fort stands on the site of the palmetto log fort that held off a British attack on the city in June 1776. The logs and sand absorbed the British cannonfire with little effect. This is the origin of the state's nickname of "Palmetto State" and also the Palmetto tree on the state flag. The current fort is the third on the site and was fully operational from 1809 through World War II. It has been restored showing its appearance through various technological stages. The fort was also the duty post of a young Edgar Allen Poe in 1828 and was where he wrote "The Gold Bug" and many other stories.

Carriage Rides - A trip to Charleston is not complete without a carriage ride. Multiple companies offer horse-drawn carriage rides through the city and will keep you entertained with stories of the history of this beautiful city. The carriage rides all start at the Charleston market.

H.L. Hunley - The confederate submarine H.L. Hunley set out from Charleston harbor in 1863 and successfully sank the Union frigate Housatonic on blockade duty. However, the submarine failed to return to port and was lost until rediscovered in 2000. The submarine is being studied at a facility north of Charleston which offers tours of the effort on weekends. Visitors can sit in a replica and get a unique experience imagining what it must have been like for those Confederate sailors.

Charleston Market - The Market was donated to the city under the condition that it remain open 364 days a year (excepting Christmas). The market buildings provide room for vendors of all types and stretch for several blocks. Visitors can find the unique sweetgrass baskets here along with prints, jewelery, woven goods, and a wide selection of hot sauces, among other goods. You can find just about anything in the Charleston market. Surrounding the market are many shops, stores, and restaurants that offer unique gifts and the taste of Charleston.

The Battery - At the southern end of Charleston is the Battery. This was once an artillery battery for defense of the city, but is now a park. It offers expansive views of the harbor and is adjacent to many of the well-preserved, picturesque homes for which Charleston is known.

The Museum Mile – Charleston's Museum Mile stretches along Meeting Street and contains one of the highest concentrations of museums and historical sites to be found anywhere. At the north end is the Charleston Museum, touted as America's first museum, founded in 1773.

Historic House Tours – Tours are available for many of the historic homes in Charleston. The city has a very strict preservation and restoration policy to preserve the unique architecture of the city. Many of these tours include more than one house in their itinerary.

Plantations – Farther afield from the downtown area are preserved plantations from an earlier time. These plantations and their gardens provide a quiet escape from the modern world among the magnolias and oaks.

Aquarium – The South Carolina Aquarium is located in downtown Charleston along the waterfront. The 10-year old aquarium focuses on the different creatures found in South Carolina from the mountains to the ocean. Included are the marshy swamps unique to the Carolinas and exhibits about Loggerhead turtles. This attraction is a big hit with families.

Shopping – Adjacent to the Convention Center in North Charleston is a Tanger Outlet mall with a wide variety of nationally known brands in over 50 shops. More upscale shopping opportunities are located in downtown Charleston along with the market for those unique low-country gifts.

Patriots Point – This is a military park located across the bridge from downtown Charleston in Mount Pleasant. The central attraction is the retired aircraft carrier USS Yorktown. This WW-II carrier fought in the Pacific and was the recovery carrier for the Apollo 8 mission around the moon. The park also includes a Destroyer and Submarine and an exhibit of a Vietnam Naval Support Base. Tours of the carrier provide visitors a unique view of Charleston.

The Food – There is not enough room here to adequately describe the cornucopia of culinary delights available in the Charleston area. From she crab soup and Lowcountry Boil through fresh seafood and barbeque ribs, the treasures of the Charleston areas could keep you entertained for far longer than the duration of the mid-year meeting.

2011 HPS Midyear Meeting Exhibitors

Exhibits are located in Hall A

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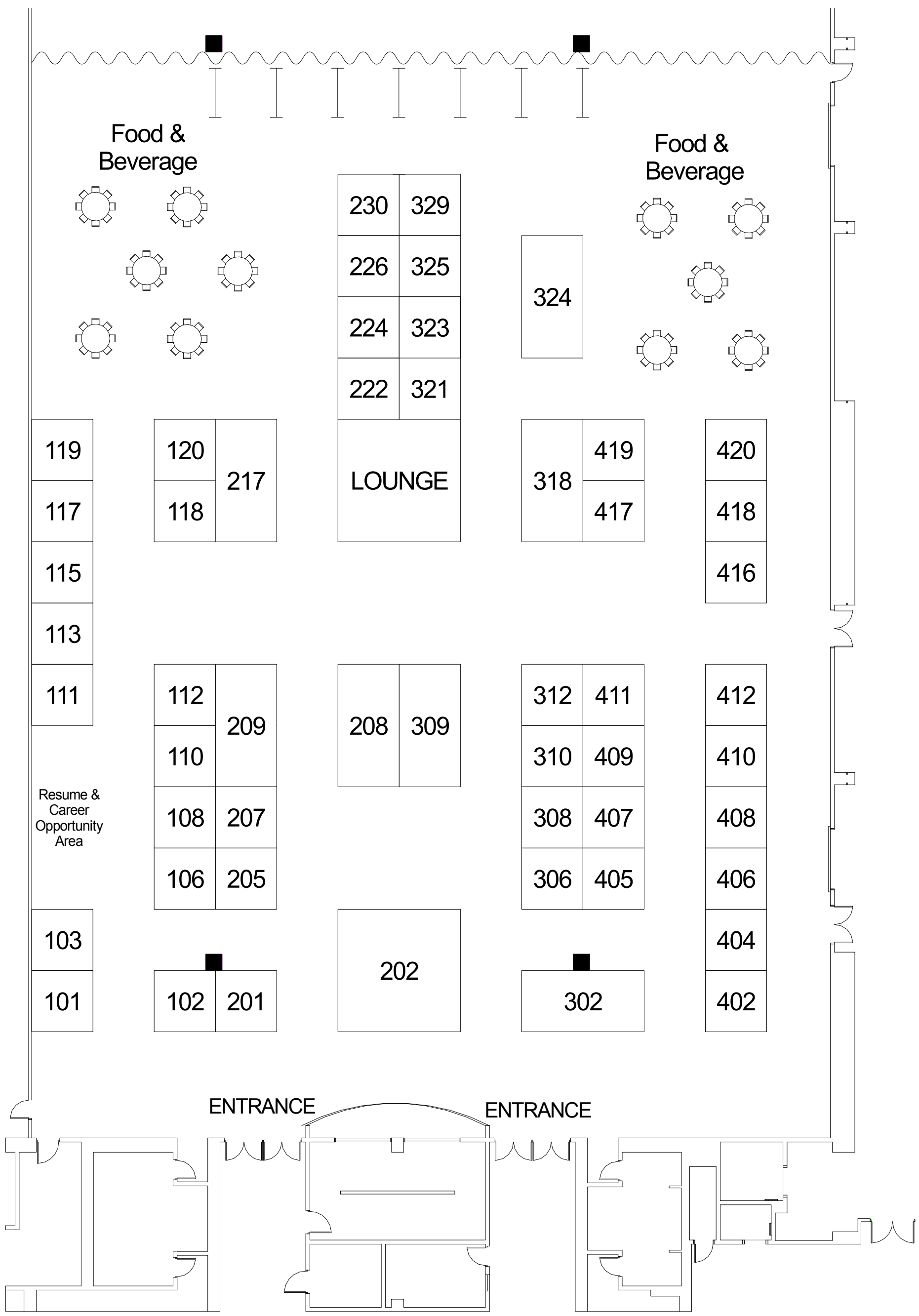
	3:15-3:45 PM	Refreshment Breaks
Wednesday	9:30 AM-Noon	Exhibits Open
	10:00-10:30 AM	Refreshment Breaks



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Exhibits are located in Exhibit Hall A

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2012 Midyear Meeting **Booth:119**
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Saphymo provides measurement devices and systems for radiation protection for the environmental protection, nuclear industry, research centers and homeland security. Product lines are environmental monitoring networks, contamination, dosimetry, portal monitors and radon. Particularly former Genitron Instruments GmbH, Frankfurt, Germany, provides state-of-the-art low-power systems with proprietary radio transmission to US customers as US EPA, DoE, NIST and other public institutes.

Spectrum Techniques **Booth: 409**
106 Union Valley RD
Oak Ridge, TN 37830
865-482-9937; FAX: 865-483-0473
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Gross Counting with Geiger-Mueller and NaI(Tl) detector systems. Spectroscopy with Multi-channel Analyzers for NaI(Tl) detector systems. Exempt Quantity Sealed Radioactive Sources. Nominal and calibrated Gamma. Nominal Alpha and Beta. Isotope Generator.

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www.teletrix.com

Teletrix manufactures simulated radiation training instruments for the nuclear power industry, first responders, the military, state and local government and others. Our products present realistic training scenarios in complete safety without the use of live sources. These high quality meters have been. Made in the USA for over 21 years.

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27 Forge Parkway
Franklin, MA 02038
505-428-3454; FAX: 505-428-3535
www.thermo.com/rmp

Thermo Fisher Scientific is the world leader in serving science. The Fortune 300 company enables its customers to make the world healthier, cleaner, and safer by providing analytical instruments, equipment, software, and services. We are excited to introduce the next-generation personnel contamination monitor, the iPCM12. Enhanced sensitivity and sophisticated electronics keeps your personnel safe and protected. Our product line is comprised of portal monitors, lightweight and accurate hand-held radiation detectors, contamination monitors, mobile radiation detection systems, spectroscopic isotope identification equipment, and ViewPoint™ Enterprise, a remote monitoring software platform that integrates data from remote sensors to provide real-time personnel and area monitoring. Our radiation measurement and protection product lines provide unequalled radiological performance, protection, and solutions for today's challenges. More information and descriptions of our instruments and systems can be found at www.thermo.com/rmp.

Unfors **Booth: 420**

48 Anderson Ave, Suite 1
New Milford, CT 06776
860-355-2588
www.unfors.com

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Barnwell, SC 29812
803-541-7656; FAX: 803-259-0186
www.unitechus.com

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US Navy Recruiting **Booth: 407**

528 Robbins Drive
Troy, MI 48083
248-588-6100; FAX: 586-558-5883

Final Technical Program

If a paper is going to be presented by other than the first author, the presenter's name has an asterisk (*)

All Sessions will take place in the Charleston Convention Center unless otherwise noted

MONDAY

7:00-8:00 am — Ballroom B

CEL 1 Fluoroscopy Occupational Dose Monitoring/EDE Dose Calc Methods
Deidre Elder
University of Colorado Hospital

8:15 am-Noon — Ballroom C

MAM-A Plenary Session

Chair: Ed Maher

8:15 AM
Opening Remarks
Maher E
President, HPS

8:25 AM — MAM-A.1
Past and Future Developments in Radiation Detectors
Knoll GF
*Professor Emeritus, University of Michigan (**G. William Morgan Lecturer**)*

9:10 AM — MAM-A.2
Advances in Instrumentation for Homeland Security
Wrobel M
Domestic Nuclear Detection Office, DHS

9:40 AM — BREAK

10:10 AM — MAM-A.3
The Future of the Modern Radioanalytical Laboratory
Bronson F
Areva-Canberra Instruments

10:40 AM — MAM-A.5
"Get Your Nose Out of My Business!" (The Role of Quality Assurance in Radiation Measurements)
Schwahn S
Oak Ridge National Laboratory

11:10 AM — Roundtable Discussion

1:00-2:15 pm — Ballroom C

MPM-A Measurement QA/QC

Co-Chairs: Ray Johnson, Jeffrey Lively

1:00 PM — MPM-A.1
How Good do Measurements Need to Be - What Quality is Defensible?
Johnson R
Dade Moeller Radiation Safety Academy

1:15 PM — MPM-A.2
The Analysis of a Signal in the Presence of Background for Few Total Counts
Alvarez JL
AlphaBetaGamut

1:30 PM — MPM-A.3
Modified Time-Interval Analysis via Bayes' Theorem for Environmental Radiation Monitoring
Luo P, Sharp JL, DeVol TA
Clemson University

1:45 PM — MPM-A.4
Use of Z-Score Methodology in Analyzing Dosimetry Quality Assurance Results
Chase WJ
Ontario Power Generation

2:00 PM — MPM-A.5
The Power of Data Imaging
Lively J
MACTEC

2:15 PM — BREAK

2:45-5:00 pm — Ballroom C

MPM-B Advances in Instrumentation A

Co-Chairs: Pavel Degtiarenko, Mark Wrobel

2:45 PM — MPM-B.1

Low-Background Gamma Spectrometry for Environmental Assessment

Haines DK, Semkow TM, Khan AJ, Beach SE, Hoffman TJ, Meyer ST*

NYS Dept Health

3:00 PM — MPM-B.2

Long-Term Environmental Radiation Measurements at Jefferson Lab

*Degtiarenko P, Dixon G**

Jefferson Lab

3:15 PM — MPM-B.3

Two Channel Measurement Design of a Multielement TEPC

Waker A, Aslam

UOIT, Canada

3:30 PM — MPM-B.4

Transformation of Geiger Muller Tube GM2416 to an Energy Compensated Counter

Machrafi R, Noor O, Kovalchuk V, Waston R

University of Ontario Institute of Technology, Bubble Technology Industries, Canberra Co.

3:45 PM — MPM-B.5

Applications of the Spectral-Sensitive High Pressure Ionization Chambers at Jefferson Lab

Degtiarenko P, Popov V

Jefferson Lab

4:00 PM — MPM-B.6

Relative Response of Plastic Scintillators to Photons and Beta Particles

Kumar A, Sh. Aydarous A, Waker A

UOIT, Canada, Taif University, Kingdom of Saudi Arabia

4:15 PM — MPM-B.7

Neutron Response and Resolution of the New Tissue Equivalent Proportional Counter System for the International Space Station

Perez-Nunez D, Braby L

Texas A&M University

4:30 PM — MPM-B.8

Response of a Proportional Counter under Moderate Pressures of Counting Gas in Low Energy Neutron Fields

Aslam, Waker A

UOIT, Canada

4:45 PM — MPM-B.9

Advances in Electron Paramagnetic Resonance Dosimetry with Fingernails

Reyes R, Melanson MA, Trompier F, Romanyukha A

Uniformed Services University of the Health Sciences, Armed Forces Radiobiology Research Institute, Institut de Radioprotection et de Seacute, Nucleaire, Naval Dosimetry Center

5:00-6:00 pm — Hall A

Exhibitor Opening Reception

TUESDAY

7:00-8:00 am — Ballroom B

CEL 2 Thermally and Optically Stimulated Luminescence and Their Application in Radiation Dosimetry and Measurement
Stephen McKeever
Oklahoma State University

7:00-8:00 am — Ballroom C

CEL3 ABHP Exam Fundamentals – Tips for Successfully Completing the Certification Process
Patrick LaFrate, Progress Energy

9:00 am-Noon — Ballroom B

TAM-A Instrument Field Use A

Co-Chairs: James Rolph, Tony Mason

9:00 AM — TAM-A.1
EPA Airborne Detection Capabilities
Cardarelli J, Thomas M, Curry T
Environmental Protection Agency

9:15 AM — TAM-A.3
Use of a Helicopter Platform Using a Multiple Sodium Iodide Detector System to Conduct Environmental Scoping Surveys
Lyons CL
National Security Technologies

9:30 AM — TAM-A.5
Development of a Detection Array for Field Work and Instructional Laboratories
Marianno C, Hearn G
Texas A&M University

9:45 AM — TAM-A.6
Methodology for Indoor Geospatial Data Capture of Radiological Contamination Using a Robotic Total Station (RTS) Integrated with a Rate-Meter and Represented with Geographic Information Systems (GIS)
Viars J, Estes B
Oak Ridge Associated Universities

10:00 AM — BREAK IN EXHIBIT HALL

10:30 AM — TAM-A.7
Final Status Survey Application of Ranked Set Sampling for Hard to Detect Radionuclides
Vitkus T
Oak Ridge Associated Universities

10:45 AM — TAM-A.8
Revision 2 to Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)
Snead K, Liles D, Powers G, Williams WA, Alberth D, Doremus S, Bhat R*
US Environmental Protection Agency, US Nuclear Regulatory Commission, US Department of Energy, US Army, US Navy, US Air Force

11:00 AM — TAM-A.9
A MARSAME-Based Release of Hanford Railroad Rails Using Standard Field Radiological Instruments, Computer Data Collection and Analysis, and Release Practices
Rolph JT, Neal JK, Glines WM, Craig JC, Draine AE
Washington Closure Hanford, DOE Richland Operations Office, Eberline Services, Incorporated

11:15 AM — TAM-A.10
Field Experience with a Portable, Field, Alpha and Photon Spectrometer for the Clearance of Property with Contaminated Surfaces
Millsap WJ, Pappin JL, Balmer DK, Glines WM, Brush DJ
Dade Moeller, Mission Support Alliance, Pacific Northwest National Laboratory, US Department of Energy

11:30 AM — TAM-A.11
Detection of Pu-239 Beneath a Monolayer of Stainless Steel Supporting Free Release of Equipment From the Z Machine at Sandia National Laboratories.
Beall PS
Sandia National Labratories

11:45 AM — TAM-A.12
Successful Implementation of Subsurface Soil Derived Concentration Guideline Level Methodology to Achieve Compliance with Unrestricted Release Criteria
Lopez AU, Posner RG, Lively JW
MACTEC Development Corp.

Noon-1:00 pm **Exhibit Hall**

Complimentary Lunch

8:30-9:45 am **Ballroom C**

TAM-B Contemporary Topics A

Co-Chairs: Matthew Barnett, Ken Veinot

8:30 AM **TAM-B.1**

The Psychology of Radiation Measurements

Johnson RH

Dade Moeller Radiation Safety Academy

8:45 AM **TAM-B.2**

Developing an Environmental Monitoring Program for Radiological Operations at a "New" U.S. DOE Site

Snyder SF, Barnett JM, Rhoads K, Poston TM, Fritz BG, Meier KM

Pacific Northwest National Laboratory

9:00 AM **TAM-B.3**

Personal Dose Equivalent Conversion Coefficients for Electrons, Photons, and Positrons

Veinot KG, Hertel NE, Sutton-Ferenci MR

Y-12 National Security Complex, Georgia Institute of Technology, Hershey Medical Center

9:15 AM **TAM-B.4**

Microscope Image Analysis of Immune-Fluorescent Foci as a Biodosimeter for Assessing Neutron-Induced Injury

*Renegar J, Gray D, Wang H, Wang C**

Georgia Tech, Emory University

9:30 AM **TAM-B.5**

New Materials for Individual Emergency Dosimetry using Optically Stimulated Luminescence

Sholom S, DeWitt R, McKeever SWS

Oklahoma State University

9:45 AM ***BREAK IN EXHIBIT HALL***

10:15 am-Noon **Ballroom C**

TAM-C Calibration A

Co-Chairs: Frazier Bronson, Clayton Bradt

10:15 AM **TAM-C.1**

Calibration of Radiation Measurement Instruments with the Help of Primordial Radioisotopes

Iwatschenko M

Thermo Fisher Scientific

10:30 AM **TAM-C.2**

Significant Improvements in Accuracy of Beam Type Calibrators

Port EA, Port NL

RSSI

10:45 AM **TAM-C.3**

Calibration of Germanium Gamma Spectrometry Systems for Radiological Surveillance by Means of Monte Carlo Calculations

Bradt CJ, Semkow TM, Kitto ME

NYS Department of Health

11:00 AM **TAM-C.4**

Determination of the Optimum Container Diameter for the Gamma-Ray Assay of Laboratory Samples

Mueller WF, Bronson F

Canberra Industries, Inc.

11:15 AM **TAM-C.5**

The Applicability of Non-Uniform Matrices for Gamma Spectroscopy Calibration of Uniform Matrices with the Same Average Density

Bronson F

Canberra

11:30 AM **TAM-C.6**

Gamma Spectroscopy Sample Geometries that Minimize Sample Preparation, Minimize the Number of Calibrations Necessary, and Minimize Calibration Uncertainty

Bronson F

Canberra

11:45 AM **TAM-C.7**

Gamma Spectroscopy Counting Geometries that Can be Used for a Wide Range of Sample Conditions with the Same Efficiency Calibration

Bronson F

Canberra

Noon-1:00 pm **Exhibit Hall**

Complimentary Lunch

Noon-1:00 pm **Exhibit Hall**

Poster Session

P.1 Developing and Implementing a Joint Health Physics Technician and Managers Program at Orangeburg-Calhoun Technical College and South Carolina State University

Beharry K, Payne J, Lewis K, Murphy R

South Carolina State University, Orangeburg Calhoun Technical College

P.2 Utilization of Two New Executable Computer Codes for Confidence Intervals, Decision Levels and Detection Limits when the Sample is Counted an Integer Times Longer than the Blank
Potter WE, Strzelczyk J
California, Consultant, University of Colorado Hospital Measurement QA/QC

P.4 Optimization of Plastic Scintillator Thicknesses for Online Beta Detection in Mixed Fields
Pourtangestani K, Machrafi R
University of Ontario Institute of Technology, Canada

P.5 Assessment of Annual Effective Dose from, and in Soil and Their Effect on Human Health
Shafiey E, Changizi
Iran

P.6 Middle East Radiation Measurements Cross Calibration Workshops
Miller M, Mohageghi A, Ghanbari F
Sandia National Laboratory

P.7 A Detector for Simultaneous Beta & Gamma Spectroscopy
Caffrey J, Mangini CD, Farsoni AT, Hamby DM
Oregon State University

P.8 Radio Frequency Identification with Radiation Monitoring Ability
Lee JH, Anderson J, Tsai H, Craig B, Liu Y, Shuler J
Argonne National Laboratory, Department of Energy

P.9 Making It Real - Building a Technical College Radiation Protection Technology Program from Scratch with State of the Art Survey and Detection Equipment
Miller W
Aiken Technical College/Savannah River Nuclear Solutions

P.10 Back-Projected Radiation Analyzer and Cell Evaluator (BRACE) for Hot Cell Characterization
Rusty JR, Farfan E, Jannik GT
Savannah River National Laboratory

P.11 Introducing Students to Detection: Aluminum Decay Labs at Oregon State University
Bytwerk D, Reese S, Higley K, Darrough J
Oregon State University

P.12 Use of Helicopter Platform for Large Area Radiation Surveys
Favret D, Lyons CL, Plionis AA
National Security Technologies

P.13 Digital Processing of Multi-Component Signal Pulses
Mangini CD, Caffrey JA, Farsoni AT, Hamby DM
Oregon State University

P.14 Radially Dependant Directional Shield (RDDS) Used for Hot Cell Characterization
Coleman JR, Farfan EB, Jannik GT
Savannah River National Laboratory

P.15 Bayesian Analysis to Produce Usable Measurement Results for Everyone, Even Those with Negative Values
Strom DJ
Pacific Northwest National Laboratory

1:00-4:00 pm — Ballroom B

TPM-A Instrument Field Use B

Co-Chairs: James Rolph, Tony Mason

1:00 PM — TPM-A.1
Determination of MDA Levels for Radiation Surveys of Potentially Activated PCB Capacitors
Butala S, Munyon W, Micklich B, Vacca J
Argonne

1:15 PM — TPM-A.2
Calculating Field Measurement Method Uncertainty
Hay S
Fallcrest, Inc.

1:30 PM — TPM-A.3
Communicating Radiation Risks with Instruments and Dosimeters
Brodsky A, Jones S
Georgetown University, Physicians for Civil Defense

1:45 PM — TPM-A.4
The Application of Super Heated Drop (Bubble) Detectors for the Characterization of Nano-Second-Pulsed Neutron Fields
Ward D, Cordova L
Sandia National Laboratories

2:00 PM — TPM-A.5
Use of Portal Monitors for Evaluation of Internal Contamination after a Radiological Dispersal Device
Palmer R, Hertel N, Ansari A, Burgett E
Georgia Institute of Technology, Centers for Disease Control and Prevention

2:15 PM **TPM-A.6**

Assessing Internal Contamination Levels for Fission Product Inhalation using a Portal Monitor

Freibert E, Hertel N, Ansari A

Georgia Institutue of Technology, Centers for Disease Control and Prevention

2:30 PM ***BREAK IN EXHIBIT HALL***

3:00 PM **TPM-A.7**

A Comparison of Shielding Components and Practices in Interventional Cardiology

*Tannahill G, Fetterly K, Hindal M, Magnuson D, Sturchio G**

Mayo Clinic

3:15 PM **TPM-A.8**

Evaluation of NCRP 147 CT Shielding DLP Method

Broga D

Virginia Commonweath University

3:30 PM **TPM-A.9**

Application of Instruments in Medical Treatment Facilities

Stewart H, Melanson M

Army, Eisenhower Army Medical Center, Armed Forces Radiobiology Research Institute

3:45 PM **TPM-A.10**

Study of TENORM in Samples Gathered from BP Oil Spillage from the Coasts of Mississippi and Louisiana

Aceil S, Billa J

Alcorn State

1:00-2:45 pm **Ballroom C**

TPM-B Calibration B

Co-Chairs: Nolan Hertel, Tom Goff

1:00 PM **TPM-B.1**

Verification of a Conservative TLD Neutron Correction Factor at the WIPP

Goff TE, Hayes RB, Sleeman RE

WIPP

1:15 PM **TPM-B.2**

Determination of a Site-Specific Spectrum Correction Factor in the Vicinity of the Holtec MPC During Drying in the Keuwanee Nuclear Power Station

Hertel NE, Blaylock D, Cahill T, Exline P, Burgett E, Olson C, Adams R, McGreal M

Georgia Institute of Technology, Idaho State University, Dominion Energy Kewaunee

1:30 PM **TPM-B.3**

Effects of Different Moderators on the Neutron Spectra, Fluence and Dose Rates from Californium Source

Radev R, Shingleton K

Lawrence Livermore National Laboratory

1:45 PM **TPM-B.4**

US Army Radiation Standards Laboratory

Howard SV

US Army TMDE Activity

2:00 PM **TPM-B.5**

Construction and Maintenance of Reference Radiological Calibration Fields of Kaeri

Kim BH, Han SJ, Kim JL, Kim JS, Lee JI, Kim SI, Chang IS

Korea Atomic Energy Research Institute, Korea Insitute of Nuclear Safety

2:15 PM **TPM-B.6**

Production of Fast Neutron Calibration Fields Using a Proton Accelerator of Kirams

Kim BH, Cho KW, Kim SI, Kim JL

Korea Atomic Energy Research Institute, Korea Institute of Nuclear Safety

2:30 PM **TPM-B.7**

Development of Automatic Clearance Measurement System Using Shape Measurement and Monte Carlo Calculation

Hattori T, Sasaki M

Central Research Institute of Electric Power Industry

2:45 PM ***BREAK IN EXHIBIT HALL***

3:15-4:45 pm **Ballroom C**

TPM-C Contemporary Topics B

Chair: Wayne Gaul

3:15 PM **TPM-C.1**

Monte Carlo Simulation of Entrance to Exit Air Kerma Ratio in Interventional Radiology

He W, Mah E, Huda W, Yao H

Clemson University, Medical University of South Carolina

3:30 PM TPM-C.2

Radiation Dose Measurement - Analysis for a 320 Slice CT Scanner

Nickoloff E, Lu Z, Dutta A, So J
Columbia University

3:45 PM TPM-C.3

Determination of Air Crew Exposure in Domestic Flights of Aseman Airline in Iran. On Board Measurements and Calculations with CARI 6 Code

Mehdizadeh S, Faghihi R, Sina S, Zehtabian M
Shiraz University, Iran

4:00 PM TPM-C.4

Lead-210 and Polonium-210 in Iron and Steel Industries

Khater A, Bakr W
King Saud University, Egyptian Atomic Energy Authority

4:15 PM TPM-C.5

Making the Most of Uncertain Low-Level Measurements

Strom DJ, Joyce KE, MacLellan JA, Watson DJ, Lynch TP, Antonio CA, Birchall A, Zharov PA
Pacific Northwest National Laboratory, UK Health Protection Agency, Mayak Production Association

4:30 PM TPM-C.6

On the Detection Efficiency of the RaDeCC System for Ra-224 and Ra-223 Measurements

Chang Z, Moore W, Tan S, Bett B
South Carolina State University, University of South Carolina

WEDNESDAY

7:00-8:00 am **Ballroom B**

CEL 4 Remodel the Facility and Remodel the Technology: A Practical Application
Jack Horne

8:20-10:00 am **Ballroom B**

WAM-A Radon/Environmental Section Presentations

Co-Chairs: James Menge, A. George

8:20 AM
Special Presentation to Peter Lyons, Distinguished Public Service Award

8:30 AM **WAM-A.1**
Radon Rejection in Next Generation Contamination Monitor
Menge JP
ThermoFisher

8:45 AM **WAM-A.2**
TRU Measurement and Screening Assay of Air Filters with Radon Progeny Interference
Hayes RB, Pena AM
WIPP

9:00 AM **WAM-A.3**
Current State of the Art in Measuring Radon
George AC, Bredhoff N
Radon Testing Corp of America, Inc.

9:15 AM **WAM-A.4**
Correction to Counting Statistics for Measurements of Radon in Air Using Continuous Monitors and Alpha-track Devices
Jenkins PH
Bowser-Morner, Inc.

9:30 AM **WAM-A.5**
Radon Reference Chambers in the U.S. and Radon Measurement Performance Testing
Jenkins PH, Burkhart JF, Palmer JM
Bowser-Morner, Inc., University of Colorado - Colorado Springs, US Environmental Protection Agency

9:45 AM **WAM-A.6**
Development and Intercomparison of Radon-in-Water Standards
Kitto M, Bari A, Menia T, Haines D, Fielman E
New York State Department of Health

10:00 AM ***BREAK IN EXHIBIT HALL***

10:30 am-Noon **Ballroom B**

WAM-B Instrument Laboratory Use A

Co-Chairs: C. Li, TM Senkow

10:30 AM **WAM-B.1**
Some Bioassay Methods for High-risk Radionuclides
Li C, Sadi B, Ko R, Kramer G
Health Canada

10:45 AM **WAM-B.2**
Alpha Spectrometry of Thick Samples for Environmental Monitoring
Semkow TM, Khan AJ, Haines DK, Bari A
New York State Department of Health

11:00 AM **WAM-B.3**
Inductively Coupled Plasma Mass Spectrometry Measurement of Technetium-99 Including Uncertainty and Detection Limit Determinations
Timm R, Strock J, Schoneman J, MacLellan J, Chambers J
GEL Laboratories LLC, Pacific Northwest National Laboratories, Bechtel Jacobs Company LLC

11:15 AM **WAM-B.4**
Deconvolution of Mixed Gamma Emitters Using Peak Parameters
Gadd MS, Garcia F, Vigil MM
Los Alamos National Laboratory

11:30 AM **WAM-B.5**
Determination of Energy Spectra and Absorbed Dose Rate of a Ni-63 Based Low-Energy Beta Source
*Gibb R, Renegar J, Wang C**
Georgia Tech

11:45 AM **WAM-B.6**
Intercomparison of Direct Radiobioassay and Radiochemical Analysis of Tissue Specimens from a Plutonium and Am-241 Contaminated Wound
Carbaugh E, Lynch T
Pacific Northwest National Laboratory

8:30 am-Noon **Ballroom C**

WAM-C Coordination and Planning for Field and Laboratory Measurements Following a Radiological or Nuclear Accident

Co-Chairs: Carl Gogolak, Robert Shannon

8:30 AM **WAM-C.1**

Uses of Field and Laboratory Measurements during a Radiological or Nuclear Incident

Shannon R, Gogolak C, McCurdy D, Litman R, Griggs J, Burns D, Berne A

Environmental Management Support, Inc., US Environmental Protection Agency, National Air and Radiation Laboratory (NAREL)

9:00 AM **WAM-C.2**

Essential Metrology for Field and Laboratory Measurements during a Radiological or Nuclear Incident

Gogolak CV, Shannon R, McCurdy DE, Litman R, Griggs J, Burns D, Berne A

Environmental Management Support, Inc., US Environmental Protection Agency, National Air and Radiation Laboratory (NAREL)

9:30 AM **WAM-C.3**

Emergency Response-Field vs. Lab Measurement

Walker E

Consultant, Tennessee

10:00 AM ***BREAK IN EXHIBIT HALL***

10:30 AM **WAM-C.4**

FRMAC Interactions During a Radiological or Nuclear Event

Wong CT

Lawrence Livermore National Laboratory

11:00 AM **Open Panel Discussion**

1:00-3:30 pm **Ballroom B**

WPM-A Accelerator Session

Co-Chairs: Samuel Baker, Roger Moroney

1:00 PM **WPM-A.1**

Commissioning of the Fission Fragment Ion Source

Baker SI, Pardo RC, Savard G, Davids CN, Greene JP, Levand AF, Scott RH, Sun T, Vondrasek RC, Zabransky BJ

Argonne National Lab

1:15 PM **WPM-A.2**

Quantification of Induced Radioactivity for a Compact 11 MeV Self-Shielded Cyclotron for Decommissioning Funding Purposes

Moroney WR, Krueger DJ, Elam CL, Plastini FL, Chance AC

Siemens MI

1:30 PM **WPM-A.3**

Comparison of Two Techniques for Measuring Gamma Dose near Berkeley Lab Accelerators

Wahl LE

Lawrence Berkeley National Laboratory

1:45 PM **WPM-A.4**

Count Rate Limitations in Pulsed Accelerator Fields

Justus A

Los Alamos National Laboratory

2:00 PM **WPM-A.5**

Neutron Operational and Protection Quantity Conversion Coefficients Under ICRP-26, ICRP-60, and ICRP-103

Veinot KG, Hertel NE, Sutton-Ferenci MR

Y-12 National Security Complex, Georgia Institute of Technology, Penn State Hershey Medical Center

2:15 PM ***BREAK***

2:45 PM **WPM-A.6**

Large-scale Production of Mo-99 Using a 100-kW Proton Beam

Nolen JA, Gomes IC

Argonne National Laboratory, I.C. Gomes Consulting and Investment, Inc.

3:00 PM **WPM-A.7**

Validation and Verification of MCNP6 as a New Simulation Tool Useful for Medical Applications

Mashnik S

Los Alamos National Laboratory

3:15 PM **WPM-A.8**

A New Method to Measure Potential Accelerator Hot-Spots

Marceau-Day ML

CAMD/LSU

1:15-3:00 pm	Ballroom C

WPM-B Instrument Laboratory Use B

Co-Chairs: MS Gadd, Ed Tupin

1:15 PM **WPM-B.2**

A Comparison of InLight Reader and MicroStar Reader Performance

Cunningham Beckfield F, Kirr M, Passmore C*
Landauer, Inc.

1:30 PM **WPM-B.3**

Development of an On-line Radiation and Detection Measurements Lab Course

Kopp DG, DeVol TA
Clemson University

1:45 PM **WPM-B.4**

Comparing LS System Detection for Liquid, Cherenkov, and Nitrogen Scintillations

*Rosson R, Lahr J, Kahn B**
Georgia Institute of Technology

2:00 PM **WPM-B.5**

Radioanalytical Criteria for Emergency Response

Tupin EA, Griggs J, Gogolak CV
US Environmental Protection Agency, Environmental Management Support

2:15 PM **WPM-B.6**

Occurrence of Natural Radionuclides in the Drinking Water Supplies of Shiraz and Spring Waters of Fars Province

Mehdizadeh S, Faghihi R
Shiraz University, Iran

2:30 PM **WPM-B.7**

Natural and Artificial Radioactivity Distribution in Soil of Fars Province, Iran

Mehdizadeh S, Faghihi R
Shiraz University, Iran

2:45 PM **WPM-B.8**

Uranium in Phosphate Fertilizer using Different Analytical Techniques

Khater A
King Saud University

3:00 PM ***BREAK***

3:30-4:45 pm	Ballroom C

WPM-C Advances in Instrumentation B

Co-Chairs: James Menge, Ben Edwards

3:30 PM **WPM-C.1**

Advanced Radiological Scanning Technologies Produce Superior Survey Results

Lopez AU, McDonald MP
MACTEC Development Corp.

3:45 PM **WPM-C.2**

Advances in Detection Technology for Homeland Security

Wrobel M
DHS/Domestic Nuclear Detection Office

4:00 PM **WPM-C.3**

Computer Program Simulation of a Moving Alpha or Beta Particle Detector Across a Contaminated Surface

Farrar DR, Alecksen TJ, Schierman MJ, Baker KR
Environmental Restoration Group, Inc.

4:15 PM **WPM-C.4**

Verification of Dose Correction Factors of MOSFET Dosimeters for Use in Anthropomorphic Phantom to Measure Equivalent Doses and Effective Dose

Cho S, Cho KW, Kim CH, Yi CY, Jeong JH*
Hanyang University, Korea Institute of Nuclear Safety, Korea Research Institute of Standards Science

4:30 PM **WPM-C.5**

Recent Progressive Developments of Radioactivity Measurement Techniques - A European Perspective

*Maushart R, Wilhelm CH**
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Continuing Education Lectures

CELs take place in the Charleston Convention Center

Monday, February 7 7:00-8:00 am

CEL 1 Fluoroscopy Occupational Dose Monitoring/ EDE Dose Calc Methods

Deidre Elder; University of Colorado Hospital

Medical environments where fluoroscopy is routinely used offer unique challenges for monitoring and reporting of occupational radiation doses. The use of lead aprons and thyroid shields results in a substantially reduced effective dose. However, the non-uniform exposure makes occupational dose monitoring and reporting more difficult. Over the decades, international and national expert bodies expressed varied opinions and made inconsistent recommendations. As a result, radiation regulations vary from state-to-state and a number of approaches involving multiple or single dosimeters are practiced. Regulatory Guide 8.40 was issued in July 2010 to describe methods that the Nuclear Regulatory Commission considers acceptable for determining the effective dose equivalent for external radiation exposures. The National Council on Radiation Protection and the Conference of Radiation Control Program Directors have also published recommended methods of effective dose and effective dose equivalent determination for individuals with non-uniform exposures due to the use of protective garments. The practical considerations of financial and behavioral issues are also considered when determining the method of occupational dose monitoring and dose determination for healthcare workers.

Tuesday, February 8 7:00-8:00 am

CEL 2 Thermally and Optically Stimulated Luminescence and their Application in Radiation Dosimetry and Measurement

Stephen McKeever; Oklahoma State University

Thermally stimulated luminescence (TL) from insulating materials has been used in radiation dosimetry since the early 1950s, although the genesis of the techniques goes back to the early studies of radiation and radioactivity. More recently, optically stimulated luminescence (OSL) has emerged as a powerful dosimetry method.

Both TL and OSL rely on the perturbation of the material from thermodynamic equilibrium via the absorption of energy from a radiation field and the creation of point defects via ionization processes. A thorough understanding of TL and OSL in any given material would require a detailed knowledge of the nature and spatial distribution of the radiation-induced defects, and their subsequent interaction during thermal or optical stimulation. However, basic – and, in many cases, sufficient - understanding can be obtained via phenomenological descriptions of the electronic transitions between energy levels during the irradiation and stimulation processes.

This talk will describe the fundamentals of TL and OSL and discuss some of the processes that give rise to the phenomena in popular TL and OSL dosimetry materials. The talk will then show how this background understanding can assist in the application of these methods in traditional and emerging radiation measurement and dosimetry applications. Modern applications include Personal, Environmental, Retrospective, Neutron, Space, Medical and Emergency radiation dosimetry and measurement and descriptions of applications in these fields will be included in the talk.

CEL3 ABHP Exam Fundamentals – Tips for Successfully Completing the Certification Process

Patrick LaFrate; Progress Energy

This presentation will show the advantages of being certified and discuss the fundamentals of the ABHP exam process – from submission of the exam application to completion of the Part 2 examination. Topics of discussion will include: *What are qualifying academic requirements? *Why require a degree? *What is meant by “professional level” experience? *How are the exams (Part 1 and Part 2) prepared? *How is the passing point determined? *What are the keys to good performance on the exam? *What pitfalls exist that detract from good exam performance?

This presentation will help persons interested in certification prepare an application that will accurately reflect the applicant’s education and experience as well as provide tips for preparing to take the exam and answering part 2 questions in a format that promotes awarding partial credit. Persons who are already certified may gain insight into the process and identify areas where they would be willing to assist in the certification process. The material presented consolidates pertinent exam policy/procedure into an easily digestible format, offering real world examples of good and poor performance.

Wednesday, February 9 7:00-8:00 am

CEL 4 Remodel the Facility and Remodel the Technology: A Practical Application

Jack Horne

This CEL addresses challenges encountered when obtaining new instrumentation and getting it implemented in the field while permitting laboratory facilities to continue full operation. During 2010, the RPL facility – a DOE Class 2 nuclear research facility – started a push to upgrade radiation protection instrumentation. The upgrades targeted three aspects of measurement performed at the facility: Hand and foot exit survey instrumentation (PCMs), single-sample LSCs for local tritium counting, and electronic dosimeters. While numerous challenges were encountered during this upgrade effort, a number of advantages (including cost savings) were realized by facility personnel and management that validated the need for the upgrade.

MAM-A.5 Get Your Nose Out of My Business! (The Role of Quality Assurance in Radiation Measurements)

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When we believe our radiation measurements to be "good," we frequently don't have a common understanding about what "good" means. We have the tendency to want to say, "Get your nose out of my business!" After all, who knows what we do and why we do it better than we do – the professionals performing our own jobs? If we develop common frames of references, we can more effectively make the largely subjective decision – whether what we are doing is "good" – or even adequate. Quality assurance is vitally important in radiation measurements. We owe our employers – even if they are ourselves – not only the best that we can do, but the highest level of quality that we can reasonably achieve. This "quality" means not only getting correct answers, but that it is achieved by knowing what the requirements are, and by approaching them in a planned and systematic fashion. Understanding how to implement quality in radiation measurements involves understanding how standards (especially consensus standards), regulations, policies, and procedures fit into the picture. Though an organization can determine for itself if it meets the requirements of standards, regulations, policies, and procedures, it is a much more convincing process to have an uninterested third party attest to its quality. This third-party attestation (often called accreditation) demonstrates to our customers that we not only have high quality in our own estimation, but also are able to demonstrate it to informed but independent observers. We should ensure that our radiation measurements are of demonstrable high quality because it's the right thing to do – people depend on our measurements to keep them safe. It helps to make our program defensible against lawsuits, can result in reduced numbers of external reviews, may be less expensive in the long run, and will likely result in increased customer satisfaction.

MPM-A.1 How Good do Measurements Need to Be - What Quality is Defensible?

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Specialists in radiation measurements, especially in a laboratory, tend to want to continuously improve the quality of their measurements. While this could be an admirable goal, the question of how the measurement may be used for decision making may be lost. For example, if you have a decision criterion for swipes of 2,000 dpm per swipe, does it really matter if a result is 20,000 dpm or 200,000 dpm? While these results differ by an order of magnitude, they will both lead to the same decision, namely to clean up. Another example has to do with minimum detectable activity (MDA). Do you need an MDA of 50 or 100 dpm when your action level is 2,000 dpm? Sometimes it seems that measurements take on a life of their own, independent of the intended use of the data. Samples may be collected haphazardly (such as swipes) and the quality of the measurement exceeds the quality of the sample. Some of the factors affecting quality of measurements include: calibration conditions, energy dependence, factors affecting ion chambers and GM detectors, operator judgments, background interference, backscatter and self absorption, geometry, and wrong detector or wrong probe. Other variables include sensitivity of instruments, counting time, size of the signal, uniformity of samples, sample location, sample selection bias, sample preparation, and weight and volume errors. Most importantly for measurement quality is that radiation is a random phenomenon, standards are uncertain, and background varies. Thus, all radiation measurements are at best only "best estimates." How many significant figures are warranted by the uncertainty of the measurement? Most often the reported significant figures exceed the quality or defensibility of the measurement. This may lead decision makers to reach conclusions that are not defensible.

MPM-A.2 The Analysis of a Signal in the Presence of Background for Few Total Counts

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The simple process of subtracting an average background from a signal plus background may result in a negative net signal count or, if positive, confidence intervals that extend below zero. Statistical analysis of a data set containing negative numbers is spurious and more so if 'less than' is used in place of negative numbers. Confidence intervals are often the result of a Gaussian propagation of error and do not represent the actual distribution. A Bayesian method is presented for determining the distribution of the net signal and the credibility interval of the signal based on equal probability intervals. The method uses the background distribution in place of the average background. The prior background distribution is described with a gamma distribution and need not be Poisson. The calculation of the distribution may require a lengthy integration by parts. An algorithm is presented that aids determining the background prior, performs the integration by parts, and calculates the likelihood and credibility interval.

MPM-A.3 Modified Time-Interval Analysis via Bayes' Theorem for Environmental Radiation Monitoring

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The application of Bayes' theorem to time-interval (time difference between two successive pulses) analysis is studied to detect changes in the environmental radiation level. Simulated time-interval data were obtained using Monte Carlo techniques to randomly sample the Poisson distributed time intervals. Experimental data were acquired with a DGF-4C (XIA, Inc) system connected to a 2x2in. NaI(Tl) detector. All statistical algorithms were developed using R (R Core Development Team, 2010). False positive rate and detection probability based on Bayesian analyses of time-intervals and counts in a fixed count time are compared with frequentist analyses of the count information. Two modifications to time-interval analysis, enhanced reset and moving prior, are proposed to improve the performance of radiation monitoring. The enhanced reset method sets a two-stage limit for maximum data points contained in the prior distribution and sets a discriminator to determine whether the current prior information should be considered further. As designed, the enhanced reset method has the potential to alleviate the weight of the prior distribution on the posterior distribution. The moving prior method relies on the latest information to calculate the posterior by updating the prior with each new data point. Both modified methods resulted in a higher detection probability than typical Bayesian analyses. The performances of the two modified methods were independent of change point relative to typical Bayesian analyses when time-interval information is utilized. Advantages and disadvantages of the two modifications of the Bayesian method as compared with the conventional frequentist analyses are discussed.

MPM-A.4 Use of Z-Score Methodology in Analyzing Dosimetry Quality Assurance Results

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As part of Ontario Power Generation's (OPG's) personal and environmental dosimetry programs, quality assurance testing is done by external facilities that either irradiate dosimeters or supply samples with unknown activity levels. The dosimeters or samples are measured and analyzed by OPG, and the external facility reports on the results, usually by reporting the relative response of the individual results, and the average relative response and the coefficient of variation (CoV) of the population of results. Typically there are regulatory or other limits for the average relative response and CoV. The performance of OPG's systems is such that the limits are seldom exceeded. To provide a better indication of the health of the measurement systems, the Z-score methodology is used. This comprises calculating the standard deviation of the average relative responses, either from a large number of past results or theoretically. The relative bias in a subsequent average relative response result is divided by the standard deviation to give the Z score, zeta. A result where |zeta| < = 2 is Satisfactory, 2 < |zeta| < = 3 is Questionable, and |zeta| > 3 is Unsatisfactory. Two examples where |zeta| > 3, one from personal dosimetry and one from environmental sampling, show how this method can be used to identify problems even though the results fall within test limits. The relationship between the zeta score denominator and the test limits is also discussed.

MPM-A.5 The Power of Data Imaging

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A popular adage is a picture is worth a thousand words. Data imaging and visual data assessment are veritable gold mines in the scientist's quest to understand and accurately interpret numerical data. Graphical displays of various aspects of a dataset offer the analyst insight to the data that no mathematical computation or statistic can provide. The advent of the global positioning satellite system has enabled scientists from many fields of endeavor to collect and view data in its spatial context. The spatial context of radiological data is an imminently powerful asset in the health physicist's data evaluation arsenal. So powerful is data collected with spatial context that a relatively new branch of mathematical statistics (geospatial statistics) has emerged. This discipline seeks to exploit this context rich data form to better understand the spatial relationships that might exist, but would be otherwise hidden in tabular data or obscured with classic statistical techniques. This presentation will show the power that spatial visual data assessment provides. It will challenge the traditional mathematical concept of detection limits for scanning. Additionally, it will demonstrate that more data, even if the individual datum comprising the dataset is of relatively poorer quality (i.e., has a larger uncertainty and, thus, a larger calculated minimum detection value), is significantly more powerful than a smaller dataset comprised of higher quality measurements. This presentation will cause the open-minded health physicist to rethink how he or she prescribes, collects, evaluates, and makes decisions based upon radiological scan data.

MPM-B.1 Low-Background Gamma Spectrometry for Environmental Assessment.

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Low-background gamma spectrometry is an important tool for basic and applied projects involving gamma-ray measurements and assessments. The applied uses at New York State Department of Health involve monitoring of environment, surveillance of facilities, as well as health physics and homeland security applications. Low-background gamma spectrometry is applicable to very low activity matrices, such as water or chemically separated samples. Specific projects include analysis of radium in drinking water and monitoring of cesium near nuclear power plants. We currently use a 132% efficient Ge detector in an ultra-low background cryostat configuration. The detector is inserted into a 3-layer lead shield consisting of Boliden- and Plombum-grade lead, each 3-inch thick. We have recently added a 2-cm thick Alpha-lo grade lead insert. The lead shield is surrounded from the top and the sides by a plastic-scintillator muon shield. The spectrometer is placed in a 6-inch thick wall steel room made of pre-World War II steel, which is located under a 47-story building providing 33 m of water equivalent shielding. Detailed operation of the ã spectrometer and all its components is described. We have achieved an overall background reduction by a factor of 9436 relative to the ambient. Our integrated background rate in the energy range of 50-2700 keV was measured at 15 counts/ks/(kg Ge) (approximately 2 cpm). This is second best to the IAEA MEL Monaco laboratory, which achieved the value of 10, among ground-level located laboratories. In spite of this substantial background reduction, residual background was detected and attributed to cosmic-ray neutron induced activation and excitation, natural radioactivity, as well as the remaining muon field. Additional measures are attempted to further reduce the background, such as ultrapure materials for cryostat construction, bottom plastic scintillator, low-Z fillers, and electronic tuning.

MPM-B.2 Long-Term Environmental Radiation Measurements at Jefferson Lab

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The Continuous Electron Beam Accelerator Facility (CEBAF) at Jefferson Lab (JLab) is a radiation-generating installation in the Hampton Roads, Virginia metropolitan area. Experimental accelerator operations produce prompt neutron and gamma dose rates at the CEBAF boundary, mostly through neutron or gamma penetration and cascading in the relatively thin roofs of the experimental halls, with subsequent re-scattering and cascading in the atmosphere (neutron or gamma skyshine). Accelerator operations also produce small quantities of radioactive isotopes in the accelerator tunnel, experimental halls, and beam dumps. Environmental radiation monitoring at JLab presents the challenge of measuring weak signals over the natural radiation background, which exceeds the signals of interest and varies over time. Administrative requirements limit the yearly operational radiation dose accumulation at the boundary to no more than 10 mrem (about 10% of the background level). This paper presents results of neutron and gamma radiation measurements at the CEBAF boundary. The boundary dose varies from 1 to 5 mrem per calendar year; neutrons contribute about 80%. Until present, neutron skyshine signals could be measured with the precision needed. Operational gamma dose was evaluated indirectly using the relative production of gamma and neutron radiation, measured separately. A recent innovation in low-level environmental gamma monitoring, using spectroscopic high pressure ion chambers, now achieves and sustains necessary sensitivity and long-term stability. During the last two years of operation we have measured environmental gamma dose rate contributions from CEBAF operations at levels down to 0.3-0.5 microrem/h, nearly continuously. Operational gamma dose accumulations measured at the CEBAF boundary are in good agreement with the previously established gamma to neutron dose ratios.

MPM-B.3 Two Channel Measurement Design of a Multielement TEPC

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Tissue equivalent proportional counters (TEPC) have been long considered suitable candidate instruments for more accurate neutron monitors in nuclear power plants, however, it is highly desirable to have counters with increased sensitivity and smaller physical size to enable production of truly light-weight radiation protection devices. An advanced design of an in-house built tissue equivalent proportional counter (TEPC), which consists of 61 individual cylindrical counting volumes in a compact configuration in a single block of tissue equivalent plastic, is described. We demonstrate its performance to produce a truly light-weight radiation protection device by comparing its sensitivity and the measured lineal energy spectra with that of a ten times larger commercially available TEPC in the neutron energy range between 34 and 354 keV. The mixed photon and neutron fields were generated by employing 7Li(p, n)7Be reaction at the McMaster University 1.25 MV double stage Tandetron accelerator with proton beam of energies between 1.89 and 2.56 MeV. The performance of such counters and those commercially available is undermined to an extent in radiation environments where the dose rate of the low LET radiation component dominates over the neutron component. To enhance the performance of our advanced design of TEPC for such radiation environments, the design is further optimized to measure the photon component with 7 sub-elements and the neutron component with the remaining 54 counting elements. The design of this two channel measurement system is discussed along with a preliminary set of measurements to demonstrate its performance for mixed photon and neutron monitoring in nuclear power plants.

MPM-B.4 Transformation of Geiger Muller Tube GM2416 to an Energy Compensated Counter

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Geiger Muller counter GM2416 produced by CANBERRA company has been redesigned to an energy compensated Geiger Muller tube. The commercially available counter has an acceptable flat response when the emitting gamma radiation source is far away at a distance of around two meters. A special shielding, with a composition of different materials in different thicknesses, has been used. A Monte Carlo model using MCNP/X has been build to simulate the response of the counter and measurements have been carried out, with different gamma energies ranging from 56 to 1332 keV, using an X-ray machine as well as 137C and 60Co gamma sources. With different configurations of the shielding material, the level of flattening the response of the detector was brought to an acceptable level. Results of this investigation will be presented and the ability of the counter to, independently, respond to different gamma energies at closer distance will be discussed.

MPM-B.5 Applications of the Spectral-Sensitive High Pressure Ionization Chambers at Jefferson Lab

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We have recently developed new pulse mode front-end electronics readout, and customized signal processing algorithms for the industry-standard Reuter-Stokes RSS-1013 argon-filled high pressure ionization chambers (HPIC). The new systems are capable of detecting individual events of gas ionization in the HPIC, caused by interactions of gammas and charged particles in the gas. The new schematic does not require a DC connection to the detector, and thus allows the user to avoid known problems of the temperature-dependent voltage and current biases in the front-end electronics cascades of current-integrating (or electroscopic) type. Demonstrated stability makes the system practical by eliminating the need in frequent calibrations. The technique provides enough spectroscopic information to distinguish between several different types of environmental and man-made radiation. The sensitivity and stability of the readout allow long-term environmental radiation monitoring with unprecedented precision. Several HPIC devices of this type are used in the low-level environmental measurements at Jefferson Lab for more than two years. We discuss briefly the novel features of the detectors' front-end electronics readout and customized signal processing design, illustrated by the examples of the long-term environmental measurements. Future applications of the technique in the area of the radioactive waste monitoring are presented.

MPM-B.6 Relative Response of Plastic Scintillators to Photons and Beta Particles

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A scintillation counting system has been constructed with the use of BC-400 series plastic scintillators along with a subminiature photomultiplier tube to investigate the effect of increasing plastic scintillator thickness on system integrated counts. Measurements have been carried out using four different gamma sources with different energies ranging from 6keV to 1.332MeV and a Ni-63 beta source of maximum energy 66keV. Scintillator thicknesses ranged from 10ìm to 2500ìm and the response of the system was determined by measuring the integrated counts as a function of scintillator thickness. The results of these measurements showed that there was a positive linear correlation between scintillator thicknesses and integrated counts for all the gamma sources while the slopes of the correlations of each gamma source depended on the source energy. The beta particle response showed an initial linear increase of counts with scintillator thickness followed by a plateau. The result of these findings will be used in an assessment of the potential of a plastic scintillator system forming the basis of a tritium monitor for the detection of tritium in a high energy gamma background for Canadian nuclear power workers.

MPM-B.7 Neutron Response and Resolution of the New Tissue Equivalent Proportional Counter System for the International Space Station

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The original Tissue Equivalent Proportional Counter (TEPC) used on the International Space Station (ISS) will be replaced by a new system taking advantage of improved technology including spherical detectors with laminated walls to provide uniform gas gain. Each detector, with its entire preamplifier, will be housed in an independent vacuum chamber filled with propane to simulate a 2 μ site size. The two detectors, differing in diameter by a factor of 3, will cover the dose rate produce by the solar activity. The first stage of the preamplifier was built on a special circuit board made of Rexolite which exhibits excellent electrical properties, contributing to a significant decrease in electronic noise. The noise reduction allows a lower level discriminator setting around 15 eV/μ. An AmBe source is used to characterize the whole system. The neutron drop point resolution is equal to, or slightly better than, the resolution achieved with the high performance proportional counter with helical grids.

MPM-B.8 Response of a Proportional Counter Under Moderate Pressures of Counting Gas in Low Energy Neutron Fields

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When low energy neutrons interact with a medium they generate secondary charged particles which have a range of a few microns in unit density media. Moderate pressures of less than 760 torr (1 atm) of hydrogen containing gases are enough to stop these secondary charged particles. The experimental work described in this paper is a preliminary study of the total energy deposition by low energy neutrons in an in-house built proportional counter (PC) filled with TE gas. The response of the counter at these energies is solely due to the neutron interactions in the sensitive volume of the counter. The McMaster University 1.25 MV double stage Tandetron accelerator was employed to generate low energy neutron fields of mean energies extending from 34 keV to 354 keV from the 7Li target using proton beam energies ranging from 1.89 to 2.56 MeV. The counter was calibrated using a built-in alpha source. The TE gas contains a small quantity of nitrogen which induces the 14N(n, p)14C capture reaction with thermal neutrons generating 580 keV protons. These capture protons are completely stopped inside the gas and a completely resolvable Gaussian peak is observed in the spectrum where the incident neutron spectrum is limited to very low neutron energy. Another group of secondary protons are produced due to the elastic recoil of hydrogen with the neutrons. The maximum energy deposited by this group of neutrons can be inferred from the measured spectrum. Additional information can be extracted from the measured spectrum using available unfolding techniques. The neutron energy spectrum in nuclear reactor environment is mainly limited to below 1 MeV and a negligible portion of the spectrum exists beyond 1 MeV. A proportional counter filled with moderate pressure (~1 atm) of a hydrogenous gas in principle is able to stop the recoil protons generated by low energy neutrons and therefore has the potential to act as a compact energy spectrometer for reactor environments and other low neutron energy applications.

MPM-B.9 Advances in Electron Paramagnetic Resonance Dosimetry with Fingernails

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Accidental radiation dosimetry can be achieved by electron paramagnetic resonance (EPR) using fingernails as biophysical markers for radiation exposure. EPR is well-established for dose measurements in tooth enamel; however, tooth samples are rarely available immediately after radiation accidents. Fingernails samples are, and fingernail EPR dosimetry offers the advantages of having a relatively low dose limit (estimated 1-2 Gy) and simple sampling processing. This makes it possible to complete rapid dose assessments on site if small portable spectrometers are available. The first operational protocol of dose measurements has been developed and tested. The proposed sponge model explains most of fingernails' dosimetric properties. According to this model, a fingernail can be described as spongy tissue, which is deformed at the time of clipping. There is also a noticeable dose dependence variation at different levels of mechanical stress samples. This paper discusses most recent results and perspectives of fingernail dosimetry.

TAM-A.1 EPA Airborne Detection Capabilities

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The EPA Airborne Spectral Photometric Environmental Collection Technology (ASPECT) Program provides airborne ortho-rectified imagery, video, chemical and now radiological information directly to emergency response personnel via a commercial satellite link onboard the aircraft. EPA initiated the ASPECT Gamma Emergency Mapper GEM Project in 2008 to improve its airborne gamma-screening and mapping capability for monitoring any ground-based gamma contamination. The aircraft is equipped with eight 2”x4”x16” NaI(Tl) crystals and state-of-the art signal processing technology. This presentation will cover (1) an overview of the ASPECT program, (2) details on how the aircraft is calibrated to provide ground-based concentrations and exposure rates, (3) minimal detectable activities, and (4) examples from recent deployments.

TAM-A.3 Use of a Helicopter Platform Using a Multiple Sodium Iodide Detector System to Conduct Environmental Scoping Surveys

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The Department of Energy National Nuclear Security Administration maintains an aerial radiation mapping capability for emergency response to radiological incidents. The emergency response radiation mapping mission has many of the same objectives as defined for a scoping survey under the Multi-Agency Radiation Survey and Site Investigation Manual(MARSSIM). The objectives are to identify the radionuclide contaminants, establish radionuclide ratios and map the levels and extent of contamination. The 12- 5 centimeter x 10 centimeter x 40 centimeter sodium iodide detectors carried by the Remote Sensing Laboratory’s (RSL) Bell 412 helicopter record spectra each second with GPS location information. The RSL has developed analysis software and algorithms that convert the gross counts from the detector system to produce data products showing background areas within the survey, areas of gross man-made activity and areas of specific isotopic activity. The presentation will discuss the aerial survey parameters of altitude, GPS location, line spacing and data recording necessary to produce a product that can be used to designate Class 1, Class 2 and Class 3 areas for subsequent ground-based surveys and sampling. In addition, the presentation will include examples of previous scoping surveys to illustrate the advantages of aerial measurements for large scoping surveys.

TAM-A.4 A Methodology for the use of Handheld X-Ray Fluorescence (XRF) Technology and (Multi-Agency Radiation Survey & Site Investigation Manual) MARSSIM Guidance to Characterize Non-Radiological Metals Contamination at Radiologically Contaminated Sites

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Many radiologically contaminated sites also have non-radiological contaminants, such as metals, associated with them. Unlike radiological contaminants which are readily detectable with walkover scans based on MARSSIM guidance, non-radiologic[illegible]ntaminants are not as easily detectable and may [illegible] located with radiological contaminants m[illegible] measurement impossible. With[illegible]thod to detect metals in the field [illegible]an concentration of contaminants [illegible] judgmental sampling locations [illegible]icult. In order to estimate a mean conta[illegible]centration for metals in a Final Status Survey [illegible] unit, or during the characterization process, a handheld X-Ray Florescence (XRF) detector can be employed for rapid detection of metals in the field. Using a Ranked Set Sampling (RSS) design for creating random sampling locations we have developed a method for estimating a mean concentration of non-radiological metals contamination in a given FSS unit or study area. Additionally, XRF scanning similar to walkovers or systematic sampling prescribed under MARSSIM guidance are now possible in order to locate potential judgmental sampling locations to augment the random sampling effort. Combining the random sampling, judgmental sampling, and the scan/systematic data one can draw more confident conclusions about the location and extent of metals contamination in a study area.

TAM-A.5 Development of a Detection Array for Field Work and Instructional Laboratories

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Detector arrays are used in several field applications for radiation detection. Developing a useful array for laboratory or field use sometimes requires custom electronics and software to ensure each detector is properly synchronized and calibrated. Electrical engineers, software engineers, and radiation specialists collaborate to create such systems. Recently developed commercial hardware is now making it easier for independent researchers to link groups of detectors together. Using ORTEC's DigiBASE-E, health physics researchers and students are developing sodium-iodide crystal-based detector arrays that will be used for field work and in hands-on student laboratories. Software is being developed to be used in conjunction with this hardware. This computer program will allow a number of detectors to be linked and have their acquired data synchronized and displayed. This talk will describe this software, the DigiBASE-E and how they are being used in radiation detection research at Texas A&M University.

TAM-A.6 Methodology for Indoor Geospatial Data Capture of Radiological Contamination Using a Robotic Total Station (RTS) Integrated with a Rate-Meter and Represented with Geographic Information Systems (GIS)

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While Global Positioning Satellites (GPS) and geographic information systems (GIS) have enabled surface scans of radiologically contaminated areas to be geospatially recorded and visualized, doing the same inside of buildings has remained elusive. However, taking the same knowledge of GIS and incorporating the use of Robotic Total Stations (RTS), indoor radiological surveys can now be geo-located with real world or Cartesian coordinates and geo-statistical analysis can then be conducted. With this, virtually all aspects of outdoor radiological surveys are now available for use indoors. This can range from surface scans including walls and ceilings, sample layout, hotspot location and relocation, and statistical analysis using the large volumes of data recorded during scanning activities. Various GIS software allows many styles of surface representation such as 2D, 3D, and exploded views which would now be available for indoor scans. As with GPS, the RTS can incorporate multiple types of detectors as needed for different survey requirements. The result of the use of RTS and GIS coupled with rate-meters is a truer representation of indoor contamination and the ability to make more certain decisions based on absolute values as opposed to ranges.

TAM-A.7 Final Status Survey Application of Ranked Set Sampling for Hard to Detect Radionuclides

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The MARSSIM (Multi-Agency Radiation Survey and Site Investigation Manual) guidance has become widely implemented at sites undergoing decommissioning during the conduct of final status surveys. One of the fundamental tenets of the MARSSIM is the integration of statistically-based sample population with direct surface scans for demonstrating compliance with release criteria. Within Class 1 survey units, the required number of samples is directly coupled to a required scan MDC (minimum detectable concentration). The required scan MDC ensures localized areas of contamination in excess of the release criteria (hot spots) are identified, while the sampling provides the quantitative values for estimating the mean residual concentrations. For soil survey units, this approach in most cases assumes the presence of a detectable gamma-emitting radionuclide is present as a contaminant and will provide detection capability of hot spots. Such a gamma emitter may be used as a surrogate when other non-gamma emitting contaminants are present, commonly referred to as hard-to-detect (HTD) radionuclides. Many sites do not have a gamma emitting contaminant present or else there may be no consistent ratio between the potential surrogate contaminant and the HTD. This paper discusses the preliminary field studies and possible application scenarios for using ranked set sampling (RSS) as a means for evaluating survey units for HTDs and demonstrating release criteria compliance. The limiting factors and necessary assumptions for applying RSS are also presented.

TAM-A.8 Revision 2 to Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)

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The state of the science in radiation measurement has improved and the type and number of radiation measurement methods have increased since MARSSIM (Multi-Agency Radiation Survey and Site Investigation Manual) was first issued in 1997. In addition, "lessons learned" over the past thirteen years of use have yielded suggestions for improvements to the MARSSIM process. As a result, the MARSSIM Workgroup intends to issue Revision 2 to MARSSIM to incorporate these changes. The MARSSIM Workgroup seeks input from manual users on proposed revisions, and seeks to keep users informed on updates to the manual. Proposed revisions and a request for input to the revision process are outlined, including a methodology for submitting written, electronic, and verbal comments, as well as contact information for further questions.

TAM-A.9 A MARSAME-Based Release of Hanford Railroad Rails Using Standard Field Radiological Instruments, Computer Data Collection and Analysis, and Release Practices

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The radiological survey methodology and release criteria used to release approximately 8,000 railroad rails for reuse from the Hanford site are presented. The material release of metals is permitted under U.S. Department of Energy (DOE) Order 5400.5, Radiation Protection of the Public and the Environment, as long as reuse is consistent with its original intended purpose. Over 160 kilometers (100 miles) of railway was constructed and used to move supplies, equipment, and reactor fuel around the Hanford site until the railway system was shut down in 1998. Approximately 48 kilometers (30 miles) of Hanford rail, weighing about 3.4 million kilograms (8 million pounds), was targeted for this release survey. This rail has substantial value to maintain existing railroads through reuse and reduces the amount of useful material that would otherwise be transported to and disposed as waste in Hanford's Environmental Restoration Disposal Facility (ERDF). DOE Office of Health, Safety and Security document, DOE/HS-0004, Multi-Agency Radiation Survey and Assessment of Materials and Equipment Manual (MARSAME) was used as guidance in determining the classification and required level of effort for the disposition survey. Instrumentation and protocols were developed to allow for in situ monitoring of the rail. The complete disposition survey of the rails consisted of initial, follow-up, and confirmatory surveys following MARSAME guidelines. The use of this methodology ensured the materials were safe for release for the purpose of reuse and reduced valuable materials from being disposed as waste. In turn, the objectives of protecting the health and safety of workers and the public, and environmental sustainability through material reuse, were achieved through the application of these methods.

TAM-A.10 Field Experience with a Portable, Field, Alpha and Photon Spectrometer for the Clearance of Property with Contaminated Surfaces

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The radiological clearance of property from the Department of Energy's Hanford Site is often complicated by the presence of naturally-occurring, alpha-emitting radionuclides, which are indistinguishable from manmade, alpha-emitting radionuclides when using portable survey instruments. To address this problem, an alpha and photon field spectrometer, and applicable protocols for its use, have been developed. The conceptual design of this spectrometer, and initial testing of the spectrometer and protocols for measurements of known, natural radioactivity on metal surfaces were described in a presentation at the 2010 Health Physics Society's Annual Meeting. This presentation is a follow up to that presentation and will focus on actual field use of the spectrometer. A brief review of the information presented at the 2010 Annual Meeting will be provided. The presentation will then describe specific use of the spectrometer and protocols for the radiological clearance of personal property. The results and interpretations of the spectrometer measurements, and how these results are incorporated into the final clearance decision will be described.

TAM-A.11 Detection of Pu-239 Beneath a Monolayer of Stainless Steel Supporting Free Release of Equipment From the Z Machine at Sandia National Laboratories

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The Z Machine is used to conduct change of state physics experiments. When radioactive material is the sample being analyzed an enclosure is used to limit spread of contamination to the machine interior. If the enclosure does not work properly, machine parts could be coated with a mixture of the radioactive material and the metal structure supporting the sample. The most recent tests involved Pu-239 as the radioactive material. To support release of equipment that could be potentially contaminated with Pu-239, a method was developed to detect the L and K shell x-rays emitted by Pu-239 and the 60 keV photon from Am-241 decay. The method used a Ludlum 44-19 probe with a 2mm thick NaI detector. Measurements were taken on Pu-239 and Am-241 sources using an Aluminum absorber set to estimate the density thickness needed to totally absorb the low energy emissions from the Pu and Am contamination. Measurements of both radionuclides were detectable through the absorbers. The efficiency for Plutonium detection was such that no certainty could be given to the results. The Americium efficiency and detection capabilities proved that, with sufficient in growth, we are able to detect the parent's (Pu239) presence through thin layers of deposited metal. Results of this work are presented in the paper.

TAM-A.12 Successful Implementation of Subsurface Soil Derived Concentration Guideline Level Methodology to Achieve Compliance with Unrestricted Release Criteria

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Approximately 24 acres of a Nuclear Regulatory Commission (NRC)-licensed nuclear fuel manufacturer's site has been undergoing decommissioning for several years. The objective of the decommissioning effort is to demonstrate compliance with the facility's Decommissioning Plan (DP), and to meet the radiological release criteria for unrestricted use in accordance with 10 CFR 20, Subpart E. Due to physical limitations, as well as escalating excavation, transportation, and disposal costs of radiologically-impacted soils, the site implemented an innovative and proprietary subsurface soil derived concentration guideline level (SS-DCGL) methodology. The SS-DCGL methodology is a dose-based method that can be readily applied to many sites with residual radioactivity in subsurface soils. This innovative technique also establishes a rigorous set of criterion-based data evaluation metrics (with analogs to the MARSSIM methodology) that are used to demonstrate compliance with SS-DCGLs. By employing the proprietary SS-DCGL methodology, the licensee has been able to achieve NRC acceptance and confirmation that significant areas of its site meet the radiological release criteria for unrestricted use. This paper presents the process of designing and implementing a Final Status Survey using this SS-DCGL methodology, demonstrating compliance with SS-DCGL data evaluation metrics, and achieving regulatory confirmation for the unrestricted release of the area.

TAM-B.1 The Psychology of Radiation Measurements

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Good and defensible radiation measurements require several steps: 1) deciding what to measure (contamination or exposure), 2) choosing the proper instrument for the intended measurement, and 3) using the instrument properly. Assuming you have accomplished these three steps appropriately (there are countless pitfalls in these steps), you now have measurements to interpret. Several questions now arise: 1) what do the numbers mean, 2) are the measurements defensible, and 3) how much would you be willing to commit for resources on the basis of these measurements? This is where the psychology of radiation measurements could become very significant. Interpretation of radiation measurements may have as much to do with attitudes and perceptions of radiation risks as it does about technology. The very same measurement may have a wide variety of meanings to different people. For example, a technician at a nuclear plant saw a small blip on the readout of a whole body scan of a worker and announced, "Wow, we have a hot one here!" While the blip was technically interesting, although of no health significance, the worker heard the result as a matter of life and death. Litigation followed. A worker at an industrial facility observed the RSO taking readings with a Geiger counter and saw the meter go off scale. That was enough information for this worker to start an uproar that eventually involved several hundred other workers, the union, and management. A common aspect of each of these scenarios is the assumption that if radiation is measureable, it must be bad. Interpretations of measurements become a matter of responding to fears of radiation. One person defending their conservative decision said, "Why take chances?" There are two axioms on measurements, 1) measurements have no meaning until interpreted and 2) measurements only have meaning in terms of how they are interpreted."

TAM-B.2 Developing an Environmental Monitoring Program for Radiological Operations at a "New" U.S. DOE Site

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The U.S. Department of Energy (DOE) manages two facilities in southeastern Washington. The DOE Office of Environmental Management (DOE-EM) oversees environmental restoration and waste management activities at the Hanford Site. The DOE Office of Science (DOE-SC) Pacific Northwest Site Office (PNSO) oversees the Pacific Northwest National Laboratory Site (PNNL Site), a research and development facility located just south of the Hanford Site 300 Area. Facilities on the PNNL Site are new within the last 15 years and were built on raw land with no historic Hanford legacy facilities. PNSO is in the process of formalizing the environmental programs at the PNNL Site, independent of those of the Hanford Site EM environmental programs. The process of developing air monitoring and other associated environmental programs for radiological operations at this "new" DOE site is discussed. Both federal and Washington State requirements are addressed with regard to anticipated stack emissions characterization, offsite monitoring, and measurement. Efficiencies were pursued to develop a monitoring program consistent with long-standing Hanford Site programs, while establishing them as independent entities. Programs were developed under the U.S. Environmental Protection Agency's Data Quality Objectives report approach.

TAM-B.3 Personal Dose Equivalent Conversion Coefficients for Electrons, Photons, and Positrons

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The personal dose equivalent is the quantity recommended by the International Commission on Radiation Units and Measurements to be used as an approximation of the protection quantity Effective Dose when performing personal dosimeter calibrations. The personal dose equivalent can be defined for any location and depth within the body, typically, the trunk where personal dosemeters are worn. In this instance a suitable approximation is a 30 cm X 30 cm X 15 cm slab-type phantom. For this condition the personal dose equivalent is denoted as Hp,slab(d) and the depths, d, are taken to be 0.007 cm for non-penetrating and 1 cm for penetrating radiation. In operational radiation protection a third depth, 0.3 cm, is used to approximate the dose to the lens of the eye. A number of conversion coefficients for photons, electrons, and positrons are available for incident energies up to several MeV, however, data to higher energies are limited. In this work conversion coefficients up to 1 GeV have been calculated for Hp,slab(10) and Hp,slab(3). For Hp(0.07) the conversion coefficients were calculated, but only to 10 MeV. The conversion coefficients were determined for discrete incident energies, but analytical fits of the coefficients over the energy range are provided. The conversion coefficients for the personal dose equivalent are compared to the appropriate protection quantity, calculated according to the recommendations of the latest International Commission on Radiological Protection (ICRP) guidance.

TAM-B.4 Microscope Image Analysis of Immune-Fluorescent Foci as a Biodosimeter for Assessing Neutron-Induced Injury

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DNA double strand break (DSB) has been identified as the most important initial damage caused by ionizing radiation. Unrepaired or misrepaired DSBs can lead to mutations, chromosome aberrations, genomic instability, cell death, or cancer. Recently, a new technique has been developed to assess radiation-induced DSBs via observing microscope images of the immuno-fluorescent foci formed due to accumulations of proteins that are involved in DSB sensing and repair pathways. The most widely studied protein is g-H2AX. Other proteins include NBS1, MRE11, Rad50, and 53BP1, etc. A promising application for this new technique in health physics is that it may serve as a "biodosimeter" to assess the radiation injury of an individual shortly after a radiological incident. In this case, the lymphocytes of the individual's blood can be used for foci counting. Because high-LET radiations (e.g. heavy ions, alpha particles and protons) tend to produce a streak of several DSBs (or foci) in a single radiation track, one may further develop this technique to allow assessment of neutron-induced damage in a mixed field of neutrons and gamma rays. That is, neutron-induced DSBs can be extracted from a background of foci-filled image by identifying the streaks of foci representing the recoil proton tracks. To confirm this idea, a series of in-vitro neutron irradiation experiments using a medical grade 252Cf source is currently being carried out at Georgia Tech. The experiment involves irradiating, fixing, and analyzing a total of 16 cell sample dishes to cover a wide range of dose, irradiation time, and DSB repair time. The neutron and gamma-ray dose rates are estimated to be 0.5 Gy/hr and 0.3 Gy/hr, respectively. The cells used are U87 glioblastoma cells. The preliminary results indicate that 1 Gy of fission neutron dose produces approximately five proton tracks in a cell nucleus and that the proton tracks can be successfully identified in a microscope image.

TAM-B.5 New Materials for Individual Emergency Dosimetry using Optically Stimulated Luminescence

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Optically stimulated luminescence (OSL) is one of the most sensitive methods capable of detecting radiation-induced centers in many substances. Due to this feature, OSL is proposed for use in individual emergency dosimetry through measurements of the OSL signal from suitable materials that were at/on an individual during the exposure event. Because an optimum material has yet to be found, testing of new materials remains a continuing task. Two new potential emergency OSL dosimetry materials – business cards and plastic buttons – were tested in the present study. Both materials can easily be found on many people in locations protected from external light exposure. Materials were studied with a research-grade OSL reader that uses 490-520 nm light for stimulation of the OSL from the samples and collects emission in the ultraviolet wavelength range. Business cards from different suppliers and from private collections and miscellaneous buttons from local stores were used in the study. Exposure of the samples to different doses below 10 Gy was done using a beta source. Basic characteristics studied were: (1) dose-response; (2) variability of radiation sensitivity and (3) stability of the dosimetric signal after exposure. It was found that only 25-30% of business cards and 20-25% of buttons are sensitive enough to be used as emergency OSL dosimeters. All sensitive samples showed a linear dose-response in the studied dose range. Radiation sensitivity varied significantly from sample to sample; corresponding values of the minimum measurable dose were from 30 mGy to about 1 Gy. OSL signals faded almost completely during a few hours after exposure if samples were stored under normal room light. For samples stored in the dark the fading was approximately 50% during one month after exposure. Our findings demonstrate that business cards and plastic buttons could be used as suitable OSL dosimeters for application in emergency exposures, especially for triage.

TAM-C.1 Calibration of Radiation Measurement Instruments with the Help of Primordial Radioisotopes

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There is an inherent disadvantages of conventional check sources for radiation detectors: Each source is individual and unique with regard to its exact activity and emission rate, so that these data, including correction for the radioactive decay, always need to be known and recorded in order to verify the instrument performance. Large area sources may furthermore exhibit different degrees of uniformity in respect to the surface emission rate and the thin surface layer is always a delicate part of the source – especially for routine use in the field. In order to overcome these issues, recently an alternative design was developed on the basis of a high density lutetium-oxide ceramics. Natural lutetium contains the primordial radioisotope Lu-176 (half life 36 billion years) with an abundance of 2,6 %. The resulting activity concentration in Lu_2O_3 is about 50 Bq/g. Lu-176 emits beta radiation with a maximum energy of 600 keV and several gamma energies up to about 300 keV. The presentation compares these properties with those of alternative natural occurring radioactive materials and typical conventional radionuclides. In respect to lutetium, the use of an oxide based on an element containing the radioisotope in its natural abundance offers the unique advantage of an absolutely constant and uniform surface emission rate. This contribution discusses the specific radiation properties of the innovative material and the practical implications in the field of radiation protection instrumentation. Among other advantages, any external radiation or ingestion risk in conjunction with the usage of these adapters can be excluded. The main practical advantage can be seen in the possibility of a direct response comparison of measuring instruments that are not at the same location or facility.

TAM-C.2 Significant Improvements in Accuracy of Beam Type Calibrators

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Calibration facilities using beam type calibrators frequently depend upon the manufacturer's data to determine the exposure rate at various distances. The manufacturers usually provide a NIST traceable measurement of the exposure rate at a specified distance, typically one meter or a calculated exposure rate at one meter based upon the source manufacturer's stated activity and the published source strength for the radionuclide in the source. Calibration laboratories use the inverse square law to calculate exposure rates at other distances. Exposure rates provided by manufacturers and exposure rates calculated form the source activity have had significant error. Using the inverse square law provides the relationships between the exposure rates and distances in free air. In most real world calibration facilities, free air conditions do not exist and scatter plays an important role in causing deviations from the inverse square law. Calibrators with wide beam cones suffer the most from this effect. Improvements have been achieved by measuring the exposure rates at least one distance using a NIST calibrated transfer instrument to eliminate the error that would result from using the manufacturer's data. Relative exposure rates were measured along the calibration range to characterizing the relationship between the exposure rates at known distances. These exposure rate measurements enabled correction for deviations from the inverse square law and eliminated errors due to scatter. Using these two techniques, a NIST transfer instrument to determine the exposure rate at a specific distance and characterization of the deviations from the inverse square law to correct for scatter has eliminated these errors in two calibrators.

TAM-C.3 Calibration of Germanium Gamma Spectrometry Systems for Radiological Surveillance by Means of Monte Carlo Calculations

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Laboratory of Inorganic and Nuclear Chemistry at Wadsworth Center, New York State Department of Health is mandated to perform radiological surveillance for New York State purposes. The surveillance involves measurements of environmental samples to satisfy National Drinking Water Regulations, monitoring of nuclear power reactors, waste repositories, hospitals, research institutions, etc., to protect population of New York State. Gamma-ray spectrometry is one of the principal techniques which we use for monitoring radioactive samples. We operate 6 coaxial Ge detectors and 2 Ge-Well detectors. We have a total of 7 geometries for coaxial detectors and 5 geometries for the well detectors. The preparation of calibration standards for each photopeak in each sample matrix and geometry of interest has become expensive, manpower and time consuming, and has as well created excessive radioactive waste. In this presentation we describe the use of Monte Carlo calculations to eliminate some of the matrix preparation burden and to apply difficult to measure corrections to the data. A method of transferring photopeak efficiencies measured on one detector and specific sample parameters (matrix composition, density, volume, and geometry) to other detectors and sample parameters by means of Monte Carlo calculations has been developed using existing computer codes. The calculated efficiencies obtained compare well (within a few percent) with measured data. We describe calculations of gamma self-attenuation corrections to the photopeak efficiencies, for matrices such as charcoal and soil. In addition, we describe calculations of coincidence-summing corrections to the photopeak efficiencies, for radionuclides of interest, such as Co-60, Y-88, Cs-134,Eu-154,and Eu-155. Monte Carlo techniques have been approved in the ANSI standards as well as by the NRC regulations.

TAM-C.4 Determination of the Optimum Container Diameter for the Gamma-Ray Assay of Laboratory Samples

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In a high-throughput gamma-ray assay laboratory environment, the reduction of measurement time necessary to achieve a particular minimum detectable activity or activity uncertainty can represent a real and significant cost savings. One method to reduce sample measurement time is to maximize the efficiency of the setup by optimizing the geometry of the sample. We have recently performed a series of studies using physically-based mathematical efficiency modeling to determine the optimum measurement geometry for a given sample volume and germanium detector combination. As part of this study we computed the efficiency of germanium detectors of different types for cylindrical sample sizes of different volumes and container diameters. We studied detectors from small (40 cc) to large (250 cc) volume and different form factors from high aspect-ratio planar to medium aspect-ratio coaxial types. For each germanium detector type and sample volume we determined the container diameter that produces the greatest efficiency. From this study we have determined that the optimum diameter of a cylindrical container for a given sample volume follows a simple power-law relationship that is relatively independent of the detector type. We will present the detailed results of this study as well as show the limits of the power-law approximation to determining the optimum container diameter to sample volume.

TAM-C.5 The Applicability of Non-Uniform Matrices for Gamma Spectroscopy Calibration of Uniform Matrices with the Same Average Density

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For the efficiency calibration to accurately represent the sample, then both must be obtained with exactly the same shape, density, and composition. It is obvious that differences of material composition and density between the sample matrix and the reference calibration matrix can cause accuracy errors. We have previously shown that non-uniform distributions of radioactivity can cause accuracy errors. But what about the situation where the concentration of radioactivity is uniform, but the density of the matrix is not uniform - i.e. has clumps and voids? How accurately can one measure a container of radioactive pebbles or apples? With mathematical calibrations it is easy to create a calibration for a uniform matrix with the exact chemical composition of the pebbles or apples. But even with mathematical calibrations it is common to assume that the average container density accurately represents this non-uniform density situation. Computations using the ISOCS/LabSOCS software were performed to assess the validity of this assumption. A 500cc container filled with water was used as the reference calibration. A series of other calibrations was performed where various numbers and sizes of spheres of "water" were distributed in a random pattern within the container. The density of these "water" spheres was adjusted so that the average container density was 1.0. The size of these spheres was varied between 1mm and 20mm. The number of spheres varied between 10 and 1000. The density of these spheres varied between 3 and 90. Efficiency calibrations were performed at low energy [60 keV] and high energy [1000 keV]. The data show that the lump efficiency is lower than the uniform efficiency. But as long as the lump density is less than about 2x the average density [60 keV] and less than about 3x the average density [1000 keV] the bias is less than 10%. Therefore in most cases, the use of the uniform efficiency is justified.

TAM-C.6 Gamma Spectroscopy Sample Geometries that Minimize Sample Preparation, Minimize the Number of Calibrations Necessary, and Minimize Calibration Uncertainty

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The conventional process is to calibrate for several fixed geometries [sample size, sample material, distance to detector, …] and then prepare all samples to fit these pre-defined conditions. One problem is that the sample preparation is labor-intensive which frequently is the major component of a sample assay cost. Another problem is the time delay to obtain the final sample results. Both of these are important in normal operation of an assay laboratory, but are expecially problematic under emergency response conditions. This presentation will show several different sample and detector geometries that greatly minimize the number of sample geometries necessary to support with calibrations, and the sample preparation labor, which still providing accuracy that is probably acceptable for most bioassay and environmental assays.

TAM-C.7 Gamma Spectroscopy Counting Geometries that Can be Used for a Wide Range of Sample Conditions with the Same Efficiency Calibration

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Accurate gamma spectroscopy requires accurate efficiency calibrations. This means that the calibration must be performed with the same sample size, shape, material composition, and density as the unknown; and these must be in exactly the same container and distance from the detector as the unknown. That concept works quite well, especially now that mathematical efficiency calibrations are available to do this quickly and accurately. But what if the actual sample composition is not well known to the analyst? What if the sample shape and size vary significantly? These variabilities were evaluated using the ISOCS/LabSOCS mathematical efficiency calibration software to determine various counting geometries where the efficiency changes very little over wide ranges of sample conditions. One solution is to keep the distance constant between the detector and the back of the sample. This allows wide variations of sample size [100:1] with only minimal change in efficiency [20%]. Another solution is to use Massimetric efficiency calibrations [infinite thickness]. This method can allow sample matrices and densities to be quite variable, yet still achieve efficiency calibration accuracies in the 5-15% range. The use of these methods can greatly simplify the necessary sample preparation, but still achieve acceptably accurate results.

P.1 Developing and Implementing a Joint Health Physics Technician and Managers Program at Orangeburg-Calhoun Technical College and South Carolina State University

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With South Carolina being among the top nuclear power producers in the United States, ranking third in nuclear capacity and nuclear generation, and combining the fact that the state has an especially significant role in low level waste disposal – there is a major need for trained personnel skilled as Health Physics/Radiation Control technicians within the state of South Carolina. To address the demand for such skilled workers a partnership was formed between Orangeburg Calhoun-Technical College and South Carolina State University to develop a coordinated health physics career tract for health physics technician (associate degree) with a clear transfer pathway without loss of credit from the two year college to the four-year university health physics managers program (baccalaureate degree). As a result, the development of such a program will accomplish a unique 2+2 program for health physics technician to health physics manager. Supported under NRC Grant Number: NRC-38-10-977

P.2 Utilization of Two New Executable Computer Codes for Confidence Intervals, Decision Levels and Detection Limits when the Sample is Counted an Integer Times Longer Than the Blank

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Frequently confidence intervals, decision levels, and detection limits are determined using the Gaussian distribution, which has a continuous, symmetric probability density distribution. Outcomes of measurements in radioactivity may be limited to an integer number of counts, so a more proper methodology to determine confidence intervals involves discrete probability, not continuous probability. In many situations both the blank count and the sample contribution to the gross count are well approximated by discrete Poisson distributions, which are not symmetric; this assumption is utilized throughout this paper. The ratio of the sample count time to the blank count time is taken to be an integer IRR, and the net count is denoted by OC. The expected blank count in the sample count time, B, is assumed either known or well known. In the well known case, the net count is taken to be the difference between the gross count, a random variable, and a constant equal to the well known blank count; otherwise the net count is the difference between two random variables, the gross count and the product of IRR with the blank count. When OC is the difference between two random variables, the probability density function for OC is given in the proceedings of the 2009 Midyear Meeting of the Health Physics Society. Confidence intervals are determined by mimicking a standard approach in discrete probability. A C++ computer code computes confidence intervals and detection limits for the expected net count. A separate code computes decision levels. Executable versions of both codes have been developed which are expected to run on most Windows based computers.

P.4 Optimization of Plastic Scintillator Thicknesses for Online Beta Detection in Mixed Fields

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For efficient beta detection in a mixed field beta gamma field, a Monte Carlo simulation model has been built to optimize the thickness of plastic scintillator, used in whole body monitor, The simulation has been performed using MCNP/X code and different thicknesses of plastic scintillator starting from 0.15 to 0.6 mm. The relationship between the thickness of the scintillator and the efficiency of the detector has been analyzed. For 0.15 mm thickness an experimental investigation has been conducted with different beta sources at different positions on the scintillator and the counting efficiency of the unit has been measured. Evaluated data along with experimental ones will be discussed.

P.5 Assessment of Annual Effective Dose from, and in Soil and Their Effect on Human Health

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There are always ionizing radiation sources in our environment and they can transfer to human being via food chain. Soil-plant-man transfer of radionuclides to a human being is recognized as one of the major pathway for transfer of radionuclides.The ionizing radiation can affect the human health and the life of other organizes living things in short time, especially when the dose of radiation exceed the ICRP standard. Average concentrations of natural radionuclides and the annual effective ingestion dose matter were not known in soil for Eilam city residents.Therefore this study was aimed to survey the safety of such materials and health promotion for the human being. Materials & Method: 23 different places in Eilam city were chosen for sampling from soil.The concentration and type of radionuclide were determined. Sieving, drying and mixturing were amongst the method utilized for suitable preparation of the materials. In this study, it has been measured the concentration of, and radionuclides in 23 samples of soil in Eilam province using gamma spectroscopic system(HPGe). The results have been compared with the reference values and other measurements in other countries. Results: The results show no existence of any artificial radionuclides. However there were some natural radionuclides such as, in different dosages in the examined samples. Conclusion: The study confirmed that there is no risk of radio nuclides exposure with regard to the soil in Eilam.

P.6 Middle East Radiation Measurements Cross Calibration Workshops

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All countries in the Middle East have nuclear monitoring and measurement capabilities associated with nuclear power and research reactors, and with radioactive sources used in medicine, commerce, and industry. Detecting the presence of radioactive sources, preventing the illicit use of radiological materials, responding to accidental radiation releases, and disposing of radioactive sources safely are common concerns. Improving and standardizing nuclear monitoring and measurement capabilities in the Middle East are essential elements of developing an approach to such concerns. The Radiation Measurements Cross Calibration (RMCC) group conducts annual workshops in the region to discuss the results, identify areas where increased technical cooperation would be beneficial, and recommend future activities. These practical workshops are designed to encourage communication among the regional radiological laboratories, develop internationally recognized laboratory standards, and provide training on relevant topics such as laboratory management, quality assurance, and gamma spectroscopy. The workshops provide opportunities for the regional participants to exchange insights into the radiological measurement problems they face in their home countries and build up the regional capacity to address these issues. Benefits from the RMCC project include increased confidence in data quality across the region, availability of a network of qualified labs for radiological measurements, and improved scientist-to-scientist communication. The project builds up the capacity in the region to produce reliable radiological data and will provide a mechanism for sharing of agreed upon information. This will enable scientists in the region to work cooperatively to create indigenous solutions to the problems in the region. The effort builds confidence by encouraging technological transparency in the region and fosters the development of a network of scientific experts in the region.

P.7 A Detector for Simultaneous Beta & Gamma Spectroscopy

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A triple-layer phoswich scintillating detector has been developed for simultaneous spectroscopy of mixed-field beta and gamma radiation in real time. Housed beneath an aluminized Mylar window, a thin layer of BC400 scintillating plastic captures most lightly penetrating electrons of less than 100 keV. A thicker inorganic crystal of CaF2(Eu) provides an additional barrier to capture electrons up to about 3.2 MeV. The third scintillator is used for spectroscopy of gamma and x-ray photons and is constructed of an inorganic NaI(Tl) crystal. This crystal is isolated from the first two scintillators by an optically transparent quartz barrier that prevents moisture degradation. A single photomultiplier tube detects the light signal from the three phosphors and provides output for processing with a single coaxial cable. Each scintillator has unique timing characteristics to facilitate discrimination of the originating light pulses using state-of-the-art digital processing techniques. Incident beta and gamma radiation may deposit its energy into the detector in one of seven distinct configurations, each producing a unique output pulse. Probabilities of these energy depositions were tabulated through MCNP analysis. For example, interactions in only the first layer are most likely caused by a low-energy electron, with only a small fraction resulting from gamma interactions. Energy deposited into both the first and second layer is almost certainly the result of a moderately energetic beta particle. Energy deposited only in the third crystal is likely the result of a gamma or x-ray photon. This detector, used in conjunction with the co-developed digital pulse processing system, has a broad range of applications that include waste management, non-proliferation, radiological incident response, worker safety, site cleanup, and environmental assessment.

P.8 Radio Frequency Identification with Radiation Monitoring Ability

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A Radio Frequency Identification (RFID) system, called ARG-US, has been successfully developed. The system is being deployed at several DOE sites to modernize the lifecycle management of nuclear materials to achieve improved accountability, security and safeguards, cost effectiveness, and — equally importantly — ALARA (as low as reasonably achievable) practices. The current MK-II tags are equipped with seal, temperature, humidity, and shock sensors to monitor the state of health of the packages on which they are mounted. A software suite with easy-to-use graphical user interface extracts, autonomously or on demand, sensor data, event history, and content manifest information through wireless means (433-MHz RF) and disseminates the encrypted information over secured Internet. In conjunction with global positioning system (GPS) and satellite communication, the system can track and monitor packages in transport. In the next-generation MK-III tag platform a radiation dosimeter is incorporated. The detector modules are modified compact personal dosimeters. The unit can monitor gamma radiation from the tagged drum and its neighbors. A custom-built carrier board for the dosimeter manages data processing, alarm initiation, power management, tag interface, and other functions. To accommodate the hardware changes, the ARG-US software suite is being modified to handle and display pertinent information, such as dose rate, cumulative dose, deviations, and areal 3-D dose-field map. In storage vaults and processing facilities, for instance, the dosimeter-equipped tags can be used for area radiation monitoring, array and processes configuration control, and supplemental tamper indication. Similar benefits can be expected in transportation applications. When developed, the dosimeter-equipped tags may prove useful beyond nuclear materials monitoring – facilities with radioactivity present can benefit from its remote and continuous monitoring capabilities as well.

P.9 Making It Real - Building a Technical College Radiation Protection Technology Program from Scratch with State of the Art Survey and Detection Equipment

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This poster will provide information on radiation survey and detection equipment utilized as part of the Aiken Technical College Radiation Protection Technology (ATC RPT) Program. ATC's RPT program began in the Fall of 2008 with 40 students and has grown to an enrollment of about 140 for the Fall of 2010. The program offers an associate in applied science degree in radiation protection technology as well as a RADCON certificate to eligible students with prior AS or BS degrees in related technical fields. From the program's inception, our industry partners, who serve on our program's advisory committee, have emphasized the importance of a relevant program of study whose graduates are ready to be employed with only site-specific training needed to complete their new hires' training. Achieving that goal necessitated the purchase of state-of-the-art equipment in order to provide a top-notch, real world education. Topics covered will include: prioritizing which equipment to obtain within a budget, obtaining funding to purchase equipment, loaned and donated equipment from industry partners, and a review of equipment we currently have and hope to acquire to continue to develop and maintain a Center of Excellence in scenario-based RPT education.

P.10 Back-Projected Radiation Analyzer and Cell Evaluator (BRACE) for Hot Cell Characterization

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For a generic radiation characterizer, using a detector material instead of an electronic detector, a method of reconstructing a contamination profile of the characterized environment is not a trivial task, particularly for complex area sources or nearly even contamination. Detector materials that provide, for example, optical data as output are typically examined by a human, similar to how an x-ray image of a patient is examined by a physician. The human examining the output must make the decisions as to where there is radiation exposure to the material and determine the direction of the source from the optical scan data. Although this method is viable for simple point sources, assuming the number of point sources is limited, it is desirable to produce an automated method of extracting contamination location, energy, and intensity from the data provided by an exposed detector material. The methodology presented in BRACETM, is an automated method of extracting source location, energy, and intensity from a collimated exposed detector material. The methodology will work for nearly any detector material which can be read or scanned into a computer data file, providing a 3D or 2D matrix of exposure values as integers or floating point numbers.

P.11 Introducing Students to Detection: Aluminum Decay Labs at Oregon State University

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Each year about a thousand college students, high school students, and members of community groups such as the Boy Scouts receive hands-on experience in radiation detection through a lab measuring the half-life of Al-28 at Oregon State University. This lab is incorporated in all general chemistry classes which means a significant portion of Oregon State's student population receives a tour of the OSU Radiation Center's TRIGA Mark II research reactor and a chance to use simple detection systems to calculate the half-life of Al-28. These labs provide a basic understanding of radiation and radiation detection to a broad cross section of society; an understanding many of them would be unlikely to encounter in their educations without this experience. The lab begins with a grounding in ionizing radiation and a discussion of the natural background level of radiation; this is followed by a discussion of radiation detection and an explanation of Geiger-Müller counters and the specific setup available in the lab. While students tour the reactor, aluminum foil samples are sealed in polyethylene vials and irradiated for one minute with reactor power at 100 watts. Students return to the lab with the aluminum and track the decay of Al-28 to stable silicon, computing the half-life. These labs are tremendously valuable for educating a wider audience as to the existence, nature, and relative risks of ionizing radiation and should be considered as an outreach tool by any institution or organization with the resources to run them.

P.12 Use of Helicopter Platform for Large Area Radiation Surveys

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With a lineage tracing back to 1958, the Department of Energy National Nuclear Security Administration has maintained an aerial radiation detection and mapping capability for emergency response to radiological incidents. The objectives of this capability are to identify radionuclide contaminants, establish radionuclide ratios and map the levels and extent of contamination. The 12 - 5 centimeter x 10 centimeter x 40 centimeter sodium iodide detectors carried by the Remote Sensing Laboratory's (RSL) Bell 412 helicopter record spectra every second with GPS location information. RSL has developed analysis software and algorithms that convert the gross counts from the detector system to produce data products showing background areas, areas of gross man-made activity and areas of specific isotopic activity. This capability, maintained on both east and west coasts, is also ideally suited for large area and baseline surveys of cities, nuclear power plants and special events.

P.13 Digital Processing of Multi-Component Signal Pulses

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A customized Digital Pulse Processor (DPP) has been developed for simultaneous spectroscopy of beta-particles and gamma-rays. Radiation signal pulses from the photomultiplier tube (PMT) of a specialized triple-layer phoswich detector are digitized using a fast analog-to-digital converter (ADC) with a 200 MHz sampling rate and 12-bit resolution and transferred to a host PC for digital pulse shape analysis and simultaneous spectroscopy. Digital and logic functions, such as over-range rejection, trigger control, partial pile-up rejection, and a circular buffer, are implemented in a field programmable gate array (FPGA). All communications, such as control commands and data transfers between software and the DPP, are performed via a high-speed USB 2.0 interface. A software algorithm was developed to control the DPP and also to characterize beta/gamma induced pulses. When a valid event occurs, the pulse stored in the circular buffer (with a duration of 10.24 microseconds) is transferred to the PC for further digital signal processing. Our custom algorithm processes the radiation pulse and if the pulse meets additional criteria, one of the energy spectra, beta or gamma, is updated.

P.14 Radially Dependant Directional Shield (RDDS) Used for Hot Cell Characterization

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The goal of RDDS is to provide a means of collimating a gamma ray source through some metallic shield such that the detector or material behind the shield is exposed to increased gamma radiation only when a source is within the engineered field-of-view (FOV) of the collimation hole. This is achieved by employing a unique shape to the collimation hole, providing a varying amount of attenuating material between the source and detector material depending on the angle of incidence. The RDDS was designed to alleviate the problems of increased noise through a collimated shield beyond its desired FOV by introducing additional attenuating material between the detector material and the source at angles beyond the desired FOV. The shape of the additional attenuating material is dependant on collimator hole diameter, solid shield shape (plane or sphere, etc.), and desired FOV of the collimator hole. To design the shape of the RDDS collimator hole, choose a collimator hole diameter which provides the line-of-sight FOV desired. This line-of-sight FOV is required such that very low energy gammas will still generate a dose rate into the detector material up to the maximum desired FOV. Next, starting at the 2D center of the collimation hole, determine the length of the "air gap" through the collimation hole at angles beyond the desired FOV. This is the amount of additional attenuating material to be added to the outside of the collimation hole. This additional material is added radially around the collimated hole, producing a symmetric shape around the hole. With RDDS in place and with a source outside the desired FOV, detector material that is in-line with the collimator hole and source has a similar amount of attenuating material to the source as the neighboring material under the solid portion of the shield. The final result is that detector material is not exposed to a dose higher than the noise that penetrates the solid portion of the shield until a source is within the desired FOV of the collimation hole.

P.15 Bayesian Analysis to Produce Usable Measurement Results for Everyone, Even Those with Negative Values

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When measurements are available for a group of people, the sample variance of the measurements has contributions from both measurement error and the natural variation of the measurands (the “true values”) within the group. We consider independent measurements with good estimates of measurement uncertainty. We disaggregate the two sources of variance and produce a lognormal probability density function (PDF) of measurands for everybody in the population with the same average as the measurements. With a slight modification, this PDF of measurands is used as a Bayesian prior PDF of the measurand for each individual. The modification consists of eliminating from the prior the individual for whom the posterior PDF is being calculated, avoiding double counting of any individual. The PDF used in the Bayesian inference is this “everybody else” prior, as contrasted with the “everybody” PDF. Creating an “everybody else” prior for each individual, and using his or her measurement results as the likelihood, we produce a nonnegative Bayesian posterior PDF of the measurand for each individual. The mean of these posterior distributions equals the mean of the measurements, indicating that, for the population as a whole, we inject no bias. The method creates a nonnegative distribution for each individual that provides a defensible probabilistic statement about the value of his or her measurand given his or her measurement result, its uncertainty, and the measurements and uncertainties of everybody else in the group. We apply the process to real bioassay data sets of baseline measurements of 90Sr, 137Cs, and 239+240Pu. The method generates measurands with small positive mean values for negative measurement results, and generates measurands that are smaller than the largest positive values in the measurement data. The method follows Bohr’s Correspondence Principle. Limitations of the method are presented, as well as plans for future research and software development.

TPM-A.1 Determination of MDA Levels for Radiation Surveys of Potentially Activated PCB Capacitors

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The Argonne Intense Pulsed Neutron Source (IPNS) accelerator facility was permanently shutdown in December 2007. Among the first items that required disposal were its PCB oil-filled capacitors, since they were classified as hazardous waste. Several hundred large electrical capacitors had been located within radiological areas outside the accelerator shield, where they were exposed to neutron radiation. Gamma spectroscopy of two bulk capacitors from each radiological area established the maximum potential activation levels. Process knowledge had identified those capacitors as having been exposed to the highest potential prompt radiation fluence. As part of the approved disposal plan, a series of quality control radiation survey measurements was done for each shipping basket of capacitors. In order to determine the MDA and MDC of accelerator activation products that could be present, measurements were made on the exterior of a loaded reference shipping basket, with and without sealed radioactive sources embedded in the center of the basket. MCNP monte carlo modeling established that a radioactive source positioned in the center of the loaded capacitor basket was the worst case geometry. The source measurements allowed a direct determination of the MDA and MDC values for the dominant activation isotopes. This talk will describe these measurements and calculations of the MDA and MDC levels for the radioisotopes of interest.

TPM-A.2 Calculating Field Measurement Method Uncertainty

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Field measurements always involve uncertainty which must be considered when field data are used to support environmental decisions. Every measured and reported result should be accompanied by an estimate of total measurement uncertainty. Guidance on calculating measurement uncertainty for field measurements is provided in the Multi-Agency Radiation Survey and Assessment of Materials and Equipment (MARSAME), but the guidance does not include information on determining or estimating contributions to total uncertainty. This presentation discusses both Type A and Type B analysis of uncertainty and demonstrates how to use this information to calculate measurement uncertainty associated with field measurements.

TPM-A.3 Communicating Radiation Risks with Instruments and Dosimeters

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Many books, articles, proceedings, and policy and guidance statements have been published on radiation risk communication since 9/11. This material is worth reading and considering in advance of talking with groups or individuals. However, we have found that demonstrations with instruments and dosimeters can be especially effective in capturing audience attention for discussing radiation risks. Ray Johnson's copious presentations for our Society have made clear that members of audiences can have a variety of communication styles. Members of our Society have seen Ray use instrument detection of natural radioactivity from his extensive collection of consumer items, to demonstrate to various audiences the radiation exposures from natural products. We have found that relating GM count rates from natural products to dosimeter readings over time can be an effective entrée to risk and protection discussions for varied audiences. This is true either for sizable audiences, or for impromptu demonstrations to one or a few individuals. These demonstrations relate audible GM count rates of thousands per minute from natural products, and the corresponding exposure rates that would be indicated in gamma fields, to readings over given times on personal dosimeters that would begin to approach certain levels of risk. Demonstrations particularly grab attention and incite interest when a hand is fearlessly used to stop (mostly beta) radiation from entering the GM tube; with practice, they can be presented within a minute or two. This then opens the listener's (or audience's) interest in discussing radiation risk avoidance, and how easy it is to tell, with proper instruments and dosimeters, how much dose and risk would be received in a given time.

TPM-A.4 The Application of Super Heated Drop (Bubble) Detectors for the Characterization of Nano-Second-Pulsed Neutron Fields

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The calculation of worker dose from neutron radiation below 20 MeV is usually accomplished by obtaining measured light output from Thermoluminescent Dosimeters (TLDs) and then applying a neutron energy-dependent dose conversion factor. Determining occupational dose at any accelerator facility can be challenging because the neutron energy spectrum at the worker location may be unknown and can be difficult to obtain especially when an extremely short pulse width precludes the use of standard electronic instrumentation. Discussed, is a methodology using Super Heated Drop (Bubble) Detectors to characterize a unique type of neutron radiation environment (dose less than 100 mrem, a single pulse and a pulse width of only a few nanoseconds). This methodology yields a low resolution energy spectrum and fluence values. This information can then be used to calculate an average fluence-to-dose conversion factor for the measured location and help determine a light output to dose conversion factor for the workers TLD.

TPM-A.5 Use of Portal Monitors for Evaluation of Internal Contamination after a Radiological Dispersal Device

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Following a radiation emergency, evacuated, sheltered, or other identified members of the public may require monitoring for external and potentially internal contamination. Affected individuals would need to be prioritized for further analysis and treatment if indicated. Expeditious screening and prioritization of individuals presents many challenges especially when a large population is affected. Current laboratory capacity for indirect bioassay is limited. In this study, the use of portal monitors as rapid screening tools for internal contamination was evaluated. The Thermo Scientific TPM-903B Portal Monitor was modeled in Monte Carlo N-Particles Transport Code Version 5 (MCNP). This computational model was validated against the portal monitor's response to a series of measurements made with four point sources in a polymethyl methacrylate (PMMA) slab box. Using the validated MCNP5 model and models of the MIRD male and female anthropomorphic phantoms, the response of the portal monitor was simulated for the inhalation and ingestion of various radionuclides that may be used in an RDD. Six representative phantoms were considered: Reference Male, Reference Female, Adipose Male, Adipose Female, Post-Menopausal Adipose Female, and 10-Year-Old Child. The biokinetics via Dose and Risk Calculation Software (DCAL) was implemented using both the inhalation and ingestion pathways to determine the radionuclide concentrations in the organs of the body which were then used to determine the count rate of the portal monitor as a function of time. Dose coefficients were employed to determine the count rate of the detector associated with specific dose limits. These count rates were then compiled into procedure sheets that can be used by first responders

TPM-A.6 Assessing Internal Contamination Levels for Fission Product Inhalation using a Portal Monitor

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In the event of a nuclear power plant accident, fission products could be released into the atmosphere potentially affecting the health of local citizens. In order to triage the possibly large number of people impacted, a detection device is needed that can acquire data quickly and that is sensitive to internal contamination. The portal monitor TPM-903B was investigated for use in the event of a fission product release. A list of fission products released from a Pressurized Water Reactor (PWR) was generated and separated into two groups: gamma- and beta-emitting fission products and strictly beta-emitting fission products. Group one fission products—the gamma- and beta-emitting fission products—were used in the previously validated Monte Carlo N-Particle Transport Code (MCNP) model of the portal monitor. Two MIRD anthropomorphic phantom types were implemented in the MCNP model—the Adipose Male and Child phantoms. The Dose and Risk Calculation software (DCAL) provided inhalation biokinetic data that were applied to the output of the MCNP modeling to determine the radionuclide concentrations in each organ as a function of time. For each phantom type, these data were used to determine the total body counts associated with each individual gamma-emitting fission product. Corresponding adult and child dose coefficients were implemented to determine the total body counts per 250mSv. A weighted sum of all of the isotopes involved was performed. The ratio of dose associated with gamma-emitting fission products to the total of all fission products was determined based on corresponding dose coefficients and relative abundance. This ratio was used to project the total body counts corresponding to 250mSv for the entire fission product release inhalation—including all types of radiation. The developed procedure sheets will be used by first response personnel in the event of a fission product release.

TPM-A.7 A Comparison of Shielding Components and Practices in Interventional Cardiology

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A variety of radiation shielding devices are available for use in interventional cardiology. To determine the effectiveness of these devices and associated best practices, x-ray scatter transmission through (or around) these devices was measured. Scatter transmission measurements were made at locations corresponding to physician position for right femoral artery and right jugular vein access points. Measurements were made at elevations ranging from 50 cm to 175 cm from the floor. The shielding components that were evaluated include a ceiling mounted leaded-glass shield, table side lead apron, disposable radiation absorbing pads, and a custom table-mounted shielding device. Because it is difficult to maintain shield position during a procedure, measurements were made using both optimal and suboptimal device positions. In addition, a cost-benefit analysis was performed to identify which devices provided optimum protection. The results provide a quantitative and qualitative comparison of the effectiveness of the shielding devices and lead to best practice recommendations to reduce occupational dose in the interventional cardiology lab.

TPM-A.8 Evaluation of NCRP 147 CT Shielding DLP Method

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OSL dosimeters were used to measure CT scatter radiation dose to assess the effectiveness of the NCRP 147 DLP method for predicating shielding. The dosimeters were placed at a variety of locations around two GE CT units located in outpatient settings. Data was collected to provide information on DLP as well as other imaging parameters. The two sites averaged 8 CT studies per day. The workload distribution between the two was slightly different. Wide variances were seen in the DLP data. The difference between the average DLP and the suggested vales in NCRP 147 was notable. The expected doses for all location were calculated using the NCRP DLP method. Two different approaches were used. The first the calculations was based on the DLP values given in NCRP 147. The second method was based on the actual DLP for each site. The results in both cases overestimated the actual dose.

TPM-A.9 Application of Instruments in Medical Treatment Facilities

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Radioactive material is used in many medical treatment facilities throughout the nation. The radioactive material, whether it is sealed or unsealed, must be accounted for from entry into the facility until it is used in a patient, decays or it is shipped for disposal. The talk will focus on the use of calibrated instruments to perform the functions required to meet regulatory requirements and thus the safety of the radiation workers, non-radiation workers and general public. Calibrated instruments are used to show compliance with end of day activities, weekly contamination surveys, spill response and patient release from the medical treatment facility. Instruments are also used to ensure that radioactive material does not leave the medical treatment facility through everyday trash and biohazard waste. Machine produced radiation, such as x-ray machines, also need to be within certain performance tolerances to be used for patient care and calibrated instruments are used to show compliance with regulatory requirements.

TPM-A.10 Study of TENORM in Samples Gathered from BP Oil Spillage from the Coasts of Mississippi and Louisiana

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The British Petroleum oil spill of MC252, an oil and gas prospect in US exclusive economic zone, in the Gulf of Mexico, about 41 miles off the coast of Louisiana, has been assessed as the largest accidental marine oil spill in the history of the petroleum industry. Alcorn Campus is located within 150 miles of the Gulf of Mexico on the bank of the Mississippi River. During the end of July a member of the faculty and a student visited the coast and prepared oil samples from the coastal area in Grand Isle in LA, and Gulf Port, Biloxi and Pascagoula in MS. These samples are under study for existence and the amount of NORM and other radionuclides. The detection and the measurement of radionuclides as an undergraduate research is underway during the Fall semester of 2010 under the supervision of health physics faculty members. The results of this study will be presented and the students involved will be acknowledged.

TPM-A.11 The research on low altitude measurement technique for nuclear terrorism emergency: a case study on the detonation of RDD

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CANCELLED

After the 9.11 terrorist attacks, prevention, detection, and response of the nuclear terrorism have been an important issue for many countries. The terrorists may detonate the radiological dispersal devices (RDD) to create mass panic and widespread economic disruption. The technique of obtaining the radiation information in the emergency situation after the detonation of RDD has captured lots of researcher¡¯s attention. One low altitude measuring techn[illegible] presented in this paper. The measuring sys[illegible]tes the unmanned aerial vehicle (U[illegible]na detector which is made of a lar[illegible]d Cesium Iodide (CsI (Tl)) scintill[illegible].The monitoring theory is based [illegible]ments at measuring girds which are [illegible]ore the measurements. During the measure[illegible] vehicle carries the detectors to the monitoring are[illegible]nd stops above grids to acquire the radiation information including dose rate, activity and position. Two operational parameters have to be obtained before measurements, one is the width of the measuring grid and the other is the height of the detector above the ground. According to the mathematic model the parameters are obtained by solving the integral function which is derived from the calculation of fluence rate of detector. Two kinds of reconstructing algorithms are made to map the radiation distribution on the ground. One algorithm is based on the response factors of detector to the source; the other is based on the deconvolution theorem. The simulation is made to testify the performance of the reconstructing algorithms. The simulation shows that the reconstructing algorithms based on deconvolution theorem performs well when the detecting height is lower 20m, that the reconstructing algorithm based on response factor of detector to the source performs relative badly when the detecting height is more than 3 meters.

TPM-B.1 Verification of a Conservative TLD Neutron Correction Factor at the WIPP

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The determination of neutron doses with a single TLD element accurately is very difficult. Potential neutron energies range over nine decades and TLD response is inverse to the dose over these energies. A correction factor is usually applied to the neutron sensitive TLD element response and this is based on the assumed neutron energy distribution. In facilities with consistent neutron energy distributions, the neutron correction factor can be based on calibrations which are consistent with the determined spectra in each facility. At WIPP however, wastes originate from several different sites which can generate neutron spectra from multiple sources and with unpredictable moderation. While use of a very conservative neutron factor is easily defendable, it would result in an increase by a factor of eight (8) in neutron dose determinations. The potential solution is based on the Hankins nine inch/three inch (9"/3") sphere response ratio. This is commonly used with dose rate instruments to determine the appropriate calibration source for the neutron correction factor determination. As WIPP faces potentially different neutron spectra with each shipment, a more integrated approach is necessary. This presentation describes the use of 9"/3" spheres with internal TLDs to provide an average ratio over the course of a monitoring period. In addition to describing the system, an evaluation of the calibration results and impacts of different dose rate and energy environments will be presented.

TPM-B.2 Determination of a Site-Specific Spectrum Correction Factor in the Vicinity of the Holtec MPC During Drying in the Keuwanee Nuclear Power Station

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Since thermoluminescence dosimeters (TLDs) have an energy dependent response that differs from the the dose-equivalent response, a correction factor can be applied to correct for differences in the workplace and calibration spectra. Such measurements were recently performed at the Kewaunee Nuclear Power Station during two spent fuel canister loadings and emplacements. The neutron dose equivalent rates and spectra in the vicinity of the casks during the welding and drying process were measured. A Bonner sphere spectrometer (BSS) and a tissue-equivalent proportional counter (REM-500) were used to determine the neutron dose equivalent rates at several locations around the canister. The neutron dose equivalent rates were also measured with bubble dosimeters, Electronic Personnel Dosimeters and TLD personnel dosimeters. The BSS and TEPC measurements were compared with the dosimeter readings to create workplace spectral adjustment factors.

TPM-B.3 Effects of Different Moderators on the Neutron Spectra, Fluence and Dose Rates from Californium Source

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The neutron spectra, neutron fluence rates and neutron ambient equivalent dose rates have been measured from a californium-252 neutron source with the following moderators: 5 cm-, 10 cm-, and 15 cm-thick heavy water (D2O) moderator and 2 cm-, 5 cm-, 10 cm-, and 15 cm-thick polyethylene moderator. The neutron spectra are measured from thermal to fast neutron energies. The neutron ambient equivalent dose and dose rates are determined by using the measured spectral distribution and applying the corresponding fluence-to-dose conversion factors from ICRP-74. Changes to the neutron spectral distribution from the different moderators, as well as, to the fluence rates and ambient equivalent dose rates are discussed. (LLNL-ABS-453211)

TPM-B.4 US Army Radiation Standards Laboratory

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The US Army Radiation Standards Laboratory (RSL) is located at Redstone Arsenal within Huntsville, AL and is collocated with the Army Dosimetry Center. It is part of the US Army Primary Standards Laboratory which provides the highest level of radiation metrology and calibration services. As such, the RSL maintains direct traceability to the national, international, and intrinsic standards and calibrates the instruments used to calibrate Army Radiation Detection Indication and Computation (RADIAC) instrumentation and photonic devices all over the world. Our direct traceability is then transferred to the rest of the Army via our calibrating the standards used by the secondary laboratories. The Secondary Laboratories use their standards to both perform calibrations of equipment and weapons systems as well as calibrating the standards used as tertiary laboratories. The tertiary laboratories calibrate equipment and weapons systems. This briefing will discuss the composition of the Radiation Standards Laboratory, the accreditation of the Laboratory, and some of the equipment supported by this Laboratory. As I discuss these items, I will give you a brief tour of our laboratory.

TPM-B.5 Construction and Maintenance of Reference Radiological Calibration Fields of Kaeri

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The Korea Atomic Energy Research Institute (KAERI) is the biggest radiological calibration laboratory accredited by the Korea Laboratory Accreditation Scheme (KOLAS) and has served to calibrate most of radiation measuring devices and performed the standard irradiation service according to the very similar way to the ANSI N13.11 for performance testing of the personal dosimetry systems used in Korea. He has 13 kinds of official categories of calibration service accredited by KOLAS, which are classified 3 parts of radiation, radioactivity and neutron in the files of ionizing radiation, but provides several testing fields and modes to measure the response of several types of radiation detectors. The reference calibration fields for photon consists of 16 kinds of x-ray and 7 Cs-137 sources with radioactivity and other useful gamma sources are Am-241, Co-60 and Ra-226. Only two Sr-90/Y-90 sources with activity are available for routine calibration and test because of low activity of Tl-204 and Pm-147 due to its decay at present. Four kinds of neutron fields, a Cf-252, a heavy water moderated Cf-252, an AmBe and two thermal neutron fields produced by using a graphite piles and 8 AmBe sources, are used for calibration. The calibration of contamination monitors and the test of air monitors are carried out by using several kinds of large area radioactivity sources as well as semi-point disk sources. Inter-comparison measurement studies with foreign laboratories of the Pacific Northwest National Laboratory and the Japan Atomic Energy Agency have been done irregularly to check the qualities of some reference calibration fields such as x-ray beam codes, neutron fields or large area activity sources since 1994. Especially in case of neutron fields the KAERI produced the scattered neutron fields more than 10 kinds using radioactive sources and accelerators, and recently is preparing others using a Deuterium-Tritium (DT) neutron generator. These neutron fields are similar to the simulated workplace neutron fields according to ISO-12789.

TPM-B.6 Production of Fast Neutron Calibration Fields using a Proton Accelerator of Kirams

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As one of the methods to construct a simulated neutron calibration field (SNCF), it is recommended to use an accelerator by ISO-12789, the Korea Atomic Energy Research Institute (KAERI) has used the proton accelerator MC50 of the Korea Institute of Radiological and Medical Science (KIRAMS) to produce the fast neutron calibration fields. The MC50 is being operated below 45 MeV for radioisotope production and neutron research as well as the irradiation service of proton beams. In order to characterize the fast neutron calibration fields, they were simulated by using the MCNPX code and measured by using the KAERI Bonner Sphere System in the neutron therapy room of the KIRAMS when two kinds of proton energies of 35 and 45 MeV bombarded to additional targets of a 15 mm thick Be and 6.1 mm Cu targets, which is externally attached to an end side of a beam tube made of aluminum without any modification of a beam guide of the MC50, and a cylindrical PE moderator enclosing the beam tube including a target assembly. Additionally a heavy water moderator sphere with a diameter of 32 cm with a 0.5 mm thick Cd cover was placed between the end of the beam tube and the reference position of 90 cm from the target. Spectral average energies of fast neutron fields ranged from 5.6 to 17 MeV and the ambient dose equivalent rates of those were recorded from 1.25 to 17.6 mSv/h per nA. These fast neutron fields are one of the SNCFs of the KAERI and can be also used as one of the calibration fields for a type test of neutron measuring devices.

TPM-B.7 Development of Automatic Clearance Measurement System Using Shape Measurement and Monte Carlo Calculation

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The most important issue in the inspection of clearance level is how to ensure the reliance on recycled metal from nuclear facilities. It is a serious concern that the circulation of recycled metal might be hindered if post-clearance-level-inspection metals were to develop a poor reputation. In order to remove such an anxiety and ensure reliance on recycled metal after inspection, the possibility of detecting hot spots of contamination exceeding the surface contamination level must be eliminated completely. To solve the above issue, a practical clearance level inspection system has been developed using laser shape measurement and Monte Carlo calculations in order to quantify very low activity in metal wastes. This system consists of four laser scanners, eight large plastic scintillation detectors at the upper and lower sides of the radiation measurement area surrounded by a 5-cm-thick lead shield. By using the digital configuration data of waste, two Monte Carlo simulations are carried out to calculate the calibration factor and the correction factor for background reduction due to the self-shielding effect of the waste during the radiation measurement. The accuracies of the calibration and background correction were experimentally evaluated using mock-metal waste of various types of shapes, numbers and sizes with radioactive sources of Co-60 and Cs-137. The detection limit of this system, conservativeness and uncertainties of activity estimation were also evaluated with considering practical measurements.

TPM-C.1 Monte Carlo Simulation of Entrance to Exit Air Kerma Ratio in Interventional Radiology

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Purpose: The entrance air kerma in Interventional Radiology (IR) provides an estimate of the patient skin dose, whereas the exit air kerma is related to the amount of quantum mottle in the result image. In this study, we investigated the entrance to exit air kerma ratio in IR as a function of x-ray beam quality and patient size. Method: We quantified the pattern of energy deposition in water cylinders with diameters ranging from 17 cm to 30 cm. MCNP5/MCNPX 2.6.0 was used to simulated IR examinations with the water cylinder surface fixed at a distance of 55 cm from the x-ray source. Four x-ray spectra ranging from 60 to 120 kV were obtained using XCOMP3R software package. Air kerma data were obtained at the phantom entrance and exits, both of which included scattered radiation. We computed the entrance to exit air kerma ratio (R) that provides an estimate of the relative entrance skin dose when the radiation at the image receptor (i.e., quantum mottle) is kept constant. Results: For a 17 cm diameter water cylinder, increasing the x-ray tube voltage from 60 to 120 kV reduced the value of R from 54 to 21. For a 30 cm diameter water cylinder, increasing the x-ray tube voltage from 60 to 120 kV reduced the value of R from 970 to 170. At a fixed x-ray tube voltage of 80 kV, the x-ray tube voltage most frequently encountered in IR, the value of R was 33 for a 17 cm diameter water cylinder and increased to 390 for a 30 cm diameter water cylinder. Conclusion: Increasing the patient diameter from 17 to 30 cm increases the entrance to exit air kerma ratio by about an order of magnitude. Increasing the x-ray tube voltage from 60 to 120 kV reduces the entrance to exit air kerma ratio by factors of between three and six.

TPM-C.2 Radiation Dose Measurement - Analysis for a 320 Slice CT Scanner

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The new 320 slice CT sanners which irradiate a 160 mm length in a single rotation present challenges for the measurement and analysis of radiation dose. Longer CT phantoms are required. CT ionization chambers and OSL dosimeters were utilized to perform the measurements. There are various scan modes: volume scans, helical scan and view modes and thin slice helical modes. There are four selectable x-ray tube potentials: 80, 100, 120 and 135 kVp. There are also three selectable FoV's and filters. The various selectable scan options complicate the radiation dose measurement procedures. The large amount of data collected was reduced to a few simple equations that could be used estimate the radiation dose for any of the clinical procedures. Limitation to CT radiation dose assessment is also reviewed.

TPM-C.3 Determination of Air Crew Exposure in Domestic Flights of Aseman Airline in Iran. On Board Measurements and Calculations with CARI 6 Code

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The radiation dose at high altitudes is due to different types of particles, mainly photons, electrons, positrons and neutrons, with a wide energy range. In this experiment the neutron and non-neutron component of cosmic radiation dose were measured in several round trip flights using a photon detector and a neutron detector. The results were then compared with the dose estimated using CARI-6 code. The non-neutron dose of 2.20microSv was measured in the longest airplane travel (Asaluye-Rasht) when we add to it the neutron dose rate of 0.99microSv; we get a total cosmic dose rate of 3.19microSv (at the flight level of about 34000 ft) which is in close agreement with the calculated value of CARI code (3.2 microSv). The results of the measurements in other flight routes were also in close agreement with the dose calculations using CARI 6 code. Finally a number of flight personnels were equipped with TLD cards for evaluating the gamma dose and polycarbonate dosimeters for assessing the neutron dose during a period of 11 months. The measured value of the average annual dose received by the crews was 1.51 mSv/y, 0.61 mSv/y for neutron component and 0.56 mSv/y for photon component. The annual dose received by most of the crews exceed 1msv.

TPM-C.4 Lead-210 and Polonium-210 in Iron and Steel Industries

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Iron and steel industry was ranked as the largest industrial source of toxic environmental contamination in the USA. About 2-4 tones of various solid wastes (slag, sludge, dusts and scales) are generated per ton of steel produced. These wastes contain a notable concentration of heavy elements and radionuclides that could be a source of environmental contamination and occupational exposure. Composite samples of different iron and steel industry's wastes were collected from four iron and steel factories. Activity concentrations, in Bq/kg, of Pb-210 and Po-210 were measured using gamma-ray spectrometry based on HPGe detector and alpha particle spectrometry based on PIPS detector after radiochemical preparation. Activity concentrations of Pb-210 and Po-210 were in the range of < DL–4238 and 1-5656 Bq/kg, respectively. Occupational dose due to dusts inhalation was calculated. According to the assumed scenario, the occupational exposure is much lower than the reference dose limit. The environmental impact due to wastes storage and/or use should be considered generally and case by case.

TPM-C.5 Making the Most of Uncertain Low-Level Measurements

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Many measurement techniques in occupational and environmental monitoring produce results that have very substantial uncertainties. In many cases, a blank or background value must be subtracted from a gross result to produce a net result, a process that can produce non-physical, negative net results. When uncensored measurement results on a population are accompanied by well-characterized uncertainties, a novel method developed at PNNL separates the variance of the observations into two components, one arising from uncertainty and the other arising from population variability. Assuming the uncertain measurements are independent and unbiased, the method forces the arithmetic mean of the measurands (the "true values") to equal that of the measurements. Further assuming that the measurands are lognormally distributed, the geometric mean and geometric standard deviation are calculated for a distribution of possibly true results. Assuming that the Bayesian prior probability density functions (PDF) of each individual measurement is a lognormal distribution comprised of every other measurand, posterior PDFs for each measurand are computed using Bayes's theorem. The method corrects physically meaningless negative measurement results into PDFs of small positive values, but does little to alter large positive measurement results. Some surprising implications are illustrated by application to bioassay measurements of Sr-90, Cs-137, and Pu-239.

TPM-C.6 On the Detection Efficiency of the RaDeCC System for Ra-224 and Ra-223 Measurements

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The RaDeCC system is designed to simultaneously measure short lived Ra-224 (3.66 d) and Ra-223 (11.4 d) by delayed coincidence counting the alpha particles emitted from the Ra decay daughters (Rn and Po isotopes) in a Lucas cell. Because it is easy to operate and does not need a radioactive tracer for the detection efficiency, the RaDeCC system has been widely used for the measurement of the short lived radium isotopes in various geological and oceanographic investigations. However, it was found that the detection efficiency for Ra-223 is unstable and has a much larger uncertainty than that for Ra-224. A theory based on the backscattering of the alpha decay of the calibration radioisotope was proposed but never demonstrated experimentally. In order to test this theory and find the cause for the large uncertainty in the detection efficiency for Ra-223, we prepared four different calibration sources, each containing a radioisotope (Th-232 or Ac-227) absorbed on acrylic fibers or electrodeposited on stainless-steel planchets. The radioactivity of all the calibration sources was standardized by gamma-spectroscopy and gas proportional counting. The detection efficiency of the RaDeCC system for Ra-224 and Ra-223 was measured with the calibrations sources. The experimental results are discussed. The principal of testing the backscattering effect on the calibration efficiency with the sources on different substrates are also discussed in detail.

WAM-A.1 Radon Rejection in Next Generation Contamination Monitor

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The challenges associated with the detection and determination of radon by a body monitor is dependent upon several variables. Due to statistical limitations, no monitor will result in a 100% reduction of false alarms. However, the capability of the body monitor to dramatically shorten the amount of time in accurately making the determination, is a huge time-saving benefit to all concerned. In addition to the enhanced radon rejection algorithms, the effectiveness of a body contamination monitor is also dependent upon several factors: Alpha and beta detection efficiency, body placement and attenuation, and body proximity to detector face. All of these are key factors in making a determination if radon is present. The monitoring and determination of the presence of Radon is still dependent upon a strong Health Physics program which does not allow for the release of radioactive material to the environment. The next generation body monitor is the next step in preventing the release of radioactive material and minimizes the delays from exiting the RCA due to Radon. The value of accurate Rn rejection to any contamination monitoring program can only be seen as significant, and therefore one of the most important features of the next generation body monitor.

WAM-A.2 TRU Measurement and Screening Assay of Air Filters with Radon Progeny Interference

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When carrying out alpha spectrometry on air filters, the resultant spectral shapes are highly dependent on a number of parameters. These include relative abundance of radon and thoron progeny, total mass loading, activity distribution in the sample, dust loading and to some extent, the type of filter being used. As the overwhelming abundance of transuranic (TRU) activity in the nuclear complex is composed of Pu and Am, the vast majority of TRU alpha activity is bounded by an upper 5.6 MeV alpha decay energy with the rest of the TRU alpha emissions being below this limit (only some isotopes of Cm and Cf do not fall in this range). As the lowest energy alpha given off by radon progeny is 6 MeV, discrimination of TRU and NORM is dependent on resolving how much of the 6 MeV peak from radon and thoron is contributing to the potential TRU peaks which typically occur between 5 and 5.6 MeV. Solutions brought forth by WIPP are presented for these measurements which were designed to be both operationally friendly and quantitatively useful.

WAM-A.3 Current State of the Art in Measuring Radon

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According to the US EPA, radon is the leading cause of lung cancer among non-smokers and the second leading cause overall. Data from several epidemiological studies show definitive evidence of the association between indoor radon exposure and lung cancer. The discovery of elevated radon concentrations throughout most of the U.S. has become of great concern due to the health risk and therefore it is of great importance to investigate the distribution of indoor radon throughout the country. In the past 30 years, emphasis has been on measuring radon rather than radon progeny concentrations because of the simplicity, convenience and cost effectiveness of radon measuring instruments and methods. Using an equilibrium ratio between radon and radon progeny of 0.4-0.5, radon concentration measurements can be converted to working level and to exposure in working level month. The US EPA demonstrated that more than 90% of the short-term measurements are in good agreement with the long-term measurements. In recent years, over 1 million short-term measurements for radon were made annually using grab sampling, integrating and continuous radon devices. As a result of these short-term measurements more than 800,000 residences with elevated radon levels were identified and were mitigated successfully. This paper will emphasize the development and use of different instruments and methods their sensitivities practicality and cost effectiveness for making short-term measurements of environmental radon. More than 99% of indoor measurements involve radon only. Radon progeny used mostly in research and diagnostics will not be discussed.

WAM-A.4 Correction to Counting Statistics for Measurements of Radon in Air Using Continuous Monitors and Alpha-Track Devices

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"Counting statistics" underestimates the true uncertainty of the count for many methods of measuring radon due to the inclusion of correlated counts from more than one radionuclide. To correct for this effect, a factor J has been defined, which is the ratio of the theoretical true variance of the count to the theoretical value of the variance that is estimated by counting statistics; i.e. the mean of the count. The approach for determining J is to calculate the probabilities of all possible outcomes of a single radon atom, assuming that all other radon atoms behave in the same manner. The theoretical values of the mean and variance of the count are calculated as a function of the counting time and the counting efficiencies of the detected particles. Then, J is the ratio of the theoretical variance to the theoretical mean. The determination of J for three measurement methods, where the radon is collected and later analyzed, has been discussed previously. The current work considers methods where counts, or tracks, are recorded during the measurement period; i.e., continuous radon monitors and alpha-track devices. Here it is assumed that radon is continuously replenished from its parent radium-226 during the measurement period. Thus, the approach is to consider all possible outcomes of a single radium-226 atom instead of a radon atom. Then the value of J is calculated in a manner similar to that used for a grab measurement. For a scintillation cell of 5 cm diameter and 10 cm height used for a grab sample of radon, J = 1 for small values of counting time but increases to a maximum of 1.98 for a counting time of about nine hours and then decreases for longer counting times. When the same cell is used for a continuous measurement, the value of J increases with measurement duration to 2.19 and remains at that value for practical measurement periods greater than about 75 hours.

WAM-A.5 Radon Reference Chambers in the U.S. and Radon Measurement Performance Testing

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For calibration, performance testing and other quality assurance purposes, devices for measuring radon-222 in air are placed in a reference chamber where the radon concentration, temperature and humidity are well known and controlled. In the 1980's, several such chambers existed in U.S. federal facilities. Since then, there has been a shift from the federal to the private sector as radon testing companies and manufacturers of devices came into existence. Now several private companies have radon chambers of various sizes and designs depending on their intended uses. Only two radon reference chambers remain in the federal government, however, and only the facility of the Environmental Protection Agency (EPA) in Las Vegas is available to private companies for intercomparisons. This facility provides annual single-blind intercomparison tests for private companies that are certified by a national radon proficiency program to conduct performance tests of radon measurement companies certified by that program. Two such facilities have been certified by the National Environmental Health Association's National Radon Proficiency Program (NRPP); at Bowser-Morner, Inc. in Dayton, Ohio and at Radon Measurements Lab in Colorado Springs, Colorado. Both facilities use scintillation cells to intercompare with the EPA facility, and they maintain an overall agreement within ±10% of the EPA's reference values. The scintillation cells further serve as transfer standards for maintaining the calibration of their radon reference chambers. These two facilities have conducted over 1800 performance tests of various devices used by radon measurement companies certified by the NRPP. Summary statistics, including passing rates for each type of measurement device and for several ranges of radon concentration, are presented for over 400 such tests conducted by each of these two facilities.

WAM-A.6 Development and Intercomparison of Radon-In-Water Standards

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The standardization of instruments used for radon-in-water measurements typically involves handling and disposal of Ra-226 in solution. To avoid contact with the Class A carcinogen, radium-free solutions were prepared and tested for use as radon-in-water standards. Filters containing known amounts of Ra-226 were sealed in polyethylene and placed in vials filled with distilled water for over 30 days to allow the decay products to establish secular equilibrium. Over a 3-year period, voluntary intercomparisons of the radon-in-water standards were conducted to investigate the accuracy of analyses by commercial, government, and private companies. Several analytical methods, including liquid scintillation, alpha scintillation, continuous radon monitors, gamma spectroscopy, and electrets, were utilized by the participants. Results show that, at radon concentrations from 16-693 Bq/L (437-18,700 pCi/L), most participants reported concentrations within 25% of the known amounts. Outliers typically under-reported radon levels, likely due to loss of the gas during sample transfer.

WAM-B.1 Some Bioassay Methods for High-risk Radionuclides

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Following a radiological or nuclear emergency, the affected public and first responders may need to be quickly assessed for internal contamination by the radionuclide(s) involved. Urine bioassay is one of the most commonly used methods for assessing radionuclide intake and radiation dose. Rapid and field deployable bioassay methods that deliver quick assessment results are very welcome. At Health Canada, we have developed some emergency bioassay methods that are rapid and robust. Some of them are field deployable. This paper presents the methods we developed for most of high-risk radionuclides identified by IAEA TECDOC-1344, including atom counting methods for Am-241, Pu-239, Pu-240, and U-235, liquid scintillation based methods for Sr-90, Po-210, Am-241, Ra-226, Pu-238, and Pu-239, as well as field deployable methods for Sr-90, Cs-137, Co-60, Ir-192, Se-75, and Yb-160. The knowledge gaps and operational challenges for some specific radionuclides are discussed. Future tasks for method development and potential collaborations are suggested.

WAM-B.2 Alpha Spectrometry of Thick Samples for Environmental Monitoring

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Laboratory of Inorganic and Nuclear Chemistry at Wadsworth Center, New York State Department of Health performs radiological surveillance for New York State purposes. The surveillance involves measurements of environmental samples to satisfy National Drinking Water Regulations, monitoring of nuclear power reactors, waste repositories, hospitals, research institutions, etc., to protect population of New York State. An important part of this surveillance consists of precise determination of alpha-emitting radionuclides which is, however, laboratory intensive as well as manpower- and time-consuming. Therefore, there is a need for methods of fast assay of alpha emitters with application to nuclear emergencies. A new method for alpha spectroscopy of evaporated water residues was developed, consisting of evaporation of drinking water, flaming of the planchettes, and thick alpha-spectroscopic measurements using grid ionization chamber. The method can provide quantification of moderate levels of alpha radioactivity within a few hours. The detection of sub mBq/L activity concentrations is achievable with longer counting times. Detailed investigations of flaming of the planchettes, the humidity effect, and alpha spectroscopy of thick sources are described. A universal, three-dimensional calibration of the method was performed using standards containing U-238, Th-230, Pu-239, Am-241, and Cm-244 radionuclides. This calibration is valid for any sample which can be prepared as a uniform layer, such as the residues from surface water, acidic washing or leaching from materials, as well as urine. Detailed discussion is presented of novel algorithms for fitting of alpha-particle spectra for samples ranging from weightless to moderately thick, consisted of generalized-exponential and power-law functions. The developed method is appropriate as a fast identifying/screening technique for emergency response involving alpha radioactivity, and was successfully tested in the federal Empire09 Radiological Emergency Exercise.

WAM-B.3 Inductively Coupled Plasma Mass Spectrometry Measurement of Technetium-99 including Uncertainty and Detection Limit Determinations

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Inductively Coupled Plasma Mass Spectrometry (ICPMS) analysis for Tc-99 is an alternative to the conventional Liquid Scintillation Counting (LSC) method. The ICPMS technology can have several advantages over the LSC method depending on the separations chemistry performed. One advantage of the ICPMS method is that common beta emitter interferences seen in the LSC method, such as Th-234, are not a problem when ICPMS analysis is used since the 234 mass of Thorium cannot be confused with mass 99 for Technetium. Another advantage is labs which use a Tc-99M radiometric tracer are required to allow the tracer to decay before LSC counting. This can add as much as five days to the total analysis time. ICPMS does not require the Tc-99M to decay before analysis. Some laboratories have used a stable Rhenium tracer in lieu of Tc-99M. When analyzing pipe leachate samples, high tracer yields due to Rhenium in the pipe material have been noticed. The use of Tc-99M as a tracer eliminates this concern. When performing analytical measurements, a necessary component of any analytical result is the uncertainty of the measurement and the detection limit of the method. A discussion of the method used to determine measurement uncertainty and method detection limit will be discussed. Lastly, data will show the comparison between the LSC and ICPMS methods when both methods were used for analysis of samples provided by the Mixed Analyte Performance Evaluation Program (MAPEP) administered by the US Dept. of Energy, Radiological and Environmental Sciences Laboratory (RESL).

WAM-B.4 Deconvolution of Mixed Gamma Emitters Using Peak Parameters

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When evaluating samples containing mixtures of nuclides using gamma spectroscopy the situation sometimes arises where the nuclides present have photon emissions that cannot be resolved by the detector. An example of this is mixtures of Am-241 and plutonium that have L x-ray emissions with slightly different energies which cannot be resolved using a high-purity germanium detector. It is possible deconvolute the americium L x-rays from those plutonium based on the Am-241 59.54 keV photon. However, this requires accurate knowledge of the relative emission yields. Also, it often results high uncertainties in the plutonium activity estimate due to the americium yields being approximately an order of magnitude greater than those for plutonium. In this work an alternative method of determining the relative fraction of plutonium in mixtures of Am-241 and Pu-239 based on L x-ray peak location and shape parameters is investigated. The sensitivity and accuracy of the peak parameter method is compared to that for conventional peak decovolution. LA-UR-10-05626

WAM-B.5 Determination of Energy Spectra and Absorbed Dose Rate of a Ni-63 Based Low-Energy Beta Source

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Experimental evidence has shown that electrons and photons with energies less than approximately 50 keV are more effective in causing biological damage than their high energy counter parts. Since the applications of low-energy electron sources are widely found in industry, research, and medicine, there is a need to develop a method to quantitatively assess the “relative biological effectiveness (RBE)” of low-energy electrons. Ni-63 is a pure beta emitter with a half-life of 100 years. The maximum beta energy is 68 keV and the average energy is 17.5 keV. The combination of long half-life and low energy makes Ni-63 the ideal benchtop radiation source for in-vitro study the RBE of low energy electrons. We have developed an electroplating method that extracts high purity Ni-63 ions from a solution and deposit them onto the surface of a metal substrate (e.g. Cu). The design of our beta-particle irradiator is similar to that of a widely used benchtop alpha-particle irradiator in that a disk-shape source is placed immediately underneath a thin Mylar bottom where a monolayer of cells are grown and attached to. It is important to note that the beta energy spectrum and dose rate are extremely sensitive to the Mylar thickness and to the thickness of the air gap between the Mylar and the source. That is, a small increase of the Mylar thickness or the air gap thickness may drastically change the beta spectrum and its corresponding dose rate. We have used a gas-flow proportional counter (Ludlum 120)and obtained the beta spectra of the Ni-63 source we made and the results agree well with that obtained with the Monte Carlo code MCNP. We estimate that a 1 inch-dia disk source with 10-nm thick Ni-63 coated on the surface would deliver a dose rate of approximately 1 Gy/min to the monolayer of cells growing on Mylar placed immediately above the source. The total activity of such a source is merely 2.5 mCi, which is easily obtainable.

WAM-B.6 Intercomparison of Direct Radiobioassay and Radiochemical Analysis of Tissue Specimens from a Plutonium and Am-241 Contaminated Wound

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Several tissue specimens from a 1985 plutonium and Am-241 contaminated wound were analyzed by direct measurement using a thin NaI gamma scintillation detector on a field survey instrument, a planar Ge detector gamma spectrometry system, and then by destructive analysis with radiochemical separations for Pu and Am, followed by alpha spectrometry for Pu-239, Pu-238, and Am-241, and liquid scintillation determination of Pu-241. The intercomparison of these analytical methods showed that the use of a thin NaI detector coupled to a simple field survey instrument provided reasonable capability for go/no-go decisions on initial chelation therapy and for monitoring the progress of wound debridement or excision activities. The preliminary results from the Ge tissue measurements for Am-241 and Pu-239 activity showed reasonable agreement with the radiochemical analyses. The results obtained with the Ge system are considered adequate for making preliminary dose estimates. The destructive radiochemical analysis provided the definitive determinations of activity ratios for application to long-term wound monitoring of residual activity and material balance for the dose assessment process.

WAM-C.1 Uses of Field and Laboratory Measurements during a Radiological or Nuclear Incident

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There will be unprecedented demand for radioanalytical capabilities following a radiological or nuclear incident. Decisions regarding re-occupancy of places of work, schools, playgrounds, day care centers, hospitals, places of worship, etc., must be based on defensible data of demonstrated accuracy and quality. While field and laboratory radionuclide measurements will both play critical roles following a radiological or nuclear incident, there are inherent tradeoffs between laboratory and field measurements in terms of reliability, repeatability, and uncertainty and turnaround time, cost and throughput. Whether measurements are performed in the field or at the laboratory, the data generated needs to be technically defensible. Data should be obtained using rigorous and well-documented analytical protocols within the context of a robust and well-implemented quality system. Ultimately, it is the responsibility of Incident Commanders and their designees to ensure that all analytical data produced is of sufficient quality to support decisionmaking. By understanding the respective benefits and limitations of field and laboratory measurements, Incident Commanders, planners and decisionmakers will be in a position to decide under which circumstance one approach is favored over another, and how field and laboratory measurements may be used synergistically to increase the effectiveness of the response while ensuring the reliability and defensibility of measurements used for decisionmaking. This paper presents key topics discussed in "Uses of Field and Laboratory Measurements During a Radiological or Nuclear Incident" currently under preparation for ORIA NAREL, by EMS, Inc. Special attention is given to the respective strengths and limitations, and thus applicability to field and laboratory measurements, and their potential for complementary use dependent on the phase of the incident, respective action levels, DQOs and MQOs, and specific details about the type of measurement being conducted (i.e., the radionuclides present, matrix or surfaces, weathering, etc.)

WAM-C.2 Essential Metrology for Field and Laboratory Measurements during a Radiological or Nuclear Incident

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Performing any measurement, whether it is a radiological measurement performed in a field or laboratory setting, depends on the basic principles of metrology. Every measurement is a comparison to a standard. Traceability as a property of the result of a measurement whereby it can be related to stated reference standards, through an unbroken chain of comparisons all having stated uncertainties. The uncertainty indicates the degree of confidence that can be placed in a measurement. Measurement results must be reliable, and results from different organizations, possibly using different methods, be comparable. Only then can the data be objectively and confidently accepted by all those likely to use that data for decision-making. Some fundamental steps of metrology that should be followed in measurements of radiation or radioactivity are: Ensure that an overall Quality System is in place; Develop/define DQOs and MQOs; Choose or develop the appropriate measurement method to meet the MQOs; Know the method's strengths and limitations; Use suitable NIST-traceable certified reference materials; Validate that the method will meet the MQOs,demonstrate and confirm it; Identify the sources of uncertainty in the method; Derive a model or an equation for the measurement that allows the combined standard uncertainty can be determined; Evaluate the uncertainty of each measurement using a recognized approach; Clearly establish and document that results are traceable to a national standard; and Report the results and the associated combined standard uncertainty in the appropriate units and number of significant digits. These steps apply equally to measurements made in the field and to those performed in a laboratory. Many of these steps can be planned in advance, given the DQOs. By specifying the required method uncertainty (an MQO) at the analytical action level it is possible to ensure that decision errors will not exceed the levels deemed acceptable for responsible decision-making. An example template for preparing this information will be shown.

WAM-C.3 Emergency Response-Field vs. Lab Measurement

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Following a radiological or nuclear incident, assessment of the levels and extent of the contamination requires both timely and quality information on which to make the necessary decisions for appropriate recovery actions. Measurement protocols selected will depend on the radiation involved (gamma, beta, alpha), on the surfaces or matrices impacted, and on timing requirements for obtaining the required information. The latter requirement will depend on a description of the incident response which may be defined by three distinct phases: a) Immediate response phase – hours to days. Assess the extent and levels of contamination; identify impacted people, structures, and environmental media; establish controls; make preliminary isotopic identification; and assess recovery resource needs. b) Recovery/decontamination phase – days to months. Define DAG's and release limits, assess decontamination and remediation progress, establish DQO/MQO's for both the recovery phase and the release phase based on a graded approach, and acquire additional resources to meet the recovery/release objectives. c) Release phase – days to months. Release materials, equipment, structures and land areas based on agreed release limits, using protocols defined by the DQO/MQO process. Complete verification and validation of the data based on QA/QC requirements based on agreed upon measurement uncertainties. This presentation will compare the strengths and weaknesses of field direct measurements versus laboratory sample analysis that are based on the isotopic composition, the surface or media impacted, and the timing needs during the initial response and the following recovery phases. This comparison will stress the need to combine the two approaches based on the relative strengths and limitations of available technologies.

WAM-C.4 FRMAC Interactions During a Radiological or Nuclear Event

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During a radiological or nuclear event of national significance the Federal Radiological Monitoring and Assessment Center (FRMAC) assists federal, state, tribal, and local authorities by providing timely, high-quality predictions, measurements, analyses and assessments to promote efficient and effective emergency response for protection of the public and the environment from the consequences of such an event. Within the FRMAC the Assessment unit, in conjunction with the Monitoring and Sampling, and Laboratory Analyses units, develops a monitoring and sampling plan designed to meet these needs. This plan generally includes both field monitoring and sample collection for laboratory analyses. This presentation discusses the interactions between the FRMAC Assessment, Monitoring, and Laboratory Analyses units with emphasis on sampling and laboratory analyses. Laboratory Measurement Quality Objectives and how they affect sampling plans is discussed. Examples from recent exercises are included. If sufficient time is available, a short presentation on the capabilities of the new DOE Fly-away Laboratory is included.

WPM-A.1 Commissioning of the Fission Fragment Ion Source

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A Cf-252 fission source yields neutron-rich fission fragments for nuclear and astrophysics research. The CAlifornium Rare Ion Breeder Upgrade (CARIBU) project is an upgrade to the Argonne Tandem Linear Accelerator System (ATLAS) that selects radioactive fission fragment ions for acceleration. Fission fragments, from a 3% fission branch, stop in a gas catcher, are extracted into an electron cyclotron resonance (ECR) ion source to increase the charge state, and then accelerated in ATLAS. To gain experience with electrodeposited sources, tests were conducted with a 56 MBq (1.5 mCi) source. Following those tests, commissioning with a 2.7 GBq (73 mCi) source commences. Once commissioning is successfully completed, a 37 GBq (1 Ci) source will be introduced. The radiation fields produced by an unshielded 1 Ci Cf-252 source are 0.46 Sv/hr (46 rem/hr) neutron and 40 mSv/hr (4 rem/hr) gamma at 30 cm. A shielding system has been constructed that reduces the radiation fields for the 1 Ci source to less than 0.01 mSv/hr (less than 1 mrem/hr) at 30 cm from all accessible surfaces. This presentation provides the commissioning results with the 73 mCi source. *This work is supported by the U.S. Department of Energy, Office of Nuclear Physics, under Contract No. DE-AC02-06CH11357.

WPM-A.2 Quantification of Induced Radioactivity for a Compact 11 MeV Self-Shielded Cyclotron for Decommissioning Funding Purposes

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The amount of decommissioning funding required to offset future liabilities depends on the quantity of radioactivity authorized under a license. The activation products in a Siemens RDS-112 cyclotron located at the factory were determined by use of gamma spectroscopy on both the cyclotron structure and on various subassemblies and components as it was being dismantled. The activity was quantified through the application of the Canberra ISOCS software. This data was then analyzed to determine the total induced activity in the cyclotron. Furthermore the concept of "fixed" and "replaceable" components was developed to further delineate those elements which would be present throughout the operating lifetime of the cyclotron and would be the primary focus of the decommissioning effort, from the consumable elements that wear out and are periodically replaced. The fixed components include the cyclotron magnet, vacuum tank, and shielding. Replaceable components are items such as targets, ion source parts, and target windows. Determining the amount of decommissioning funding required is then a matter of calculating the "R" value by comparing the sum of the ratios for the individual radionuclide activity results to the values contained in Appendix B of 10 CFR 30, or State equivalent, and then comparing the calculated "R" value to 10 CFR 30.35(d) or State equivalent.

WPM-A.3 Comparison of Two Techniques for Measuring Gamma Dose near Berkeley Lab Accelerators

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At Lawrence Berkeley National Laboratory, we measure gamma radiation dose to the environment from two accelerators, the 88-Inch Cyclotron and the Advanced Light Source, using two techniques: a time-integrated approach that employs aluminum oxide dosimeters left in place for three months at a time and a real-time approach that uses energy-compensated Geiger-Mueller chambers to collect and display data every few seconds. At each location, dosimeters and Geiger-Mueller detectors are collocated in an environmental monitoring shack near the site boundary about 50-60 m from the accelerator. In addition, a reference station is located on Panoramic Peak about 0.6 km from Berkeley Lab, well beyond the influence of lab operations. While the annual dose to the environment from Berkeley Lab accelerators only occasionally exceeds natural background levels, a comparison of the annual dose over the past five years as measured by these two techniques indicates that they track closely. Furthermore, the gamma dose of record (the dose from Geiger-Mueller detectors) reported in the annual site environmental report may slightly overestimate the actual environmental dose.

WPM-A.4 Count Rate Limitations in Pulsed Accelerator Fields

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This presentation discusses various concepts involved in the counting losses of pulse-counting health physics instrumentation when used within the pulsed radiation environments of typical accelerator fields, in order to pre-establish appropriate limitations in use. Discussed are the 'narrow' pulse and the 'wide' pulse cases, the special effect of neutron moderating assemblies, and the effect of pulse micro-structure on the counting losses of the pulse-counting instrumentation. In the 'narrow'-pulse case, the accelerator pulse width is less than or equal to the instrument's pulse-pair resolving or dead time; whereas in the 'wide'-pulse case, the accelerator pulse width is significantly longer than the instrument's pulse-pair resolving or dead time. Examples are provided which highlight the various concepts and limitations.

WPM-A.5 Neutron Operational and Protection Quantity Conversion Coefficients Under ICRP-26, ICRP-60, and ICRP-103

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Conversion coefficients relate particle fluence to absorbed doses in phantoms and the body. There are two broad categories of dosimetric quantities: protection quantities defined by the ICRP and operational quantities defined by the ICRU. Ideally the operational quantities would be measurable and provide conservative estimates of the protection quantities. The most notable changes in the calculational methodologies of these quantities in the past 35 years are changes in the phantoms used to approximate the human body, the development of a risk-based system that utilizes organ and tissue risk weighting factors, changes in the weighting factors applied to various radiations to account for the relative detriment, and revisions to the quality factor versus linear energy transfer of radiation in water. In this work an overview of changes in the three primary ICRP recommendations of interest – ICRP-26 (1977), ICRP-60 (1991), and ICRP-103 (2007) and the impact of the changes on the protection quantities for neutrons is given. Since the operational quantities are commonly used as estimators of the protection quantities a review of these quantities, notably the ambient and personal dose equivalent, is also given for comparison.

WPM-A.6 Large-scale Production of Mo-99 Using a 100-kW Proton Beam

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A concept for energy-efficient, large-scale production of Mo-99 and other useful short-lived fission products using low-enrichment uranium has been developed. The concept, a Compact Accelerator-driven Multiplier for Isotopes (CAMI), uses a small, nearly spherical array of low-enrichment uranium that contains a minimal quantity of U-235 at 19.9% of the total uranium mass. The CAMI is irradiated with neutrons generated by irradiation of an external target with a beam from an accelerator, with initial simulations assuming 200-MeV protons with ~100 kW beam power. The multiplication of this sub-critical array is such that the beam power of 100 kW produces 1.2 MW of fission power resulting in a high yield of Mo-99 per gram of U-235. The simulations indicate that 100% of the current U.S. demand of 40,000 Ci per week of Mo-99 can be delivered by this configuration. This production mechanism can provide a reliable domestic supply of Mo-99 using well understood technology and do so without the use of highly enriched uranium. Two other accelerator-based production mechanisms are currently being investigated, one based on photo-fission of U-238, and the other based on the photo-nuclear reaction (gamma,n) on the stable isotope Mo-100. Because these reactions require very high beam power they are limited to providing a relatively small fraction of the weekly supply of Mo-99. This work was supported by the U.S. Department of Energy, Office of Nuclear Physics, under Contract No. DE-AC02-06CH11357.

WPM-A.7 Validation and Verification of MCNP6 as a New Simulation Tool Useful for Medical Applications

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During the last decade, we have developed at LANL improved codes of the Cascade-Exciton Model (CEM) and of the Los Alamos version of the Quark-Gluon String Model (LAQGSM) to describe reactions induced by particles and nuclei. We have tested our CEM and LAQGSM codes against a large variety of experimental data on particle-particle, particle-nucleus, and nucleus-nucleus reactions and have compared their results with predictions by other models. The latest versions of our codes, CEM03.02 and LAQGSM03.03, have been incorporated recently as event generators in MCNP6, the latest and most advanced LANL transport code representing a merger of MCNP5 and MCNPX, which can be a useful tool for simulations needed for proton and heavy-ion treatment of cancer, medical isotope production, and other medical applications. Here, we present a brief description of CEM03.02 and LAQGSM03.03 and several illustrative results by MCNP6 with our event generators for both thin and thick targets of interest to medical applications.

WPM-A.8 A New Method to Measure Potential Accelerator Hot-Spots

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As every accelerator has a life-cycle, it is always necessary to prepare for their eventual demise. In the case of the CAMD synchrotron ring at Louisiana State University, the lifecycle might indeed be shortened as a result of difficult economic times. Louisiana is an agreement state but any decommissioning plan must follow the multi-agency documents put out by the EPA. Although, the guidance documents known as Marssim and Marsame will eventually be used, it is necessary also to do some preliminary work so that budget and planning can begin. Recently, we have been effective in locating hot spots in the facility immediately following accelerator operation by retrieving spent photographic film, placed in areas of suspected neutron production. The film is used as a flexible ruler, so that its position can always be duplicated. Once the film has been retrieved, the maximal radiation is located using a pin hole in an 8x8 inch piece of lead coupled with a GM detector equipped with a pancake probe with a thin mica window. This allows us to determine the highest point of radiation, along the measured length [1cm increments] of the piece of film. The flexible ruler is then placed back in the precise position and the radiation hot spot can be determined and considered for additional studies to alleviate the problem. With a half-life of only 2.39 minutes for 108^Ag(m), the film is ready for reuse within 30 minutes. This quick, inexpensive measurement method has been an invaluable assessment tool in our facility. We plan to use this method for preliminary planning on what areas require particular attention for planning our decommissioning strategy for both our Linac and Storage Ring.

WPM-B.2 A Comparison of InLight Reader and MicroStar Reader Performance

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The Landauer InLight system is an automated laboratory grade instrument designed for accredited laboratories while the microStar system is manually operated and designed for laboratories needing to obtain rapid field data. A study to compare the dose readings from these two systems was performed to demonstrate the equivalency of the readers. The InLight LDR Model 2 dosimeters exposed at Pacific Northwest National Laboratory's Calibration, Research, and Accreditation facility (n=255) were processed on both readers. The sources of exposure were chosen from the following categories: Accident category photons, General Photons, Beta particles, Photon mixtures, Beta and photon mixtures, and Protection category neutron and photon mixtures. The bias, standard deviation, and performance quotient were calculated for both readers and compared on a standard deviation vs. bias performance plot in accordance with the ANSI N13.11-2009 standard. Agreement was found between both readers for Hp(10) and Hp(0.07). Several intercomparison studies have been conducted at Landauer and yield similar results. The average difference between the InLight and microStar readings for Hp(10) and Hp(0.07) were within 4%. A dosimetry system based on either reader will pass the ANSI N13.11-2009 NVLAP performance test with similar tolerance limits for all testing categories.

WPM-B.3 Development of an On-line Radiation and Detection Measurements Lab Course

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An on-line radiation detection and measurements lab is being developed with a grant from the U.S. Nuclear Regulatory Commission. The on-line laboratory experiments are being designed to provide a realistic laboratory experience for a student that cannot be on campus. This paper presents four on-line experiments: external gamma-ray dosimetry, gamma-ray spectroscopy, alpha spectroscopy/alpha absorption in matter, and nuclear electronics. The student will access the experiments through a broad-band internet connection. A webcam will be set up to stream the experiment live so the student can observe the physical instruments and receive visual feedback from the system in real time. Interactive National Instruments (NI) LabVIEW™ programs provide data acquisition control, experimental control, and live data display with real-time updates of all experiments. A LabVIEW™ program communicates with the URSA-II (SE International, Inc.) data acquisition system, which controls the detector bias voltage, pulse shaping, amplifier gain, and ADC. Detector and amplifier output pulses may be displayed with another LabVIEW™ program for the digital oscilloscope (NI USB-5132). Additional LabView programs are used to control the positions of all sources with stepper motor controllers (VXM-1, Velmex) and adjust pressure in the alpha chamber with a digital vacuum regulator (DVR-200, J-KEM, Inc.). For each experiment, several LabView programs may be integrated into one interface as necessary to provide seamless functionality to multiple instruments.

WPM-B.4 Comparing LS System Detection for Liquid, Cherenkov, and Nitrogen Scintillations

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Liquid scintillation (LS) counter are conventionally used for counting radiations from a sample by mixing an aqueous sample with an equal amount of organic scintillation cocktail, but counts can also be obtained without adding the cocktail by detecting Cherenkov radiation in water or nitrogen fluorescence in air. The current interest in rapid sample measurements when responding to incidents motivated this more extensive evaluation of the two latter processes to determine which of the three techniques can achieve counting sensitivity that meets action guides when measuring gross activity. Liquid samples, solid samples, and wipes were tested. The techniques were also compared for measuring selected radionuclides, such as Am-241, Cs.137, Sr-89 and Sr-90, after radiochemical separation. Settings of upper and lower energy response discriminators are reported for each technique.

WPM-B.5 Radioanalytical Criteria for Emergency Response

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During a radiological emergency, radiochemical laboratories likely will be faced with the problem of needing to analyze more samples than they can handle using their routine environmental measurement protocols. Some of the factors that constrain sample throughput include sample count time, sample preparation time, available laboratory equipment, and available personnel. In addition, when confronted with more samples than the laboratory can handle, the laboratory manager needs guidance on how to set priorities for sample analysis. The Environmental Protections Agency (EPA) is developing a series of media specific – water, air, and soils and sediments documents to provide guidance to address the first two issues and on setting priorities for analysis. This paper focuses on the process for establishing data quality objectives (DQO), measurement quality objectives (MQO), and the derived minimum detectable concentrations (MDC) for the analysis of water that may have been contaminated with radionuclides. The document presents a default set of MQO. Actual MQO, always will depend upon events and may need to be modified by the incident commander to better address a particular event. There are logic flow diagrams with identified decision points, similar to the logic diagrams used in developing computer algorithms. The specific details in the flow diagrams refer to the default MQO, primarily in the form of required method uncertainties, for analyzing the radionuclides of concern in water. Three radioanalytical scenarios are presented for water potentially contaminated with radionuclides. Two assume that the radionuclides are unknown. The third scenario, where the radionuclides have been identified, assists the laboratory manager in establishing the priority for processing samples based on the gross concentration screening values for the specific radionuclides.

WPM-B.6 Occurrence of Natural Radionuclides in the Drinking Water Supplies of Shiraz and Spring Waters of Fars Province

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Fars province is located in the south-west region of Iran where different nuclear sites are being established, such as Bousher Nuclear power plant. In this research, 92 water samples from the water supplies of Shiraz city and springs of Fars province were investigated with regard to the natural radionuclides the concentration of natural radioactive elements, total uranium, Ra-226, gross alpha and gross beta. Ra-226 concentration was determined by the Rn-222 emanation method. To measure total uranium concentration, a laser flourimetry analyzer (UA-3) was used. The average concentration of Ra-226 in Shiraz's water resources was 22.6 mBq/l, while 93% of spring waters have a concentration lower than 2 mBq/l. The results of uranium concentration measurements show average concentrations of 8.4 microg/l and 5.94 microg/l in the water of Shiraz and springs of Fars respectively. The gross alpha and beta measured by the use of evaporation method were lower than the limit of detection of the measuring instruments used in this survey.

WPM-B.7 Natural and Artificial Radioactivity Distribution in Soil of Fars Province, Iran

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Fars province is a populated large province located in south west of Iran. In recent years, different nuclear sites are being established in the neighboring provinces, so it is necessary to perform comprehensive environmental monitoring programs in this province. This work presents a study of natural (K-40, U-238, Th-232) and artificial (Cs-137) radioactivity levels in soil samples of Fars province. For this purpose, 126 samples were gathered from different regions of the province. The samples were analyzed by Gamma spectroscopy to quantify K-40, Cs-137, Th-232 and U-238 radioactivity concentrations using an HPGe detector. The results of this investigation show the average concentrations of 270.51 Bq/L, 8.5 Bq/L, 14.9 Bq/L and 26.34 Bq/L K-40, Cs-137, Th-232 and U-238 in Fars soil respectively. Finally baseline maps were established for the concentrations of each of the radionuclides in different regions of the province. The absorbed dose rate and the Annual Effective Dose Equivalents were also calculated for the radionuclides according to the guidelines of UNSCEAR 1988. The Average Annual Effective Dose Equivalents (AEDE) of this province was found to be 39.94 microSv. According to the results, no region was found with the concentration above the standards of the world.

WPM-B.8 Uranium in Phosphate Fertilizer using Different Analytical Techniques

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Manufactured phosphate fertilizers and their agricultural applications are considerable sources of environmental pollution. In this study, composite samples of phosphate fertilizer (PF) of different physical forms (granular, G, and water soluble powder, L) were collected from the local market of Riyadh City. The activity concentration of Uranium-238 in Bq/kg was measured using gamma ray spectrometer, Alpha particle spectrometer after chemical separation and inductively coupled plasma- mass spectrometer (ICP-MS). The main aims of this study were to evaluate PF quality according to its physical form and manufacturers (local, L, or imported, I), and the expected hazardous impacts of long-term phosphate fertilization of sandy soil. The results of uranium determination using different analytical techniques were compared. There was significant variation in uranium concentration. The annual addition of uranium to soil due to P fertilization was calculated. Our previous study indicates that the average uranium concentrations in cultivated and uncultivated soil samples did not indicate any variation due to long-term phosphate fertilization of sandy soil. These indicate that uranium added to the soil via phosphate fertilization may be redistributed to the subsurface soil layer and/or to shallow underground water.

WPM-C.1 Advanced Radiological Scanning Technologies Produce Superior Survey Results

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Land area walk over surveys performed by survey technicians, using "standard" hand-held detector systems are wrought with potential human error and inefficiency. Usually unintentional, survey technicians can misinterpret or simply neglect information provided to them by the instrument they are operating. Increased survey cost and the increase in future human exposure are a potential result of missing or mis-interpreting this information. Technically-advanced overland gamma survey systems incorporate automated data logging, a high-sensitivity global positioning system for survey location identification, computer-controlled data management and storage, environmentally-stabilized detectors to eliminate detector spectral drift, and gamma spectroscopy with background stripping capability. This technology can virtually eliminate the vast majority of human errors associated with old-fashion hand-held surveys and increase survey efficiencies. However, these survey benefits can present interesting regulatory dilemmas. The use of a technically-advanced survey system at a site involving an oversight organization that is unsure of how to interpret the results leads to an unwillingness to accept the technical merit of surveys performed. Regulators may require confirmatory surveys be conducted in parallel using technicians with hand-held instruments to validate the systems results, or in extreme cases, the oversight organization may simply rejected the use of this advanced survey equipment and refuse to accept the instrument's results. Now is the time to advance overland survey technology into the 21st century and to bring regulatory bodies up-to-speed. Regulatory acceptance of current technology will virtually eliminate the human error associated with overland radiological surveys performed with hand-held instruments, while providing a superior survey result at a cheaper cost to the client.

WPM-C.3 Computer Program Simulation of a Moving Alpha or Beta Particle Detector Across a Contaminated Surface

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A computer program was developed to estimate performance characteristics of a moving detector scanning for alpha or beta radiation emitting from a contaminated surface. The program performs Monte Carlo simulations of surveys of the contamination with the detector at each possible starting location. The program uses counting statistics applicable to radiation measurements and information from the user about the detector (e.g. dimensions, detection efficiency), the contamination (e.g. activity, dimensions), the survey parameters (e.g. speed of the detector, length of counting interval, background radiation). For each simulated survey, the highest measurement of the contamination is recorded. At the conclusion of all of the simulations, the recorded measurements are compiled and a distribution of measurements of the contamination that takes into account all possible starting locations of the detector is produced. From this distribution, the program develops a distribution of probabilities of false negatives as a function of the decision level. At the completion of the program, the distribution of false negatives is plotted on a graph along with the distribution of false positives, derived from the background radiation, length of counting interval, and appropriate counting statistics. All the distributions are saved in a file that can be used by a spreadsheet program. These results are useful in planning surveys for surface contamination where the size of the contamination is not large compared to the active area of the detector.

WPM-C.4 Verification of Dose Correction Factors of MOSFET Dosimeters for Use in Anthropomorphic Phantom to Measure Equivalent Doses and Effective Dose

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Recently MOSFET dosimeters which are very small and provide practically real-time reading were used, in an anthropomorphic phantom, to measure equivalent doses and effective dose. However, because it is mainly made of silicon and epoxy, which is not ideally tissue equivalent, the MOSFET dosimeter has some energy dependence and overestimates absorbed dose in the phantom due to the existence of low-energy scattered photons. To accurately measure organ doses, the dose correction factors of the MOSFET dosimeter at various dosimeter locations in the physical phantom were determined by Monte Carlo simulations with MCNPXTM 2.5.0. To verify the dose correction factors, this study measured the organ doses and effective dose of ATOM adult male phantom for two reference radiation fields using LiF thermo-luminescence dosimeters (TLDs) and MOSFET dosimeters respectively. A total of 38 high-sensitivity MOSFET dosimeters were used to measure the organ doses for Cs-137 and Co-60 radiation fields and then the measured values were compared with those measured by the same number of TLDs for the same irradiation condition. The measurement results of the MOSFET dosimeters, for which the dose correction factors were not applied, overestimated the dose by 10-20%, i.e., in comparison with the results of TLDs. After applying the dose correction factors, the results of the MOSFET and TL dosimeters agreed within 5% of difference for the most organs, and the measured effective doses agree within 3-4% considering both Cs-137 and Co-60 radiation fields.

WPM-C.5 Recent Progressive Developments of Radioactivity Measurement Techniques - A European Perspective

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It is true that the development of radiation measuring methods and instruments, since the beginning of this century, has lead only rarely to spectacular or surprising innovations. However, quite interesting improvements have been achieved with principally known measuring procedures and instruments, guided frequently by the requirements of new regulations. These also have influenced the direction of development of measuring instruments in the past years, not only with regard to their physical performance, but also to data processing and presentation as well as to a user-friendly and error-free application. Another factor that gains more and more importance is the cost effectiveness of measurements. In particular, due to the unified regulations issued by the European Union (EU), the role and the tasks of measurements for radiation protection may well be seen different in Europe as compared to US. Partly under some of those aspects, the paper will deal with the following measurement techniques: - Low-level Alpha and Beta Activities - Alpha Spectrometry - Gamma Spectrometry - Liquid Scintillation Counting The paper closes with a dream of a measuring laboratory head about possible future developments apt to make his daily labor easier.

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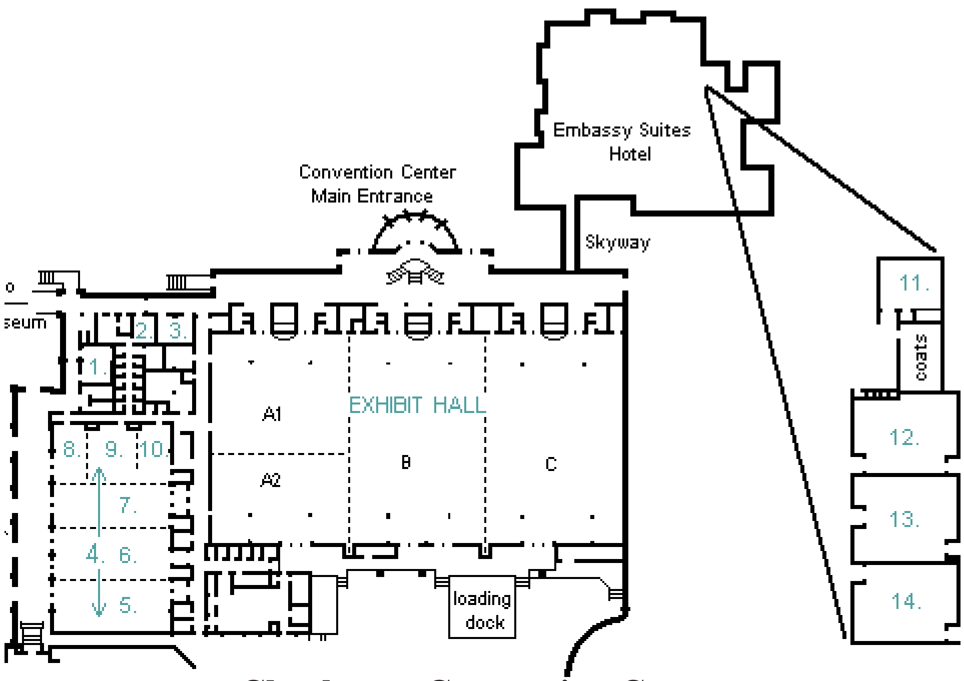

Charleston Convention Center
Main Floor

Convention Center Floorplan Key

1. Room 1
2. Room 2
3. Room 3
4. Ballroom
5. Ballroom A
6. Ballroom B
7. Ballroom C Hall
8. Ballroom C1
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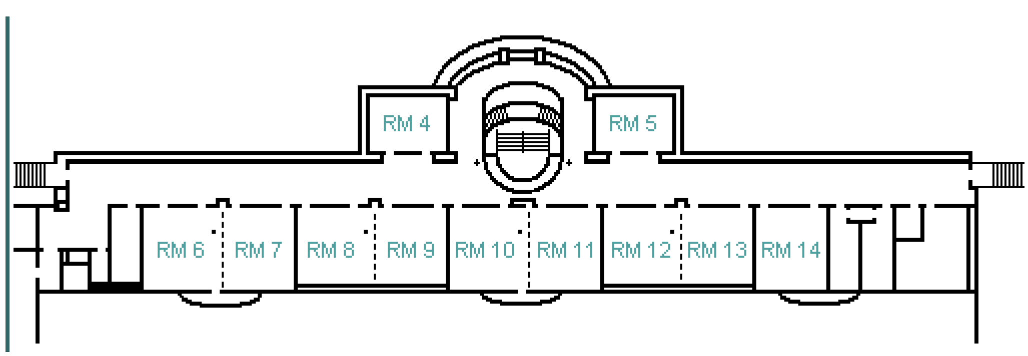

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