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We Have a Lot to Talk About

By Larry Satkowiak
INMM President

This is a busy time of the year for everyone. The weather is finally changing, spring is here in the U.S., and folks are spending more time outside, gardening and doing yard work, preparing for summer. It is also a busy time year for the INMM, holding workshops and seminars and, of course, preparing for our Annual Meeting this July.

2015 Workshops

The year started out with the Spent Fuel Seminar in January. This was the thirtieth edition of the workshop and it continues to be relevant as the nuclear industry continues to struggle with issues surrounding the disposition of commercial spent nuclear fuel. In March, the Nonproliferation and Arms Control Technical Division sponsored the 8th Workshop on Reducing the Risk from Radioactive and Nuclear Materials. The topics addressed by this year's workshop included; public perception of the risk from nuclear and radioactive materials, the impact of changing U.S.-Russia relations on the risk from nuclear and radioactive materials, strategies to mitigate the insider threat at nuclear facilities, and cybersecurity as an element of reducing the risk from nuclear and radioactive materials.

Developing the Annual Meeting Program

In early March, the Technical Program Committee (TPC) Meeting and Executive Committee (EC) Meeting were held. The Technical Program Committee has developed another outstanding program with more than 400 papers and presenters from twenty-nine countries. The TPC process for developing the technical program is described in Jack Jekowski's Taking the Long View column in this issue of the *JNMM*. Having participated in the TPC for nearly fifteen years, it is impressive how many abstracts get sorted and placed into the program in a relatively short time by a group of dedicated, very focused volunteers. Stay tuned; as the technical program develops there will be email announcements and reminders from INMM headquarters. Note to all speakers — due to popular demand (FYI — we *do* read and take member feedback seriously!) and an improving budgetary situation, the speaker's breakfast has been reinstated.

Communications Summit

As INMM president, I have the honor and pleasure of representing the Institute at a variety of events and meetings. In February, I was invited to participate in the Radiation Dose Communications Summit co-hosted by the American Nuclear Society (ANS) and the Health Physics Society. Sixteen different organizations from around the world participated, including professional, technical, and medical societies. The goal of the meeting was to attempt to establish a common language for the industry to communicate about radiation, its effects and benefits, and to bring clarity and correct information to public discussions. As with any communications regarding a technical subject, messaging is very important. The discussions were very illuminating and are continuing via email.

Looking Forward

A joint INMM/ANS Workshop on Safety-Security Risk Informed Decision Making will be held on April 26 at Sun Valley, Idaho, USA. One of our partner organizations, the European Safeguards Research & Development Association (ESARDA), is holding its 37th Annual Meeting on May 18–21 in Manchester, England. Another partner organization, the World Institute for Nuclear Security (WINS), is holding an International Best Practice Workshop on Nuclear Material Control and Accountancy in Support of Nuclear Security in Vienna, Austria on June 16–17. And of course, let's not forget our own INMM 56th Annual Meeting to be held at the Renaissance Indian Wells in Indian Wells, California, USA, on July 12–16.

No Shortage of Discussion Topics

So much is happening in the nuclear materials management world — the nuclear deal with Iran, future cooperation with Russia, the NPT Review Conference, nuclear power expansion and its effects, nuclear terrorism, the next Nuclear Security Summit, safeguards, detection enhancements, export control, future of arms control, nuclear trafficking, forensics, nuclear facility management, security issues, education/training, inventory controls, verification challenges, etc. We look forward to seeing everyone at this year's Annual Meeting. We have a lot to talk about.

Larry Satkowiak, President
Corey Hinderstein, Vice President

Providing Up-to-date Information

By Dennis Mangan
INMM Technical Editor

This issue, which I believe is one that provides much up-to-date information, begins with an interesting article by our President Larry Satkowiak, that he appropriately titled, *We Have a Lot to Talk About.*

The first technical article, *Certified Reference Material 116-A,* authored by several technologists from New Brunswick Laboratory (Argonne, Illinois, USA), Dokuz Eylul University (Torbali-zmir, Turkey), Y-12 National Security Complex (Oak Ridge, Tennessee, USA), and Savannah River National Laboratory, (Savannah River, South Carolina, USA), discusses the replacement of the original Certified Reference Material (CRM) CRM 116U metal standard with CRM 116-A. These Certified Reference Materials are used to ensure the quality of safeguards measurements to meet the needs of the nuclear safeguards and nonproliferation communities. A replacement for CRM 116U was deemed necessary after it was determined that this material had eroded to the extent that it was no longer suitable as a standard. Considerable effort, which is discussed in much detail, was needed to have a successful replacement.

The second technical article discusses *An International View on He-3 Alternatives for Nuclear Safeguards.* This article highlights the results of a U.S. Department of Energy small international workshop held at the U.S. Los Alamos National Laboratory in mid-2013 followed by a second workshop in mid-2014. Although He-3 detection has been a major non-destructive analysis tool, an immediate replacement is questionable, and considerable effort still remains to have a replacement. Potential alternatives are discussed.

The third technical article, *Outcomes of Information Analysis Technologies, Techniques, and Methods for Safeguards, Nonproliferation, and Arms Control Verification Workshop,* by Gastelum and Kreyling of the Pacific Northwest National Laboratory (Richland, Washington, USA), discusses the results of a workshop of this topic in mid-2014 in Portland, Oregon, USA. The workshop was hosted by the Pacific Northwest Chapter of the INMM and had the support of the two INMM Technical Divisions, the International Safeguards Technical Division, and the Arms Control and Nonproliferation Technical Division. Discussions were held on the status of Information Analysis status and needs in several organizations which led to plans established and needed for future efforts.

The fourth technical article, *Global Views of Spent Fuel Storage and Disposal,* authored by Carlyn Greene of the Ux Consulting Company LLC, Roswell, Georgia, USA, is a comprehensive summary of the status of spent fuel storage and disposal within many countries around the world. This summary is interesting and up-to-date in the various countries discussed, and reflects discussions at the 30th Annual INMM Spent Fuel Seminar held in Washington, DC, USA in January 2015.

The final technical article is *Spent Fuel Management in Spain,* by David Garrido Quevedo, ENSA, Maliano, Cantabria, Spain. This article focuses in detail the efforts in Spain dealing with spent fuel management.

Book Review Editor Mark Maiello presents an excellent book review on *Arms for Uncertainty Nuclear Weapons in U.S. and Russian Security Policy*, by Stephen J. Cimbala.

The last article is *Taking the Long View in a Time of Great Uncertainty*, by Jack Jekowski, *JNMM* Industry News Editor and chair of the INMM Strategic Planning Committee. In his article, he concentrates on "Making Sausage – A view into Creating the Annual Meeting Technical Program." Any member of the Technical Program Committee will endorse Jekowski's thoughts on how the Technical Program Committee meeting is conducted as the Annual Meeting Technical Program is formulated.

Certified Reference Material 116-A: A New High-enriched Uranium Metal Standard

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Abstract

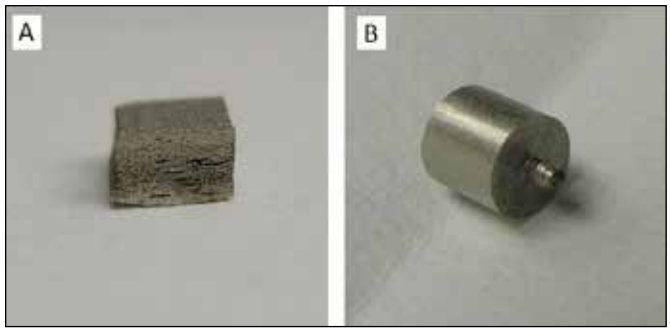
Certified Reference Material (CRM) 116-A is a uranium amount content (assay) and isotope amount ratio (isotopic composition) standard for nuclear material safeguards/nonproliferation analyses. CRM 116-A was specifically created as a replacement for the CRM 116 U metal standard. Each unit is comprised of a 1.1-g metal pellet of 93 percent ^{235}U enriched uranium. The certified U amount content value of the CRM is (0.99945±0.00014) g U • g metal^{-1} with the quoted uncertainty at the 95 percent confidence level. The amount content value was determined by High Precision Titrimetry and is directly traceable to the national metrology database for the United States. To verify the certified value and to assess the potential for sample-to-sample heterogeneity, the amount content for CRM 116-A was also measured using the Davies and Gray titrimetry and isotope dilution mass spectrometry techniques. Data from the verification analyses are consistent with the certified value but indicate that, at the limits of measurement resolution, there may be sample-to-sample variability. Accordingly, the GUM-compliant uncertainty estimate for the amount content includes a factor to account for potential sample-to-sample variability. This factor was determined by performing a second set of High Precision Titrimetry analyses on six 1.1-g CRM units. A discussion of the certified U isotope amount ratio determinations is provided in a separate publication.

Introduction

Certified Reference Materials (CRMs) that are traceable to a national metrology database and to the International System of Units (S.I.) are used to ensure the quality of safeguards measurements. For example, U and Pu CRMs characterized for amount content and isotope amount ratios play a crucial role in the calibration and testing of measurement equipment used by nuclear analytical facilities.[1] CRMs are also used to assess the accuracy and precision achieved by the analytical facilities using routine assay and isotopic measurement methods or developing new measurement techniques.[2] Some CRMs can also be added directly to unknowns as isotopic tracers for the purpose of making traceable concentration measurements or to correct for bias in mass spectrometric measurements.

The U.S. Department of Energy's New Brunswick Laboratory (NBL) has the responsibility for maintaining and certifying special nuclear material CRMs for the United States.[3] As part of this mission, NBL has characterized and certified a new high-enriched uranium (HEU) metal reference material for isotope amount ratio and amount content (assay). This standard, designated CRM 116-A, is primarily intended to meet the needs of the nuclear safeguards and nonproliferation communities and was produced as a direct replacement for CRM 116, which was used by the United States nuclear complex and the international safeguards community (e.g., References 4 and 5). As a highly enriched U metal, CRM 116-A is an isotopic and assay reference material for analysis of special nuclear material (SNM) categorized at a high attractiveness level[6] and can also

Figure 1. (A) Unit of CRM 116 after acid cleaning. The metal chip is approximately 1.5 g in mass and 0.5 cm x 0.5 cm x 0.3 cm in dimension. The dark spots forming roughly parallel bands in the metal chip are voids created by dissolutions of oxide during cleaning. (B) Machined pellet of CRM 116-A after it had been acid cleaned. The small protrusion at the top of the cylinder is a stub remaining from cutting the individual cylinder. The mass of the cylinder is approximately 1.1 g, the diameter is 0.43 cm, and the length is 0.41 cm.



serve as a starting material for isotope dilution tracers. This report provides a description of the reference material production process and the amount content characterization effort. A detailed description of the isotope amount ratio characterization and certification is provided in a separate publication.[7]

A replacement for CRM 116 was deemed necessary after it was determined that the stock of this material was no longer suitable as an amount content standard. The individual units of CRM 116 were produced in 1974 by rolling HEU metal into an approximately 2-mm thick billet that was then cut into 1.5 g metal chips. The certificate for the CRM 116 amount content standard was issued in 1978 and included a certified amount content value determined by titrimetry and the isotope-amount fraction for ^{235}U.[8] In 2002, NBL staff members were performing routine verification and packaging activities to prepare CRM 116 stock material for sale when it was observed that the metal pieces had developed planar structures (Figure 1A) that tended to form cracks and zones of oxidation. These defects made it difficult to consistently remove oxide from the metal prior to quantitative dissolution, resulting in unreliable U amount content determinations.

The replacement for CRM 116, which has been designated 116-A (Figure 1B), was produced at the Y-12 National Security Complex by machining and stamping pieces of a cast metal "log." The production method for the CRM was developed with the intent of minimizing the potential for the degradation observed in CRM 116 samples. A representative sampling of the CRM production run was analyzed using several U amount content methods. The analytical scheme, analysis methods, and data evaluation for this project were planned and executed to meet the requirement of international standards for analytical reference materials.[9-11] The resulting data was evaluated to determine attribute values with uncertainty estimates that are compliant with the Guide to Uncertainty in Measurement (GUM).[12]

Methods

Material Production

A study performed at the Y-12 National Security Complex demonstrated that HEU stock material, in the form of a standard metal "161 casting" (hollow cylinder of U metal, colloquially referred to as a "log"), is statistically homogenous for the "vast majority" of elemental contaminants.[13] Of the ninety-one elements for which the U metal was tested, only four were determined to be heterogeneous and the maximum observed difference between analyses for any element was 150 µg • g metal^{-1}. Based on this work, it was determined that a portion of a HEU metal log from the Y-12 stocks would be a suitable starting material for the CRM 116-A U metal standard. An appropriate log was selected (AP3C19-YW-2055) and the trace element data for the chosen log indicates a maximum sample-to-sample difference for any single element of 11 µg • g metal^{-1}. A portion of this log was machined at Y-12 to produce approximately 500 1.1-g metal pellets (Figure 1B) and another 1-kg of the metal was reserved for future production runs.

The production of the HEU metal pellets included six major steps:[14] 1) Metal wedges were cut from the designated U metal casting. 2) The resulting metal wedges were machined to produce rods that were 1.3 cm in diameter and up to 18 cm long. 3) These rods were then swaged down to approximately 0.64 cm in diameter and annealed in a furnace. 4) The annealed rods were further swaged down to a diameter of 0.43-cm and parted into approximately 10-cm long pieces. 5) The 0.43-cm rods were then turned on a lathe to remove any scale or surface irregularities. 6) As the final machining step, the rods were cut on the lathe to produce a series of 0.41 cm long pellets that comprise the individual 1.1-g CRM 116-A units.

As production proceeded, individual metal pellets were transferred into ten glass convenience bottles. Each bottle held approximately 50 g of metal and was numbered based on the order in which it was filled. After the pellet production was completed, Y-12 staff removed five pellets (totaling ~5 grams) from each of the convenience bottles and transferred each subset of pellets to another, separate convenience bottle.

Table 1. Samples sets analyzed for CRM 116-A certification work. The "Stock Bottle" numbers refer to the Y-12 identifier for the stock container associated with each sample or sample set. The "12EU0070" sample number is the NBL LIMS number for the original CRM 116-A characterization samples. Each of the 12EU0070 sample sets consisted of five metal pellets that were processed and distributed for various analysis methods. The "CRM 116-A" samples listed in the table were selected from packaged CRM units created from the bulk material provided in the Y-12 stock bottles.

Stock Bottle	NBL No.	Mass (g) (N=5)	NBL No.	Mass (g) N=1
LH-V2NH	12EU0070-01	5.6	C116-A 00001	1.11
LH-V2NJ	12EU0070-02	5.6		
LH-V2NK	12EU0070-03	5.6	C116-A 00040	1.08
LH-V2NL	12EU0070-04	5.7		
LH-V2NM	12EU0070-05	5.7	C116-A 00086 C116-A 00087	1.11 1.03
LH-V2NN	12EU0070-06	5.6		
LH-V2NP	12EU0070-07	5.7	C116-A 00137	1.09
LH-V2NR	12EU0070-08	5.7		
LH-V2NT	12EU0070-09	5.5	C116-A 00182	1.08
LH-V2NV	12EU0070-10	5.7		

These pellets were segregated as representative characterization samples that encompass the entire production run. The ten containers of characterization samples were then shipped to NBL where the material was given internal tracking numbers corresponding to the filling order of the sample bottles (Table 1).

Five of the stock convenience bottles were later shipped from Y-12 to NBL for creation of CRM units. Each CRM unit produced was assigned a unique number with the form C116A 0000X. As with the characterization samples, the first group of CRM 116-A units was produced from the first stock bottle to be filled, the next group from the second bottle, and so on. After being cleaned with acetone (see analysis preparation section below), the pellets were individually packaged to create the CRM units. Each pellet was placed into a small (2.5 cm x 2.5 cm) zip-top plastic bag which was closed, folded, and sealed in a labeled 5-mL Teflon® snap-cap vial that was, in turn, placed in a Mylar® bag and heat-sealed. The heat-sealed sample was then placed into outer packaging, comprised of a cardboard cylinder, which was labeled and sealed with shrink wrap. Subsequent to completion of the packaging project, six of the individual CRM units were selected for additional characterization analyses.

Analysis Preparation

CRM 116-A samples were analyzed using the NBL High Precision Titrimetric (HPT) method to determine a certified value for U amount content. For verification purposes and to assess potential heterogeneity, the material was also analyzed using the NBL-modified Davies and Gray titrimetric method (D&G) and Isotope Dilution Mass Spectrometry (IDMS). The results from the different methods were compared to one another and to amount content values that were determined by Y-12 using a "difference" method.[15]

The characterization samples consisted of ten sets of five U pellets with each pellet having a mass of approximately 1.1 g. A distribution plan for these samples was devised so that analyses by HPT, IDMS, and D&G would be performed on samples from throughout the production run and that results from each analysis technique could be compared for individual samples or samples from the same characterization set. Note that the primary HPT analyses performed for this study required three sample pellets (~3 g of material) for each analysis. Accordingly, ten HPT samples were created by taking three pellets from each characterization set and another two composite HPT samples were created by combining single pellets taken from six of the ten sets. All D&G, IDMS, and isotopic analyses were performed on aliquots of solutions created from individual 1.1-g U pellets. After the initial amount content analytical work was completed, six of the packaged 1.1-g CRM units were selected for additional HPT analyses. As with the 3-g samples, these units were selected to encompass the breadth of the initial production.

Analysis samples and pellets packaged to create the CRM units were rinsed with acetone to remove residual cutting oil from the metal. The metal pellets were transferred from sample containers into new borosilicate glass beakers (10-mL to 25-mL sizes) that had, at a minimum, been cleaned with a Micro-90® [16] cleaning solution and rinsed with deionized water (18 MΩ). A volume of reagent-grade acetone, sufficient to cover the metal, was poured into each beaker and the pellets were allowed to soak for approximately two minutes. The acetone was decanted from each beaker then the samples were rinsed with additional acetone, which was also decanted to waste. The samples were then left to air dry before being prepared for mass determination or packaging.

For the characterization samples, the last processing step prior to sample weighing and dissolution was to acid clean the metal pellets. Each pellet was transferred to a pre-cleaned

borosilicate beaker and soaked in 8 mol • L^{-1} nitric acid until the metal surfaces appeared to be free of oxide. The samples were rinsed with deionized water and dried with acetone before the final quantitative mass determination. The mass of each sample or sample set (for the ~3-g titration samples) was determined on a calibrated and checked Mettler AT 20[17] analytical balance (readable to 0.000002 grams) by weighing each sample repeatedly until a stable mass reading was achieved (agreement within 0.000006 g). During weighing, the laboratory temperature, atmospheric pressure, and relative humidity were periodically recorded so that standard buoyancy corrections could be applied to the measured masses. The weighed samples were then dissolved in accordance with the NBL procedures for HPT or D&G methods (see below).

High-Precision Titration (HPT)

HPT analyses were performed on 3-g samples (N=12) of the U metal to determine a mean attribute value for amount content. A second set of HPT determinations, made on individual 1 g samples (N=6), was performed to estimate an upper bound for sample-to-sample variably. Only a general description of the HPT procedure is provided in this report along with any details specific to the CRM 116-A project. Full descriptions of the HPT method can be found in other references.[18-19]

Each HPT sample was dissolved in a new, 400-mL borosilicate beaker that had been cleaned with Micro-90® laboratory soap and fluxed with nitric acid. The metal was dissolved using 50 mL of concentrated phosphoric acid, two drops of 2 percent potassium dichromate, and approximately 1 mL of 48 percent HF (as needed to completely dissolve the metal). Uranium in the resulting solution was then reduced to the U(IV) valence state by adding a prepared ferrous sulfate solution. Any excess Fe (II) was then eliminated by a molybdate-catalyzed oxidation reaction with nitric acid. In turn, the nitrous acid produced by this reaction was eliminated by adding sulfamic acid to the solution. The prepared solution was then diluted with cold (4°C) deionized water and vanadyl sulfate was added to increase the rate of the titration reaction. The titrant used for the HPT analyses was a potassium dichromate certified reference material[20] (NBL CRM 99) that was primarily added to the U solution as a solid. The appropriate quantity of solid titrant was estimated based on the mass of the metal dissolved for each sample. The mass of solid titrant was carefully determined on a calibrated Mettler AT 20 balance by weighing repeatedly until a stable mass reading was achieved (agreement within 0.000006 g). The small amount of potassium dichromate necessary to complete the titration was added in the form of a quantitatively prepared solution dispensed from a squeeze bottle. The quantity of solution was determined by the difference in the mass of the squeeze bottle as measured before and after the titration. The U solution was titrated to a final endpoint of 600 ± 15 mV as measured using a saturated calomel reference electrode and a platinum wire electrode. The quantity of U in the solution was determined based on the oxidimetric potential of the potassium dichromate (w_{CRM99}) and the total amount of titrant added ($\rho_{CRM99} \times M_{CRM99}$). The mass of U in the solution was then calculated by multiplying by the atomic weight of the HEU ($A_{r(C116\text{-}A)}$), which was previously determined[7] to be (235.18572 ± 0.00011) g • mol^{-1}. Finally the U amount content (w_U) was derived by dividing the mass of U by the buoyancy corrected mass of the metal (M_{metal}) dissolved to create the solution (Equation 1).

$$\frac{\{\rho_{CRM99} \times M_{CRM99} \times w_{CRM99} \times A_{r(C116-A)}\}}{M_{metal}} = w_U \quad (1)$$

A variety of protocols were incorporated into the HPT measurement scheme to minimize the potential for systematic biases and to assure a high level of quality. In addition to calibration and validation of the balance prior to use, a simulated weighing experiment was performed to assess the accuracy and precision of the Mettler AT 20 balance used for the most sensitive mass measurements. The instruments used for measuring laboratory temperature, pressure, and humidity are calibrated annually by the manufacture or by an ISO 17025 accredited laboratory. To capture any analyst-to-analyst or day-to-day variability, the highest precision 3-g HPT measurements were performed by two analysts with each performing the analyses over the course of two days. Carefully prepared samples of U metal CRM 112-A[21-22] were analyzed along with the unknown samples to assure that the method was in control. The HPT measurements performed on the 1.1-g samples were made to assess sample-to-sample variability, so these measurements were performed during one day by an individual analyst.

NBL Modified Davies and Gray Titration

The average mass of U in each D&G analysis aliquot was approximately 25 mg, IDMS samples were 10 mg, and mass spectrometry sample loads were 5 μg or less. The CRM 116-A samples processed for analysis had an average mass of 1.1 g. Therefore, only a single 1.1-g pellet from each of the ten

characterization samples was quantitatively dissolved to produce sample aliquots for all D&G, IDMS, and/or thermal ionization mass spectrometry (TIMS) analyses. The dissolution of the 1.1-g pellets was performed in new acid-cleaned 400 mL borosilicate beakers. Approximately 25 mL of 8 mol • L^{-1} nitric acid and 2 mL of concentrated sulfuric acid were added to the sample beakers that were then placed on a steam bath. After the dissolutions were complete, the solutions were quantitatively transferred to new, acid-cleaned 250-mL glass bottles and diluted to a concentration of 5 mg U / g solution.

Only a general description of the D&G method used for this study is provided below (extensive details for this analytical procedure are provided elsewhere[23-25]). Six of the dissolved 1.1-g samples were analyzed by D&G for uranium amount content. Four aliquots of each sample solution were created with each aliquot containing approximately 25 mg of U[26]. The aliquots were dispensed to pre-cleaned 400 mL borosilicate beakers and were then dried on a steam bath. The initial step in the D&G titration process was to re-dissolve the samples by adding 13-15 mL of distilled water to the beaker. A Teflon-coated magnetic stirring bar was placed into the beaker, then 40 mL of pre-mixed phosphoric acid reagent (~1 mL of 6 percent potassium dichromate in 2.2 L phosphoric acid) was added, followed by 5 mL of 1.5 mol • L^{-1} sulfamic acid, and 5 mL of 1 mol • L^{-1} ferrous sulfate to reduce all of the U to the U(IV) valence state. Next, 10 ml of nitric acid oxidizing reagent was added to eliminate any excess Fe(II) and the solution was allowed to stabilize for three minutes. After this period, 100 mL of 1 percent sulfuric acid was added to eliminate any nitrous acid, then 75-125 mg of solid vanadyl sulfate was added as a catalyst. A saturated calomel reference electrode and a platinum wire electrode were inserted into the solution. Then the stirrer rotation rate was set to create a vortex in the solution. A standardized titrant solution was added to the sample until a potential of 580 to 605 mV was reached. The titrant solution used for the D&G analyses was created from the National Institute of Standards and Technology (NIST) potassium dichromate Standard Reference Material (SRM) 136f.[27] A Titrant Equivalency Factor (TEF) was determined for the SRM 136f solution by repeated titrations of a precisely known solution produced from CRM 112-A U metal. The standardized titrant solution was dispensed using a squeeze bottle and the mass of solution dispensed for each titration was determined from the mass difference of the dispensing bottle as measured on a Mettler AT400[17] balance, readable to 0.1 mg. To determine the quantity of U in the titrated solution, the mass of the titrant solution ($M_{SRM136f}$) was multiplied by the TEF, which was determined by the analyst specifically for the batch of SRM 136f solution used in this study. The TEF yields a mass value based on a natural U isotopic composition, so the calculated mass of U has to be corrected for the relative difference in atomic weight between the U in CRM 116-A ($A_{r(C116\text{-}A)}$) and CRM 112-A ($A_{r(C112\text{-}A)}$). Finally, the corrected amount content (w_U) value for each titration is determined relative to the mass of dissolved metal (M_{metal}) in

$$\frac{\left\{M_{SRM136f} \times TEF \times \frac{A_r(C116-A)}{A_r(C112-A)}\right\}}{M_{metal}} = w_U \qquad (2)$$

As with the HPT measurements, the analysis scheme for the D&G measurements was devised to minimize the potential for systematic biases and to assure a high level of measurement quality. The goal of the D&G measurements was to verify the HPT measurements and to detect any sample-to-sample variability. Accordingly, three replicate measurements were made for each dissolved sample (the fourth aliquot of each was created as a spare) but only one analyst performed the titrations and the order of titrations for each sample was randomly distributed over the course of two days. Each day of titrations was initiated with a reagent "blank" measurement and the measurement of a carefully prepared known solution. If the known solution measurement was in control, then the CRM 116-A samples were analyzed along with three quality control samples. All of the balances utilized were in calibration and were validated prior to use.

Isotope Dilution Mass Spectrometry

In general, the isotope dilution method of amount content determination is well established. More detailed descriptions of the specific procedure used for U IDMS at NBL are provided elsewhere.[28-29] IDMS amount content measurements were made for six of the ten previously processed CRM 116-A metal pellets, with two IDMS mixes prepared for each analyzed sample. The dissolution of the metal was performed as described in the D&G section, above, and a carefully prepared solution of the CRM 112-A metal (normal isotopic composition) was used as the tracer. A batch of 20-mL scintillation vials was cleaned and soaked in a nitric acid bath in preparation for use as sample containers for the IDMS mixtures. Between 1.3 and 1.5 mL of the tracer solution was dispensed to each vial with the mass determined by replicate weighings on a calibrated Mettler AT

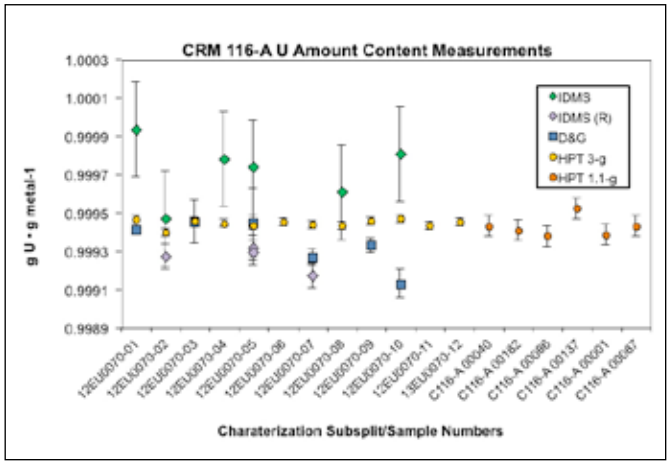
Table 2. CRM 116-A measurement data for samples analyzed in this study. HPT data represent individual measurements. D&G data points are the mean of three replicate measurements and IDMS data are the mean value of duplicate measurements. Note that the sample standard deviations for individual D&G samples are determined for the replicate data on that sample. The summed data in the table is based on the full data sets and not on the mean data for individual samples. † The expanded uncertainties listed in this table are estimated for the independent data sets and incorporate observed variability as well as Type-B evaluated components such the expanded uncertainty for titrant, spike, and calibration CRMs but do not incorporate uncertainty factors for potential sample-to-sample variability.

Characterization Sample Fraction	HPT (3 g) g U•g metal^{-1}	D&G g U•g metal^{-1} Std Dev	IDMS g U•g metal^{-1}
12EU0070-01	0.999466	0.9994144 0.000028	0.999936
12EU0070-02	0.999400		0.999472
12EU0070-03	0.999458	0.9994551 0.000112	
12EU0070-04	0.999446		0.999782
12EU0070-05	0.999434	0.9994481 0.000180	0.999740
12EU0070-06	0.999451		
12EU0070-07	0.999441	0.9992702 0.000041	
12EU0070-08	0.999436		0.999610
12EU0070-09	0.999456	0.9993314 0.000038	
12EU0070-10	0.999471	0.9991303 0.000073	0.999807
12EU0070-11	0.999434		
12EU0070-12	0.999453		
Average Standard Deviation	0.999445 0.000019	0.999342 0.000126	0.999733 0.000246
Expanded Uncert †	0.000062	0.000062	0.00080

Sample No.	HPT (1-g) g U•g metal^{-1}	Characterization Sample Fraction	IDMS (R) g U•g metal^{-1}
C116-A 00040	0.999431	**12EU0070-02**	0.999274
C116-A 00182	0.999409	**12EU0070-05 (1)**	0.999318
C116-A 00086	0.999379	**12EU0070-05 (2)**	0.999295
C116-A 00137	0.999523	**12EU0070-07**	0.999172
C116-A 00001	0.999387		
C116-A 00087	0.999431		
Average Standard Deviation	0.999427 0.000052	**Average Standard Deviation**	0.999265 0.000064
Expanded Uncert †	0.000078	**Expanded Uncert †**	0.00080

201[17] balance, readable to 0.01 mg. Each aliquot contained approximately 9 mg of U. The solutions in the scintillation vials were dried on a steam bath. Once the CRM 116-A solutions were prepared, the vials were re-weighed on a Mettler AT 201 balance, 2.5-mL aliquots of CRM 116-A solution were added (10 to 12 mg of HEU per aliquot), and the vials were weighed again to determine the mass of solution. The solutions were

Figure 2. Measurement data for sample analyses of CRM 116-A performed for this study. HPT data represent individual measurements. D&G data points are the mean of three replicate measurements of the samples. The IDMS and IDMS (R) data are the mean value of duplicate measurements. (R) indicates re-analysis samples. The error bars for the HPT and IDMS data represent the sample standard deviation of the complete data set for that specific method. The error bars for the D&G data represent the sample standard deviation of the data set for the individual samples. Note that the error bars for the 3-g HPT data points are approximately the same size as the data symbols.



equilibrated by drying on a hot plate, adding approximately 1 mL of 1 mol • L^{-1} nitric acid and 0.2 mL of 30 percent peroxide to each vial, and drying again. Then, approximately 2 mL of 8 mol • L^{-1} nitric acid was added to each vial and dried. Both equilibration steps were repeated and the solutions were dried for a final time. The IDMS sample mixes were prepared for mass spectrometry by dissolving the U in a volume of 1 mol • L^{-1} nitric acid sufficient to produce a concentration of 0.25 mg • mL^{-1}. Each of the twelve IDMS solutions was then analyzed, in duplicate, for the $^{235}U/^{238}U$ isotope amount ratio by TIMS using a Total Evaporation method[30-31] on a MAT 261 multi-collector mass spectrometer.[32] All of the sample analyses were interspersed with analyses of CRM U630 [33] for mass bias correction and CRM U500[34] for quality assurance. To determine the amount content of the U metal, appropriate variables were input to a verified spreadsheet that incorporates a standard IDMS equation.[35] The input variables included the mass bias corrected $^{235}U/^{238}U$ isotope amount ratio, aliquot masses, tracer solution concentration, and U isotopic abundances of CRM 116-A and CRM 112-A.

Following completion of the CRM 116-A certification project, NBL staff performed an additional set of IDMS measurements. Several of the solutions created for the CRM 116-A characterization samples were re-analyzed for amount content.

Four IDMS analyses were made, in duplicate, on three of the CRM 116-A analytical solutions using a newly prepared tracer solution with a depleted U isotopic composition, CRM 115.[36-37] Otherwise, the processing and mass spectrometry procedures for the re-analyzed samples were essentially identical to the original analyses.

The IDMS analysis plans for the initial analyses and the re-analyses were designed to assure measurement quality and to avoid the influence of any spurious data. All mass measurements were made on balances that were in calibration and verified before use. Two IDMS aliquots were measured for each sample, the isotopic composition of each IDMS solution mixture was measured in duplicate, and quality assurance analyses were incorporated into the TIMS measurement plan to assure instrument performance and to verify that appropriate mass bias corrections were applied.

Results

The results for the CRM 116-A amount content measurements are shown in Table 2 and Figure 2. The 3-g HPT data are highly repeatable and indicate an average amount content of 0.999445 g U • g metal^{-1} with a standard deviation of 0.000019 g U • g metal^{-1} (0.002 percent relative). The data for the 1.1-g HPT analyses has a mean value of 0.999427 g U • g metal^{-1} with greater variability resulting in a standard deviation of 0.000052 g U • g metal^{-1} (0.005 percent relative). CRM 112-A quality control samples, analyzed with the HPTs samples, indicated that the measurements were in control, with average relative deviations from the certificate value of +0.0043 percent for the 3-g samples and -0.0052 percent for the 1.1-g samples.

The average value for samples analyzed by D&G indicate a lower amount content value of 0.999342 g U • g metal^{-1} with a standard deviation of 0.000126 g U • g metal^{-1} (0.013 percent relative). The D&G analysis technique, however, allows for several measurements from individual samples. The average variability for the within-sample D&G measurements is 0.000079 g U • g metal^{-1} (0.008 percent relative), which is significantly smaller than the variability for the D&G data set as a whole. The D&G quality control "unknowns" indicated that measurements were in control on both days, with a mean relative deviation from the target value of +0.034 percent.

The initial IDMS analyses indicate an amount content of 0.999733 g U • g metal^{-1} and a standard deviation of 0.000246 g U • g metal^{-1} (0.025 percent relative) but the reanalyzed IDMS measurements indicate a significantly lower mean value of 0.999230 g U • g metal^{-1} with tighter standard deviation of 0.000053 g U • g metal^{-1} (0.005 percent relative).

The HPT analyses performed on 3-g samples represent average values due to the fact that analyzed solutions were made from multiple pellets. The results from these measurements are, however, very precise and students t-test statistical analyses[38] of the data (Table 3) demonstrate that there is no statistically significant day-to-day or analyst-to-analyst differences. The 1.1-g HPT analyses and the D&G and IDMS verification analyses were performed on solutions or aliquots of solutions created from individual pellets. In fact, the 1.1-g HPT analyses were performed specifically to assess repeatability at the level of individual CRM units. The mean amount content value for the 1.1-g HPT samples does not indicate a statistically significant difference from the 3-g HPT results (Table 3) but the variability of these measurements is nearly 2.5 times larger. Although both the 1.1-g and 3-g analyses are highly repeatable, HPT yields only a single amount content determination per sample. So, the variability in the 1.1-g HPT data could be indicative of sample-to-sample differences however it is not possible to isolate the effects of systematically lower measurement precision associated with smaller HPT sample sizes[39] from variability that is the result of heterogeneity. Accordingly, the variability of the 1.1-g HPT analyses represents a conservative estimate of the magnitude of sample-to-sample variability that could be due to heterogeneous U metal.

The D&G analysis data can provide a more quantitative assessment of homogeneity. The distribution of the D&G values determined for this project appears to indicate that the samples are heterogeneous at the 1.1-gram unit size. An Analysis of Variance (ANOVA)[40] for the data sets from individual samples indicates that, at the 95 percent confidence level, there is statistically significant sample-to-sample variability (Table 3). The distribution of IDMS analysis results also appears to show variability but the repeatability and uncertainty of the IDMS measurements are not sufficient to reliably resolve sample-to-sample differences in the amount content of U at a level of less 0.05 percent relative. Furthermore, the initial IDMS data set and the re-run IDMS analyses indicate that there is a small but recognizable systemic bias for the analytical method, as implemented. Finally, it is noteworthy that there are four samples for which both D&G and IDMS determinations were made. Only one of these samples (12EU0070-07) shows the same magnitude or direction of bias relative to the mean value for that measurement method (Figure 2).

Table 3. The data in Table 3 are from Students t-test and Analysis of Variance (ANOVA) statistical assessments. If the absolute value of the t-statistic or the F-statistic is smaller than the t Critical or F Critical values then the data sets do not indicate a statistically significant difference at the 95 percent confidence level. The P-value indicates the calculated level of significance for the data sets analyzed. A P-value of 0.05, or lower, corresponds to a statistically significant difference between data sets at the 95 percent confidence level. The abbreviations for the analysis methods are as defined in the text.

Data Sets Evaluated (t-test)		*t-statistic*	*t Critical 1* tail / 2 tail	*P-value 1* tail / 2 tail	Notes
Day-to-Day	HPT	-1.05	2.13/2.77	0.18/0.35	6 per day
	D&G	0.07	1.75/2.12	0.47/0.94	9 per day
Analyst-to-Analyst	HPT	-1.57	1.81/2.22	0.07/0.15	6 per analyst
Method-to-Method	HPT-HPT	-0.85	1.94/2.44	0.21/0.42	HPT 3-g to 1.1-g
Data Sets Evaluated (ANOVA)		***F-statistic***	***F Critical***	***P-value***	**Notes**
Sample-to-Sample	D & G	5.25	3.11	0.009	6 Units: 3 per sample

Discussion

A CRM is defined as "material characterized by a metrologically valid procedure for one or more specified properties, accompanied by a certificate that provides the value of the specified property, its associated uncertainty, and a statement of metrological traceability."[41] Detailed requirements for production of a certified reference material are outlined in ISO Guide 34[10] and ISO Guide 35.[11] These requirements include evaluation of 1) material stability, 2) homogeneity, 3) metrological traceability, 4) reproducibility, and 5) the assignment of GUM compliant measurement uncertainties. The certification project for the CRM 116-A was planned and executed specifically to meet these requirements.

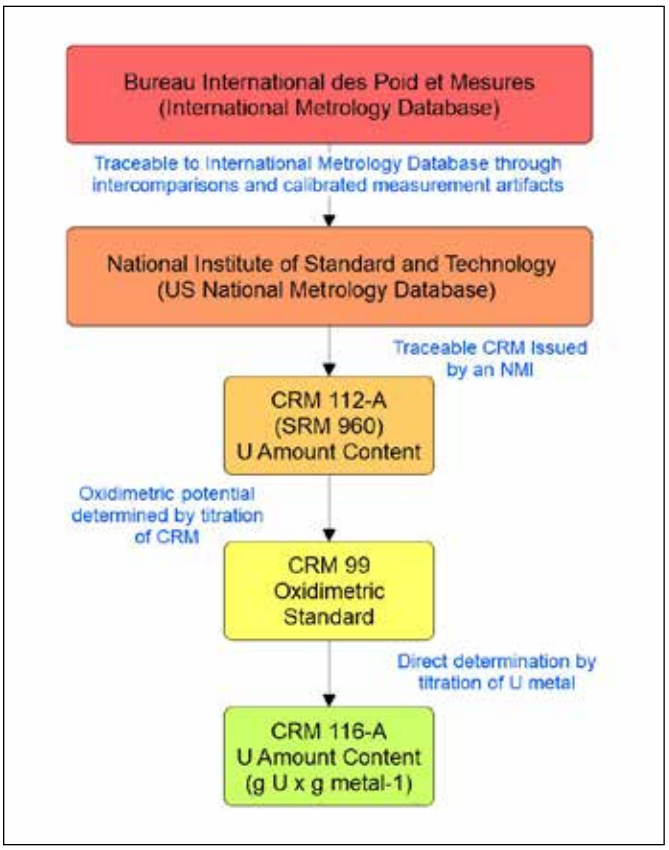
1) Reference Material Stability

Under normal storage conditions, the CRM 116-A units should last indefinitely. NBL has considerable experience with U metal standards, including CRMs 112-A, 115, and 116. Although U metal forms an oxide surface coating when exposed to air, the bulk of the metal material remains intact under the coating. Accordingly, the certificates for NBL U CRMs include specific instructions for removing oxide prior to use for quantitative analyses. After the oxide has been properly removed, U metal samples of all sizes (e.g., 4-g rods of CRM 112-A, 1-g pieces of CRM 115) consistently yield reproducible results, even after storage for durations of twenty years or more (the current CRM 116-A production batch is anticipated to last ~ten years). The original C116 material is the notable exception to this observation but minimizing the potential to form planar cracks and internal zones of increased oxidation was the primary motivation for the six-step production method used to create the CRM 116-A units. Furthermore, CRM 116-A has been packaged in a manner that will limit exposure to oxidizing conditions, prevent any oxide coating from being dispersed, and will prevent rattling that might lead to enhanced degradation of the sample due to mechanical processes.

2) Reference Material Homogeneity

For all practical purposes, CRM 116-A is homogeneous. Although there is some data indicating potential heterogeneity between the 1.1-g sample units, the observed variability is at the limits of the high precision techniques used in this study (maximum observed sample-to-sample differences of 0.014 percent to 0.03 percent). The D&G titrations performed on 1.1-g samples appear to show statistically significant sample-to-sample bias with a maximum observed difference of about 0.03 percent but the observed bias of samples in the D&G sample data is not confirmed by either a similar magnitude or the relative direction of bias in IDMS data from the same samples (e.g., Sample 12EU0070-01, -05, and -10). Also, there is no evidence for systematic variability across the production run based on the 1.1-g HPT analyses nor is there any systematic variability for the more precise 3-g HPT analyses. It is noteworthy that the samples analyzed by Y-12 for the HEU stock material show little variability despite being from a broader distribution of sampling locations and indicate an amount content value that is indistinguishable from the HPT data (0.999445 g U • g metal^{-1} with a 0.000008 g U • g metal^{-1} standard deviation). Finally, a detailed analysis of U isotope amount ratios for the same set of characterization samples indicates that the metal is isotopically homogeneous.[7] Although the isotopic data do

Figure 3. Schematic diagram showing the traceability chain for the HPT method as outlined in this study. Only data from the HPT analyses were used to determine the certified attribute value for amount content in CRM 116-A. "NMI" is National Metrology Institute as defined by Bureau International des Poid et Mesures (BIPM).



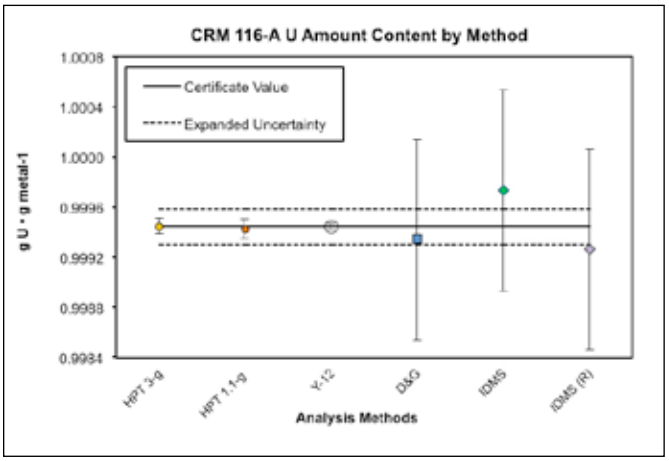
not preclude variability in the amount content, it does demonstrate that the material is not a heterogeneous mixture.

The 3-g U HPT analyses were performed to provide the most precise and accurate average amount content value for the CRM. As previously noted, however, it is not possible to assess sample-to-sample homogeneity with the 3-g HPT samples but the variability of the 1.1 g HPT data may be consistent with limited sample heterogeneity. Therefore, the uncertainty estimate for the certified U amount content was conservatively modeled to account for potential sample variability. This was accomplished by incorporating the sample standard deviation observed for the 1.1-g HPT analyses (rather than the standard uncertainty) as a discrete variability component in the uncertainty estimate (see further discussion below).

3) Measurement Traceability

To establish the traceability of the certified attributes for CRM 116-A it is necessary to demonstrate that the values "can be related, with a stated uncertainty, to stated references, usually national or international standards, thorough an unbroken chain of comparisons."[42] The certified attribute described in this report, U amount content, has a fairly simple traceability chain (Figure 3) tied directly to the national metrology database for the United States as maintained by the National Institute of Standards and Technology (NIST, formerly National Bureau of Standards). The certified amount content of CRM 116-A was determined by direct titration using the CRM 99 potassium dichromate oxidimetric standard. CRM 99 was, in turn, calibrated against U metal CRM 112-A. Therefore, the amount content of CRM 116-A is directly traceable to CRM 112-A which was originally issued as SRM 960 by the National Bureau of Standards.

Figure 4. Mean amount content values as determined by various measurement methods. Error bars represent expanded uncertainties estimated for data sets associated with each analysis method. Limited data was provided by Y-12 therefore, the estimated expanded uncertainty (0.000023) is a minimum estimate and the resulting error bars are shown within the symbol. Note that the uncertainties for individual analysis methods do not account for any sample-to-sample bias (see Table 2). The certificate value for the CRM 116-A metal and the expanded uncertainty envelop are shown for purposes of comparison. "IDMS (R)" indicates re-analysis samples.



4) Attribute Value Reproducibility

The general accuracy and reproducibility of measured attributes can be assessed by comparing the results of independent measurements[43] (Table 2 and Figure 4). The amount content values for the twelve 3-g samples analyzed by HPT are, within expanded uncertainties, indistinguishable from the 1.1-g HPT, D&G, and IMDS analyses. Furthermore, the value based on the 3-g HPT analyses is identical (within rounding) to the value obtained by Y-12 for the starting material. It should be noted, however, that the HPT, D&G, and initial IDMS measurements

Table 4. Uncertainty budget and certificate value for U amount content of CRM 116-A as measured by HPT. The HPT amount content values for individual measurements are calculated using a verified, in-house spread sheet. The uncertainty was estimated using the GUM Workbench program. "W_u" is the mean U amount content value with an expanded uncertainty. "W_{umeas}" is the mean value and standard uncertainty of the 3-g HPT measurements. "δ_{Var}" is an uncertainty factor derived from the sample standard deviation of the 1.1-g HPT measurements and is included to account for potential sample-to-sample variability. "δ_{mass}" is an uncertainty factor to account for potential biases in the mass measurements of metal and the potassium dichromate titrant. "δ_{CRM99}" represents the combined standard uncertainty provided for the oxidimetric potential value of potassium dichromate CRM 99. Note that the potential contributions of buoyancy corrections were assessed and found to be insignificant (<< 1.0 percent total uncertainty value).

Quantity	Value	Standard Uncertainty	Degrees of Freedom	Relative Contribution
W_{umeas}	0.9994453	0.0000054	11	0.8 percent
δ_{Var}	0.0	0.0000519	5	73.6 percent
δ_{CRM99}	1.0	0.0000300	50	24.5 percent
δ_{mass}	0.0	0.0000064	∞	1.1 percent
W_u ($g\ U \bullet g\ metal^{-1}$)	**Value**	**Expanded Uncertainty**	**Coverage Factor**	
	0.99945	**0.00014**	**2.32**	
Uncertainty Estimate Equation	$(W_{Umeas}+\delta_{var}+\delta_{mass})\times\delta_{CRM99}=w_U$			

are not completely independent. CRM 112-A was used to calibrate the CRM 99 oxidimetric standard used for HPT; to determine a "Titrant Equivalency Factor" for D&G measurements; and as the spike for the original set of IDMS measurements. Accordingly, if the CRM 112-A has erroneous attribute values, the resulting CRM 116-A amount content measurements will be similarly biased. Aside from a common link to CRM 112-A, the three measurements methods were, otherwise, independent. For instance, the D&G and HPT analyses utilized different oxidimetric standards and IDMS is a fundamentally different measurement method for determination of concentration.

The second set of IDMS analyses were completely independent of the titration methods because CRM 115 was used as the tracer material. Also, the assay determinations by Y-12 were based on "Uranium by Difference" (Amount Content = Total Material - Impurities). Both of these independent data sets also indicate that the amount content for CRM 116-A is reproducible within stated uncertainties.

5) GUM Compliant Measurement Uncertainty

Uncertainty for the CRM 116-A amount content attribute value is provided as an expanded uncertainty and was calculated in accordance with the ISO Guide for the Expression of Uncertainty in Measurements[12] using the Workbench© software developed by Metrodata.[44] Expanded uncertainty (*U*) is the product of a combined standard uncertainty (u_c) and a coverage factor (*k*) necessary to achieve a desired confidence level, 95 percent for this determination. The combined standard uncertainty for the U amount content consist of Type A and Type B evaluated uncertainty components. These include the Type-A evaluated components for the variability of the 3-g HPT measurements expressed as the standard uncertainty for twelve measurements and the variability of the 1.1-g HPT measurements expressed as the standard deviation of six measurements (Table 4). The standard deviation of the 1.1-g sample data was used in the uncertainty calculation to account for low-level heterogeneity that could be present in the HEU material. Other potential Type-A evaluated uncertainty components, including those associated with weighing, are confounded in the observed variability of multiple measurements used to determine the certified value. The primary Type-B evaluated uncertainty component for the amount content is the certificate uncertainty provided for the oxidimetric potential of the CRM 99 used for the HPT titrations. Another Type-B evaluated component, to account for potential non-linearity in mass determinations, was incorporated into the uncertainty model but was found to make an insignificant contribution to the total uncertainty. The coverage factor (*k*) for the expanded uncertainty is the Student's t-factor necessary to provide a 95 percent level of confidence, which is based on the effective degrees of freedom for the measured value.

Conclusion

The certified amount content value and expanded uncertainty for CRM 116-A is (0.999445±0.00014) $g\ U \bullet g\ metal^{-1}$. The individual units of the CRM were specifically prepared to be a highly reliable HEU amount content reference material for use as a calibration standard, an isotopic tracer, or high-quality U base material. The amount content was determined using

traceable measurement techniques and a GUM compliant uncertainty is provided. Accordingly, the HEU metal characterized in this study is suitable for distribution as Certified Reference Material 116-A.

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An International View on ^{3}He Alternatives for Nuclear Safeguards

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Abstract

The U.S. National Nuclear Security Administration NA-241 Office of Nuclear Safeguards and Security hosted a small international workshop of invited experts at Los Alamos National Laboratory on the subject of ^{3}He alternatives for use in international safeguards June 25–27, 2013. Specifically, the workshop addressed neutron assay techniques that employ neutron coincidence counters for the verification of declared nuclear materials under safeguards and for monitoring purposes. This article provides in outline a record of the meeting and shares the spirit of the discussions and main observations and recommendations. The scene is a rapidly moving one and so this report only provides a snapshot in time. How the safeguards community will respond to a shortage of ^{3}He for use in neutron detection systems is not yet settled and moving forward we encourage broad participation in formulating both technical and policy options. Particularly perceptions may lag actual capability and there is an opportunity for researchers and developers to engage with the community to ensure decisions are well informed. The workshop concluded that no single technology can replace ^{3}He for all safeguards use cases and that performance metrics need to be re-defined for alternative neutron detection system and concepts of operation. The move from scientific feasibility to proven engineering solution requires a phase of practical demonstration under realistic routine safeguards conditions and many developing alternative technologies are reaching this phase. Well-coordinated demonstration was identified as the best way to move forward. Lastly, challenges associated with Figure-of-Merit definitions and proportional counter high-voltage plateaus were discussed and are addressed in this article. A follow-on workshop was held in the second half of 2014.

Introduction

Importance of ^{3}He in International Safeguards

The first of two planned technical workshops was held to review the status of ^{3}He alternatives and ^{3}He replacement technologies for international nuclear safeguards applications under a U.S. Department of Energy (DOE)/ National Nuclear Security Administration (NNSA) and EURATOM cooperative agreement. Specifically, technologies of interest address neutron assay techniques that employ neutron coincidence counters for the verification of declared nuclear materials under safeguards and for monitoring purposes. Twenty-three experts were invited to the meeting (reference the Acknowledgement Section for individual attendees) including two from the International Atomic Energy Agency (IAEA), one from the EURATOM inspectorate, three from the Japanese Atomic Energy Agency (JAEA), two from the European Commission's Joint Research Centre (JRC), one from a U.S. university, two from the DOE/NNSA headquarters, and twelve from the U.S. domestic national laboratories.

During the information sharing portion of the workshop, twenty-two presentations reviewed the emergence and response to the ^{3}He shortage, projected safeguards needs for neutron instrumentation, general requirements for successful alternatives, as well as specific detection technology options and the present status of capability demonstration. The group was then split into three breakout sessions and asked to make observations on the evolving landscape before reassembling to compare notes as a collective group.

This publication provides a summary of the deliberations so that a wider audience learns and benefits from the outcome of this particular meeting and importantly also contributes to the ongoing research and development (R&D) effort and associated dialog. Importantly we caution the reader that this short article is not intended to provide a detailed, thorough, or comprehensively referenced review of this expansive field. Many

of the aspects raised here are deeply nuanced and will need to be revisited. Our aim is to capture the spirit and feeling of the discussion. The two technical workshops under the mentioned cooperative agreement were in response to a meeting that was held at the IAEA in 2011. This work builds on previous efforts performed by workshop participants[1] and the publication of select outcomes from the initial IAEA meeting.[2]

The ^{3}He Supply and Demand

The U.S. supply of ^{3}He came about as a waste product of the U.S. nuclear weapons program; a radioactive decay product of the tritium that was produced. The nature of the production of this rare isotope dictated low pricing. With the termination of weapons production, the supply of ^{3}He became static. After the vast deployment of ^{3}He detectors, prompted by September 11, together with the collective demands for many other uses, such as neutron scattering science, it was determined that the supply was not sufficient to carry on business as usual, and alternative technologies to ^{3}He neutron detectors should be considered.

The ^{3}He supply picture for safeguards is not dire in the near term. Despite the common opinion that the ^{3}He problem has been "solved" at the time of this workshop the supply situation seemed unlikely to improve. Safeguards instrumentation uses significantly less ^{3}He than other pertinent uses, i.e., radiation portal monitors (RPMs) and neutron science facilities.[3] The opinion was expressed during the meeting that IAEA, Euratom, and the JAEA do not expect to need more than a few hundred liters a year for the next five to ten years. For high priority needs, small volumes such as this are likely to be made available. This relatively low expectation relies on existing instrumentation to serve current needs and the expected lack of additional reprocessing or mixed-oxide (MOX) facilities to come online in that timeframe. Although higher estimates have been made, predicting the need for 2,000 liters per year and thus forecasting that the official U.S. government allocation will run out in 2024. The higher estimates take into account demand direct from facilities for various monitoring purposes, and who often field joint use equipment and undertake waste assay. Despite the lack of supply, with today's safeguards requirements it is anticipated that in 2024 the market for ^{3}He will still prevail.

The IAEA is primarily depending on member states (in addition to some case specific internal research and development) to do the work necessary to find suitable alternatives to address nuclear safeguards needs. Ideally, these alternatives would be identified in time for any future increase in safeguards demand.

In addition to finding alternatives, incremental cost and resource savings gained by using less ^{3}He should remain a priority. More detector optimization work can be performed considering Monte Carlo simulation tools have improved since current systems were originally designed. Smaller diameter or lower pressure ^{3}He tubes combined with improved front-end electronics could achieve similar demand reductions without the additional complexity and the loss of the high-voltage plateau used for detector calibration and optimized performance.[4,5] Suggested ideas on the combination of ^{10}B and ^{3}He in a proportional counter have been attempted (such as those being pursued by GE Reuter-Stokes (GE R-S)[6] but have not yet proven to be better (than existing solutions) with respect to efficiency and use of resources. Such hybrids have been criticized on the basis that the well-defined desirable voltage plateau is lost and neutrons are wasted. Demonstrations under realistic conditions are needed to address these questions. Lastly, engineering new methods to recycle ^{3}He has become a popular option, not only with regards to radiation detectors, but also in medical applications.

The IAEA View on ^{3}He Shortages for International Safeguards

When the crisis over ^{3}He shortage first emerged, radiation portal monitors (RPMs) held most of the scientific community's attention, and concerns regarding safeguards were not fully addressed. Nuclear security applications of radiation detection are continually expanding their instrumentation in the field while safeguards instrumentation that is currently in use does not rapidly expand or change. Solutions for ^{3}He replacement in RPMs were studied in considerable depth and candidates were identified.[7,8] The scientific community is now recognizing that current solutions do not boast the same versatility gained with ^{3}He detectors. There is not a "one-size-fits-all" solution to all ^{3}He problems. Efforts to identify alternatives to ^{3}He in safeguards applications are necessary and justified. An application specific, case-by-case, strategy is one option.

^{3}He supports all aspects of nuclear safeguards in most all stages of the nuclear fuel cycle. The IAEA heavily relies on ^{3}He measurement systems for characterizing various forms of SNM. The primary use for ^{3}He in safeguards is neutron coincidence counting for attended non-destructive assay (NDA) measurements. The next large consumer, the Japanese MOX

fabrication plant, was allocated sufficient ^{3}He (1,000 liters) for most safeguards applications. For spent nuclear fuel characterization, the IAEA will introduce a lower precision ^{3}He alternative for a collar measurement system bolstered by other safeguards measures. For unattended NDA ^{3}He is primarily used for direction monitoring for continuity of knowledge, and some for item counting, ~50 liters per year. Assay uses are growing, particularly for glove-box monitoring and well counters for plutonium quantification. These are specialty requirements that take place in individual purchases. The projected need is ~150 liters per year.

Nuclear safeguards research and development also uses ^{3}He to develop novel safeguards systems and to improve current techniques. Research and development efforts must instead acknowledge the issues faced due to the supply and demand of ^{3}He and focus on developing appropriate alternatives.

The future of safeguards may bring changes that affect the projected demand and use of the ^{3}He supply. As the State-Level Concept (SLC) of safeguards gains acceptance and spreads, more frequent inspections may allow for reduced precision requirements for measurement systems. With reduced requirements, ^{3}He-alternative technologies may be more readily applicable for safeguards field use for specific use cases.

Alternative Technology Efforts

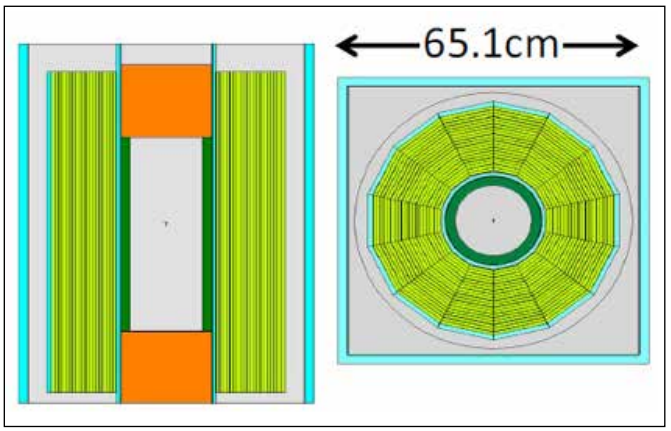
There are many ongoing research efforts that address ^{3}He replacement, several of which are arriving at prototype stage or are commercialized. The perception that there are many ways to detect neutrons and that one or more of them will prove viable must be tempered by the reality that few can compete with ^{3}He for the most demanding applications, and that considerable effort is needed to move from scientific feasibility to a reliable engineering solution. Some further discussion on the all-important concept of operation is provided in later in this paper. The IAEA, EURATOM, and JAEA have significant research and development efforts underway for safeguards that are complementary to what the U.S. Government supports, specifically represented at the meeting were the DOE/NNSA's Defense Nuclear Nonproliferation (DNN) Next Generation Safeguards Initiatives (NGSI) and Research and Development (R&D) offices. With many successful feasibility tests underway, attention has now shifted to matters of reliability, sustainability, and maintainability.

Replacement technologies to current ^{3}He systems are generally judged based on their efficiency, sensitivity to gamma rays, stability, dead-time, die-away time, scalability, safety, and sustainability.[9] Alternative technologies to current ^{3}He systems have the opportunity to perform beyond that of current systems in some regards by putting a high priority on measurement time and efficiency, footprint, ease of use, facility hazards, and transparency to vendors. After several years of effort to identify a sound alternative to ^{3}He, it was the feeling of the meeting that it has become clear that no single solution exists across the many applications that make up this problem. In nuclear safeguards research and development a set of distinct technologies have emerged for safeguards applications. We emphasize here that what we have included and omitted in this article is in no way intended to endorse nor to discount a particular endeavor. It only reflects the cross-section of technologies and discussion during this single workshop. Interested readers are strongly encouraged to consult the original scientific literature, speak with experienced and knowledgeable colleagues and vendors, and to attend the conferences where the latest thinking on these topics is being presented.

^{6}Li Capture Based Detection Technologies

The shortage of ^{3}He also led to a call for the development of ^{3}He-free alternatives to meet the large demand for RPMs, neutron scattering science, and handheld devices. In some ways RPMs are a less demanding application space than safeguards. For example, space restrictions may be less, the counting rates tend to be lower, only the singles rate is informative, some modest small level of double pulsing and cross-talk can likely be tolerated, and the gamma-ray insensitivity requirement may not be as restrictive in part because absolute accuracy is not needed if the goal is only to raise an alarm. On the other hand it could be argued that RPMs applications may be more demanding of gamma discrimination if neutron alarms trigger a significant response, and false alarms are not tolerated. They must also function acceptably over a wide range of environmental conditions. In general the time-window of interest is longer and the measurement times are shorter (tens of seconds) for RPMs. However, some of the technologies that have emerged from these security-focused programs could be transferrable. An example being pursued by Symetrica[8] and others is based on (electron/gamma-ray) thin ^{6}LiF:ZnS(Ag) sheets assembled into a layered structure comprising high-density polyethylene (HDPE) moderator, scintillator and wavelength shifting plastic for readout by photomultiplier tubes (PMTs). The neutron efficiency is scalable by both area and number of internal layers.

Figure 1. Multiplicity counter design utilizing ^{6}LiF/ZnS to replace the Epithermal Neutron Multiplicity Counter (ENMC)[10]



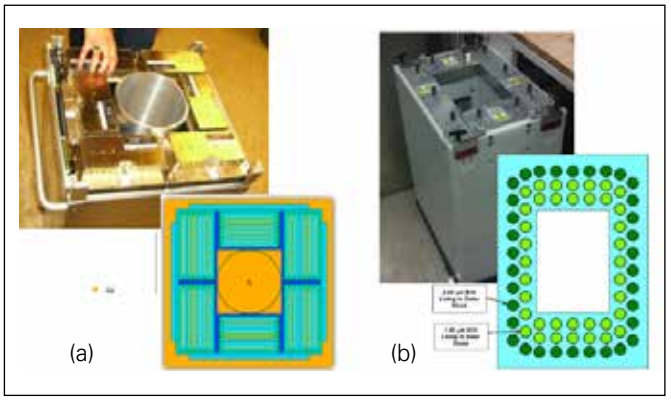
Test panels comprising modules roughly 10 cm wide by 100 cm long have demonstrated a neutron and gamma ray performance that readily meets RPM requirements. This group is now actively exploring designs based on this basic detection technology for a replacement High Level Neutron Coincidence Counter-II (HLNCC-II). Pacific Northwest National Laboratory (PNNL) has related technologies under development to replace the Epithermal Neutron Multiplicity Counter (ENMC)[10] and is in communication with Symetrica and others. Figure 1 shows the concept of a neutron multiplicity counter comprised of ^{6}LiF/ZnS panels. Current challenges with the ^{6}LiF/ZnS technology include real-time gamma-ray discrimination, pulse pile-up, and gain stabilization of the sensors to temperature variations.

^{10}B Capture Based Detection Technologies

Substituting the neutron capture cross section of ^{10}B for ^{3}He has been a preferred method for investigating ^{3}He alternatives. ^{10}B could be used in the gas form, BF_3, and directly replace ^{3}He gas in proportional counters conceptually similar to existing equipment with minimal infrastructure changes, that is to thinking, training, hardware and software. Unfortunately, in addition to a loss in detection sensitivity, it is popular opinion that BF_3 is too hazardous to be accepted for field use even with engineered mitigation. Alternate approaches have since focused on novel iterations of boron-lined proportional counters.

For boron-coated proportional counters, it seems that robust and uniform layers of, e.g., B_4C highly enriched (>96 at. % ^{10}B) can be reliably deposited. The optimum thickness is a trade off, and can be tailored depending on need, although one to two micrometers is a typical range. The first principle calculation of efficiency requires an estimate of the fraction of (n, α) events that result in pulses above the set threshold. This is currently approximated to be the fraction of (n, α) reactions that deposit energy above some threshold in the gas. These electronic efficiency factors differ significantly from unity (~0.7-0.5). Modeling is possible[11] but more experimental benchmark spectral data are needed to enebale general users to accurately and confidently predict performance.

Figure 2. a) Design and prototype of a High-Level Neutron Counter Boron (HLNB) with system parameters derived from the High-Level Neutron Coincidence Counter (HLNC-II)[12] and b) prototype of an Alternative Boron-Based Uranium Neutron Coincidence Collar (ABUNCL) designed to meet the performance of the Uranium Neutron Coincidence Collar (UNCL-I)[16]



The High-Level Neutron Counter Boron (HLNB) is based on parallel plate design consisting of detection chambers defined by boron-coated thin metal sheets interleaved with polyethylene for moderation, enclosed inside a sealed gas counter, shown in Figure 2.[12] The full counter, consisting of six modules manufactured by Precision Data Technology (PDT), underwent initial performance testing at Los Alamos National Laboratory (LANL) and was shown at the workshop. General Electric Reuter-Stokes (GE R-S), a company well known for ^{3}He proportional counters and other nuclear instrumentation manufacturing, [13,6] built a neutron coincidence collar prototype design called the Alternative Boron-Based Uranium Neutron Coincidence Collar (ABUNCL). The initial evaluation of this unit was performed by PNNL and included use of LANL mockup fresh fuel assembly.[14] The collar was then shipped to Oak Ridge National Laboratory (ORNL) for further study. In particular for end-plugs to be manufactured so that the units could be reconfigured into a well counter design. The JAEA approach was novel and ambitious, stepping outside of the standard boron-lined tube approach. The JAEA efforts involve the use of ceramic scintillator ribbons (ZnS/$^{10}B_2O_3$)[15] on glass plates mounted on a diagonal of a square reflecting boxes with photomultiplier

tubes at each end. Additional information on these and other approaches are available in the proceedings of the May 2013 ESARDA meeting and the July 2013 INMM meeting.

There was no presentation of the Proportional Technologies Inc. (PTI) straw detector technology at the workshop, although in the discussion sessions it was clear that the participants were acquainted with it as a candidate to replace ^{3}He-based systems for a number of applications. The approach adopted by PTI is to increase the surface area of boron carbide by deploying many small diameter (~4.4 mm being typical) proportional counters using a technology similar to the tracking devices deployed at ATLAS-CERN. Young and others have investigated how arrays of straws embedded into an HDPE moderator would operate as a well counter.[17] This strategy also helps to achieve a short die-away time key to reducing the influence of accidental or chance coincidences.

For thermal counters the evidence and feeling of the meeting was that several promising technologies had been identified and had attracted viable commercial interest.

Liquid-Scintillator Based Detection Technologies

As a general purpose replacement, organic scintillators have several perceived drawbacks, discussed further below. However, certain use cases exist (e.g., assay of fresh low-enriched uranium fuel) where liquid scintillators have a distinct potential advantage.[18] A number of characteristics of traditional liquid scintillators prevent them from being an obvious ^{3}He alternative in nuclear safeguards. During the discussion it was expressed that some traditional liquid scintillators are flammable, *thought of as* carcinogenic, and nuclear facilities operating under strict criticality controls often do not allow the use of liquids that have even a slight potential to leak. In terms of operational impact and how to manage it, the temperature stability of these detectors is still not fully understood. Note the perception that traditional liquid scintillators are carcinogenic is not in fact generally true. The most commonly used traditional solvents are toluene and xylene, neither of which is classified as carcinogens. While a minority of modern liquid scintillator solvents (e.g., naphthalene) are classified as "possible" carcinogens, the majority of them (pseudocumene, PXE) are not. This is an example of where long held perceptions need to be challenged against fact.

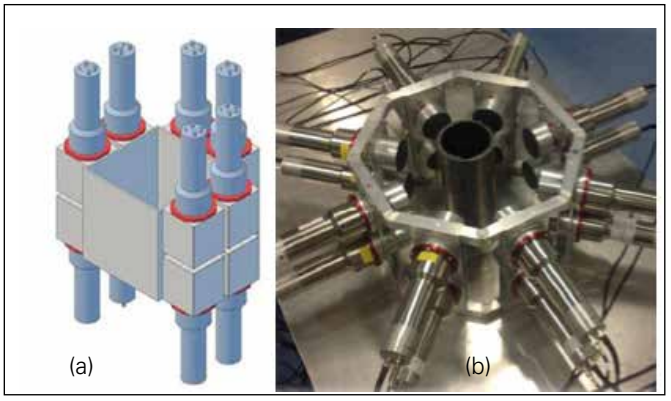
Beyond the direct disadvantages of the detectors, their ability to accurately predict SNM characteristics may be subject to significant uncertainty. The gamma-ray sensitivity of organic scintillators is relatively high, requiring a combination of both attenuation and pulse shape discrimination (PSD). The sensitivity of alternative technologies to gamma rays was a concern. The use of PSD-capable plastics as well as advanced stilbene crystals have become attractive alternatives, but at the time of the workshop, PSD performance from plastic scintillators and reliability of stilbene crystals were not yet as good as liquids. It was also clear that there is a need for a gamma-ray/neutron discrimination criterion for each category or application of safeguards. Additionally, the fast neutron detection efficiency is likely to be highly sensitive to matrix composition and density, especially hydrogen content. Special calculation tools (e.g., MCNPX-PoliMi)[19] need to be used to predict performance and to develop methods to compensate for the matrix effect. Coping strategies such as gamma shielding, bespoke electronics, optimized digital signal processing algorithms, and the benefits to be had from increased processing power are areas for ongoing study.

Despite the challenges and drawbacks of organic scintillation detectors, the theoretical attraction is twofold: first the insensitivity to neutrons below a specified energy threshold and second the low accidental rate. Insensitivity to thermal and epithermal neutrons allows them to detect fission neutrons during active neutron interrogation. To understand the accidental rate issue, it is important to first recap the situation for traditional thermal well counters. This issue is further developed later in this paper. The paper by Karpius et al.[20] on the optimization of light guides, reminds us that detector design is of interest to many measurement communities with innovative ideas passing in all directions. Considering the advancements made in the field of high functional density electronics could lead to significant advancements.

The deployment of safeguards systems based on organic scintillators will require revisiting coincidence counting electronics and algorithms. The timescale of operation is three orders of magnitude shorter than the current shift-register modules, resulting in reduced effective die-away time (in the conventional paradigm). With digital electronics the detectors have, in principle, zero dead time[21] but can only register a single detection per fission chain. Thus, new electronics, algorithms, and training will be needed to maintain a significantly different technology in the field.

The IAEA, in collaboration with the JRC, Ispra, and Hybrid Instruments UK, is evaluating the use of high flash point (~144°C) non-toxic liquid organic scintillators for use in neutron

Figure 3. Liquid scintillators used for neutron detection in a) a collar configuration and b) a multiplicity counter configuration[23]



coincidence counting.[22] Similarly, a multiplicity counter prototype was developed at the University of Michigan in collaboration with the Idaho National Laboratory that has undergone successful initial testing on gram-size samples of plutonium.[23] Early concepts and demonstration prototypes of both systems are shown in Figure 3. Considerable progress has been made on cell optimization and practical real-time digital PSD. Many challenges remain at the system level including treatment of cross-talk between cells (a neutron can contribute a count to two cells); how to assess and correct for gamma-ray misclassification; how to correct for item specific matrix perturbation; and how to engineer an easy-to-use thermally stable instrument.

Technical Topics Needing Further Development

In this section, we provide some discussion representative of the nature and flavor of the discussions that took place at the meeting.

Performance Metrics

When attempting to compare alternative technologies, many workers often turn to a simple Figure-of-Merit (FOM) that has been traditionally used to inter-compare thermal neutron ^{3}He measurement systems. The assumptions made in the process of defining FOMs for thermal systems need to be reconsidered when systems that have greatly differing performance, or when systems that use quite different detection mechanisms, are to be compared. In this section, the use of a FOM to compare new technologies is addressed and will start by recapping the situation for traditional thermal well counters.

Let S and D denote the singles and doubles count rate, much larger than the background, observed from a certain item using the familiar shift-register electronics logic with a coincidence gate width of T_g. The reciprocal variance of counting precision in the regime where accidental or chance coincidence dominates the real doubles rate (that is $S(ST_g) >> D$) has the functional dependence, $\varepsilon^2 \times f^2 / Tg$, where f is the gate utilization factor, the fraction of coincidences available for counting that actually fall within the finite gate.

In comparing two ^{3}He proportional counter-based systems for high rate applications, this functional dependence (combination of system parameters) may be used as a FOM. For systems that may be adequately described by a single exponential die-away, f^2 / Tg, may be optimized by picking $Tg \sim 1.26\tau$, where τ is the $1/e$ time constant of the detector. Hence, in place of the full expression e^2/τ is often used as a crude FOM to compare thermal-neutron detector-based alternatives to ^{3}He proportional counters. However, it is important to remember that in making this step we are implicitly assuming that the two systems will be notionally similar in performance (i.e., efficiency and accidental rate), will operate with the same value of predelay, and that the long lived tails present in the signal-triggered die-away profile have similar contributions. In detail, these factors are never perfectly matched and so we advocate here using the more complete FOM expression, $\varepsilon^2 f^2/Tg$ for high rate applications, with the gate utilization factor evaluated with practical predelay and gate width values. This FOM definition, however, does not reflect relative performance in the extreme where accidentals are of little consequence (that is for low level counting applications) where acquiring the most signal is the most important factor and longer gate widths are favored. In this regime $\varepsilon^2 f$ is a better FOM, which becomes $\sim\varepsilon^2$ in the limiting case, since when a gate width of about twice the die-away time is used, little further improvement is possible for most systems.

In practice, it would be ideal to compare performance estimates under realistic conditions of interest, that is on a scenario or intended use basis and this approach should be considered whenever applicable. For initial performance evaluation, when realistic samples might not be available and ^{252}Cf sources are used as surrogates, a well-defined performance characterization (FOM or such) should be developed to provide firm evaluation basis.

The situation for fast neutron detectors is quite different in character because the duration of the coincidence gate is only of the order of a few 10s of ns, rather than a few 10s of

μs, and the detection efficiency is generally far more modest. Organic scintillator-based systems are usually, therefore, operating in the low accidental rate regime (for the same item) and $\varepsilon^2 f^2/Tg$ is therefore not a fair FOM to use when comparing the coincidence counting precision of fast-neutron organic scintillators based systems against thermal-neutron detector based systems. If we assume the number of scintillator cells is large (so we can ignore the fact that a given cell can respond to only a single event per fission chain) a better comparison requires that we consider also the emission rate of the item and the degree of correlation in the emission rate. In other words, it is our strong recommendation, based on the discussions during the workshop, that, when inter-comparing system performance and when applicable, a scenario based approach is taken and that results across a range of realistic objects are compared, because no simple FOM can capture the richness of the underlying physics.

There was also considerable discussion during the meeting on how test results on modules, the basic sub-systems used to build larger detector arrays, should be best presented. The definition of intrinsic efficiency, for example, can be misleading when the area of the test module is much larger than that of the internal sensors, which may often be the case when just a few sensors with a non-optimal arrangement are being tested. In general, it would be ideal to project to a system level performance that includes full reflection and anticipated configuration, however, it is difficult to achieve such advanced geometry during the initial system evaluation. Therefore, a proper definition of test module configuration should be a key part of the evaluation best practices.

Concept of Operation

In addition to counting precision performance metrics, other over-arching requirements remain important. All prototypes that are tested should also be assessed based on their ability to be put into operation. Examples of requirement parameters include: reliability, footprint, portability, measurement time, stability, gamma-resistance, maintenance, safety, tamper-ability, calibration frequency and difficulty, quality assurance/quality control, transparency of design, cost, ease of use, operational temperature range, acceptable humidity range, and real-world-use robustness. Attention to this long list of requirements would help to standardize best evaluation practices. Such organized information should be readily shared with vendors to accelerate the production of ^{3}He alternative options for deployment.

The question of short-, medium- and long-term stability was repeatedly discussed during the workshop. This has a bearing on the associated uncertainty contribution that must be propagated. Based on discussions, one of the largest concerns is temperature variability. Plant temperatures have a wide range, and radioactive samples themselves can be warm and cause heating. High-pressure (six to ten atmosphere partial pressure in ^{3}He) one-inch (25.4 mm) diameter ^{3}He-filled cylindrical proportional counters are essentially black to the thermal neutrons they sample in moderated assemblies. This lessens any inherent temperature dependence. Together with electronic, dimensional, and other effects, ^{3}He-based systems have demonstrated relative temperature coefficients of 0.03 percent per °C (about twice this for "Reals" coincidence counting). This level of stability is essential when ambient temperature is not under control and can swing downwards as much as 20°C and upwards as much as 40°C for field applications.

An interesting feature of boron-coated detectors is that because the coatings are neutronically "thin," they respond more closely to the interaction cross-section. At low energies this varies approximately as the reciprocal of the neutron populations average speed which translates into inverse root (absolute) temperature dependence. This equates to about 0.2 percent per °C at room temperature coming from the reaction kinematics alone. The stability of full systems is yet to be proven. Scintillator and PMT based systems are expected to be far more sensitive to temperature and schemes to control gain are possible in principle but have also yet to be proven at the engineering level. Temperature testing techniques must be improved over the methods that rely on the use of simple heat lamps and thermometers designed to show if there will be a problem in order to provide a basis for an acceptance standard.

Not mentioned above are the safeguards performance requirements for data acquisition, data processing, signal processing, maintenance diagnostics, and state-of-health reporting. A thorough and systematic approach must be taken to address these also when identifying ^{3}He alternatives for international safeguards applications. In some cases the data acquisition systems are, by necessity, going to be more elaborate and with this comes an opportunity to improve on the status quo.

When it comes to putting ^{3}He alternative technologies to use, there are basic topics that are yet to be fully explored at a fundamental level as well as some practical matters that remain to be fully resolved. (These could provide a basis for future international collaborative discussions.)

Performance affecting processes:

- high-voltage characteristics
- charge-signal profiles
- double pulsing and other non-ideal behaviors
- pulse pile-up detection and minimization
- scintillation light propagation
- UV-processes in Townsend gas-discharges
- item matrix effects, especially for organic scintillators
- gamma-ray influence and neutron/gamma-ray misclassification
- PSD performance (charge integration to avoid photon shot noise, throughput, etc.)
- base-line transients (fast recovery and throughput, bipolar shaping, front-end electronics matched to detector and application)

Modeling needs:

- complete modeling of the neutron counting performance for systems encompassing novel detector concepts such as $^{6}LiF/^{10}B_2O_3$ ceramics

Evaluation best practices:

- development of a general purpose performance metric (FOM should be supplemented by relative standard deviation plots versus scenario using base-line systems of comparable level of development)
- development of application specific performance requirements (use-cases)
- standardization and harmonization (e.g., define appropriate single-module configuration, how to set all the thresholds/gains of a system's charge amplifier using a charge pulser, how to properly select operating parameters to maximize performance, while taking into account neutron/gamma discrimination)

Implementation related considerations:

- required modifications of existing electronics infrastructure
- status of associated software to support the emerging hardware
- re-training associated with the new technology and required operator experience

Other potential near-term technologies:

- evidence to support the view that glass-fiber detectors are not feasible for this domain
- revisit objections to BF_3-gas proportional counters (widely used excellent alternative to ^{3}He)
- manufacturing for reliability, maintainability, and total cost of ownership over the life-cycle

High-Voltage Plateau

At the workshop it was noted that unlike the case for gas (^{3}He or BF_3) filled proportional counters, nearly all alternative technologies to replace neutron well detectors do not have a clean physical (energy) discrimination between neutrons and gamma rays. For instance, the charged particle pulse height distribution (PHD) from the $^{10}B(n,\alpha)^{7}Li$ reaction, where charged particles enter the counter gas from a boron coating, extends all the way to zero and therefore overlaps with the electron energy deposition spectrum created by gamma-ray interactions in the walls and gas. Figure 4 shows the charged particle pulse height distribution (PHDs) for a ^{3}He and a ^{10}B proportional counter. The separation between gamma-rays and neutrons is therefore achieved by setting an energy threshold. The higher the threshold the greater the discrimination factor but the price is that a larger fraction of neutron detections are also rejected. Once an instrument has been built, the threshold is usually set experimentally (by observing gamma-ray and mixed n/γ sources) by setting a threshold so as to achieve an acceptable compromise for a given task. For reasons of stability one also would like to operate in a region of the high-voltage characteristic where small changes in the high-voltage setting have minimal impact in the efficiency.

In the safeguards community, familiarity of the behavior and use of ^{3}He proportional counters is deeply engrained. In this case, the high-voltage characteristic is usually referred to as the high-voltage plateau. The choice of high-voltage setting is commonly just above the "knee" and on the relatively flat portion of the curve. This is a place where essentially the entire neutron and none of the gamma-ray events are recorded. The reason why the plateau exists is because of the finite energy gap between gamma ray and neutron events in the gas. Figure 5 shows high-voltage plateaus for a ^{3}He, ^{10}B, and a hybrid $^{3}He/^{10}B$ proportional counter.

Based on this experience it might be tempting for a person setting up a boron-lined proportional counter to mistakenly look at the integral high-voltage characteristic in a similar way. That

Figure 4. PHD comparison of a ^{10}B proportional counter to a ^{3}He proportional counter. The ^{3}He spectrum demonstrates a neutron capture by the gas that releases energy from 190 KeV to 764 KeV in the full energy peak. The ^{10}B spectrum shows the alpha particle edge at 1.47 MeV, the lithium particle edge at 840 KeV and the continuum of the wall effect. All information below ~50 KeV is electronically eliminated. The dashed line shows a typical placement of a lower level discriminator.[24]

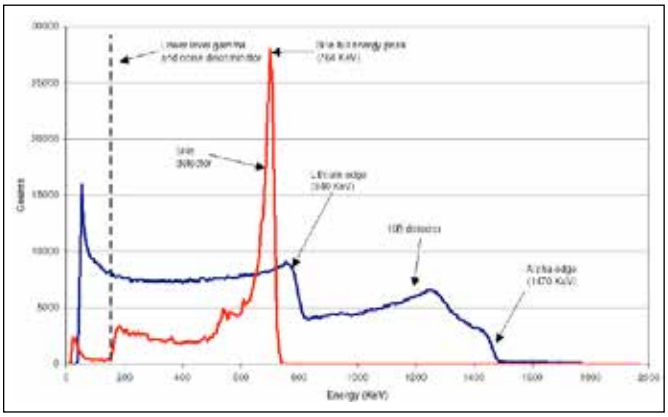
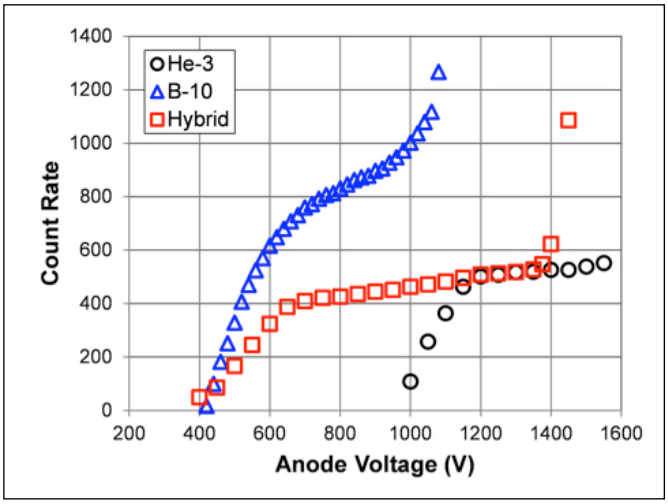


Figure 5. High voltage plateau plots for a ^{3}He, ^{10}B, and hybrid $^{3}He/^{10}B$ neutron detector. All plateaus were run with electronics that have a 0.5 or 1 microsecond time constant. The low energy cutoff of the curves is due to electronics and physically would go down to zero volts.[24]



is, to interpret the inflection region of the slope (between the sharp rise in counting rate as a function of high voltage and the ultimate "breakaway" when the detector is increasingly sensitive to gamma-ray) as a plateau. However, there is in fact no analogous region because there is no physical energy separation between neutron and gamma-ray induced events (shown in Figure 4). The purpose of this discussion is simply to note that when adopting any new, different, or unfamiliar technology, a degree of re-training and re-education will be needed to ensure that past practices or habits are not carried over without appropriate scrutiny.

To illustrate how the "appearance" of a plateau (albeit with a steeper slope than would be expected from a ^{3}He-filled proportional counter) can form for boron-lined detectors, consider the simple case of a unit normalized rectangular PHD with the upper edge just at the pulse-height threshold. If we double the gain we would get half the counts above the threshold. If we triple the original gain we get two-thirds of the counts, etc. Algebraically the counting rate, *C*, varies with the gain, *G*, as:

$$C = \frac{(G-1)}{G} = 1 - \frac{1}{G}$$

The relative gain of a proportional counter varies roughly exponentially for gains (typically) above about five up to about 1,000 or so, and proportional counters are usually run with gains in the range of 10s to 100. Thus, we may approximate the gain by an exponential, as shown below, and obtain the counting rate in the form of a saturating exponential.

$$G = e^{(V-Vr)/a}$$

$$C = 1 - e^{-(V-Vr)/a} - \textit{a saturating exponential}$$

When we replace the rectangular PHD with a more realistic shape, increasing at lower energies, we can see how there is a tradeoff between the rising intensity and the gain variation that is slow to saturate. Thus, the high-voltage characteristic of a boron-lined proportional counter will have the appearance of a plateau, with a steep slope (unlike a ^{3}He proportional counter plateau which should be reasonably flat, 1 to 3 percent per 100 V depending on the fill pressure and amplifier shaping). This can be formalized more quantitatively but the discussion is outside of our present scope as it involves the realities of detector fabrication, gas gain with voltage, neutron flux depression across the liner, and charged particle transport into the gas to achieve the highest level of fidelity.

At the workshop, an add-on to the widely used Monte Carlo radiation transport code MCNPX was presented, which addresses this particular issue. The new option launches the reaction charged particles on an event-by-event basis and tallies the energy deposition in the gas. As we do not always know the layer thickness or uniformity of the surrounding materials, this approach may be too elaborate for certain scoping studies. An alternate approach, which would speed up the calculation, is to add a simple step to MCNPX after the $^{10}B(n,\alpha)^{7}Li$ interaction. The neutron (n,α) event would be counted in accordance with a user entered probability. This probability would be generated by

a formula based calculation and would produce the coincidence tally much faster and in most practical applications would be 'good enough' for rapid design. This is just one example of the way in which the study of ^{3}He alternatives has required existing tools and approaches to be adapted. Some manufacturing processes involve steps which involve commercially protected information, but details of coating thicknesses, materials, and uniformity were understood to be important for a first principles understanding and prediction of detector performance.

Consensus Observations from the Working Group

Detector Evaluation and Comparison Methodologies

Safeguards performance requirements and needs should be developed and distributed to all of the entities striving to develop novel technologies. Laboratories with well-known plutonium samples for passive counting and well-known uranium samples for active counting could be established to provide testing facilities for new systems. This task will not be trivial considering it is likely that subgroups of technologies for that vast amount of different applications will need to be developed. The requirements need to be different for each application. It is critical to compare technologies against one another, because, whatever replacement technology does eventually make it into the field will become the norm and will not be easily changed by improved alternatives. The definition of small and conservative international safeguards use cases will aid in organizing competitive evaluation of ^{3}He alternative technologies.

Use Cases: Mapping the technology onto the application space

As we move from the scientific development of potential alternative neutron technology subsystems, we expect there to be a greater focus on the application space and problem solving. The constraints and measurement goals of realistic scenarios will provide the discerning optic needed to make good system design choices. State of the art technology is not required, provided that available technology is proficient, robust, and simple — either inherently or by engineering. Suitable case studies would include, for illustration, the following:

1. Small Pu/MOX < 10g
2. Medium Pu/MOX < 100g
3. Large Pu/MOX
4. Fresh Pu/MOX
5. Scrap Pu/MOX
6. Spent Fuel
7. Fresh MOX
8. Waste, typically weakly multiplying but otherwise highly variable
9. UF_6 Cylinders
10. Monitors for Continuity of Knowledge (COK)
11. Power Monitors
12. U Metal, Oxide
13. Research Reactor Fuel Fabrication
14. Research Reactor Spent Fuel
15. Holdup (accumulations and deposits in process equipment)

Conclusions and Future Outlook

Specific points regarding ^{3}He alternatives for safeguards were emphasized at the workshop. It was apparent that NDA with neutrons is an essential part of integrated safeguards. There is a need to create a realistic vision of future ^{3}He needs, by the IAEA and others. Additionally there is a need to create a plan that envisions different safeguards solutions, for instance under the State-Level Approach (SLA), that can be implemented with less ^{3}He and in some areas with less capable neutron detectors. If ^{3}He is prohibitive, then under a SLA, NDA precision and bias requirements could be relaxed if there is compensating data. The IAEA is one user of neutron detectors, and have demanding requirements but they are not the major consumer of ^{3}He because the plants often have additional needs and often purchase systems. There are a broader range of global needs of interest to this community that would also benefit from a coordinated planning review. These include meeting various treaty obligations, implementing Additional Protocol options, executing bilateral agreements and technical exchange engagements, global nuclear security, accident response (e.g., Chernobyl and Fukushima), and on-going field tests and demonstrations to maintain and grow capability. Viable neutron detection technologies are needed to satisfy all of these needs. At this point in time, many neutron detection methods exist, but few provide viable alternatives to ^{3}He for the highest performance needs (e.g., multiplicity counting) of safeguards and material control and accountability.

Instrumentation users and vendors must be engaged at this point in the progression of alternative ^{3}He technologies for safeguards. Technical meetings can work as a medium for presenting the information discussed at the workshop and eliciting further feedback. The field greatly benefits from the information exchange between development scientists and actual

users and maintainers of measurement systems. Venues may include the IAEA Safeguards Symposium or publication in the *Journal of Nuclear Material Management* or the *ESARDA Bulletin*. Many of the technologies discussed at the workshop were also reported at the ESARDA meeting held in Belgium earlier in 2013 and at the annual INMM meeting held in the U.S. subsequently. The proceedings of these international meetings can be recommended as a source for detailed technical information.

More near-term strategies can include ^{3}He recovery and reuse. When ^{3}He was plentiful and cheap it was used extravagantly; making do with less (e.g., less than ten atmospheres partial pressure in one-inch diameter tubes) is an important step to using annual allocations wisely. When a basic alternative ^{3}He neutron detector will work adequately, it would be helpful to implement it as soon as possible and preserve ^{3}He resources. Other safeguards concepts and approaches can also be used to ease ^{3}He needs. Revisiting traditional approaches may also provide a viable alternative—for instance the use of active methods, calorimetry, etc. Familiar methods should also be re-explored, for instance, to assess neutron multiplication in different ways, or to make greater use of quantitative gamma-ray-assay/verification instead of neutron counting.

The general purpose solutions of the ^{3}He era can be expected to give way to more sophisticated single purpose designs. Continued development on detector design (geometries, materials) and tailored front-end electronics is expected with a focus on increasing throughput and reducing dead-time. There is a significant step between establishing scientific feasibility on a small test piece and engineering a field instrument. The time and cost of making this transition should not be discounted. Future workshops must focus on mapping the technologies under development onto the application space. In addition, implementation questions, such as associated data acquisition electronics, analysis software and required operator experience should be considered. Particularly, items listed in the *Concept of Operation* section should provide a basis for future considerations and discussions. Demonstrations can then be organized to prove the abilities of ^{3}He alternatives against currently used technologies and identify appropriate paths forward.

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Outcomes of the Information Analysis Technologies, Techniques, and Methods for Safeguards, Nonproliferation, and Arms Control Verification Workshop

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Abstract

The Information Analysis Technologies, Techniques, and Methods for Safeguards, Nonproliferation, and Arms Control Verification Workshop was held May 12 – 14, 2014, in Portland, Oregon, USA. The objective of the workshop was to facilitate dialogue between policymakers, analysts, and technical experts on information analysis for safeguards, nonproliferation, and arms control verification. The workshop had more than fifty participants from across the globe, representing research and development organizations, universities, commercial software developers and analytic service providers, government and non-governmental organizations, and international safeguards bodies.

The workshop featured high-level opening and closing plenary speakers, seven technical sessions in which researchers presented their work, and a software demonstration session focused on nonproliferation and arms control applications. In addition, the agenda was highlighted by a special address from one of the workshop sponsors on societal verification, and a moderated discussion focused on the challenges for this field. Final recommendations from the workshop include the following:

1. Existing information analysis and information technology capabilities developed and used outside of the nuclear verification domain that could be applied to support the domain should be identified.
2. The international verification community should engage the IAEA in discussions regarding information analysis for safeguards, and readjust research and development priorities to be more in line with IAEA needs.
3. Priority should be given to data integration for IAEA safeguards.

Introduction

The international nonproliferation community is entering the information age. The past two decades' leaps forward in computing technology, combined with the international attention to verification of international safeguards, nonproliferation, and arms control agreements, has resulted in greater prominence of information analysis. The field is evolving rapidly, and because of the interdisciplinary nature of the field, dialogue between experts who develop information analysis technologies, techniques, and methods, and those who are using those capabilities for nonproliferation verification, is critical.

To facilitate an opportunity for analysts, policymakers, and researchers to exchange ideas, discuss current work, and understand lessons learned, Pacific Northwest National Laboratory (PNNL) organized the ***Information Analysis Technologies, Techniques and Methods for Safeguards, Nonproliferation and Arms Control Verification Workshop*** from May 12 – 14, 2014, in Portland, Oregon, USA. The workshop was hosted by the Pacific Northwest Chapter of the Institute of Nuclear Materials Management (INMM), with the support of the INMM Divisions of Safeguards and Nonproliferation and Arms Control.

The workshop included more than fifty participants coming from across the globe, representing twenty-two organizations across research and development organizations, universities, commercial software developers and analytic service providers, government and non-governmental organizations, and international safeguards bodies. A copy of the workshop proceedings, containing full text papers, is available on the workshop website: http://www.inmm.org/Information_Analysis_Technologies.htm.

Workshop Synopsis

The workshop featured seven technical sessions, covering a broad spectrum of information analysis topics and their relevance to safeguards, nonproliferation, and arms control verification. Paper topics included:

- societal verification
- open source information analysis
- big data challenges
- machine learning
- cyber security issues
- protection of sensitive information during joint activities
- data visualization
- graph analytics
- the use of testbeds
- novel application of statistical methods
- modeling and simulation techniques
- data integration techniques and software

In addition, the workshop included an opening plenary speaker from the International Atomic Energy Agency (IAEA), a software demonstration session, a special address from one of the workshop sponsors on societal verification, a moderated discussion with workshop participants on lessons learned, and an inter-agency closing plenary panel discussion.

Thomas Lorenz made the opening plenary address on behalf of Jacques Baute (director, Division of Information Management, Department of Safeguards, IAEA) on Information Analysis Needs of the IAEA. The opening plenary described the various data sources and information cycle used by the IAEA Department of Safeguards to draw safeguards conclusions. It also described some challenges faced by the department, such as balancing information security and collaborative workflows, expanding information sources, and improving the effectiveness and efficiency of analytical processes.

The workshop also featured software demonstrations of eight commercial and laboratory-developed software packages and their role in supporting information analysis for safeguards, nonproliferation, and arms control. Software demonstrations included the following vendors and packages:

- Tableau and GNDA Risk Tree Visualization
- T. Rex
- Chain of Custody Data Management System
- Velocity
- Iconics Genesis64
- Genetec Security Center
- Adaptive Data/Vendor X
- Tableau Software

The software demonstrators were provided a common dataset featuring the following four data types for use, if desired:

1. **Scientific and technical publication data:** 3,500 entries of scientific and technical publications from the IAEA's International Nuclear Information System (INIS) scientific literature database, including publication details and paper abstracts. The data spanned 2009–2014, and focused on the topics nuclear security and nuclear safety.
2. **Seismic data:** 8,500 entries of earthquake data from the United States Geological Survey, spanning March 18, 2014, through April 17, 2014. Details included time, location, and magnitude of earthquakes recorded.
3. **Social media data:** 200,000 rows of social media data from data and analytical services vendor Recorded Future, spanning February 1, 2014, through February 14, 2014.
4. **Images:** 5,062 of images of buildings around Oxford, England, collected from Flickr via the University of Oxford's Visual Geometry Group's Oxford Buildings Dataset.

In addition, a special address was delivered by the Nuclear Threat Initiative's (NTI's) Kelsey Hartigan on the outcomes of NTI's Societal Verification Working Group. In her address, Hartigan outlined the research done by the Working Group, specifically calling out legal and ethical issues related to societal verification for nonproliferation and arms control that were highlighted in a meeting the week before the Information Analysis Workshop. She noted that though some research has been completed on the potential to use societal verification for nonproliferation and arms control, additional case studies were needed to determine the full extent to which such an approach could be implemented, and how verification regimes might legally incorporate societal data.

Before the closing plenary session, Technical Committee Chair Seán Kreyling moderated a discussion on what we learned in the workshop, remaining questions, and ideas for how to move forward as a verification community. The discussion focused on six main themes:

1. The availability and trustworthiness of datasets potentially relevant for verification
2. Information security challenges, especially those due to the multilateral nature of many verification regimes
3. The difficulty of implementing information analysis capa-

bilities for verification organizations prior to those capabilities becoming obsolete

4. The need for increased communication and collaboration between policymakers, developers of analysis tools and techniques, and analysts and other users
5. Understanding the potential applications of societal verification, and the importance of cultural consideration (such as use of cell phones and the internet)
6. Fostering a cadre of information analysis technique and tool developers who also understand the nonproliferation and safeguards.

The closing plenary panel focused on prospects of future research and development needs in the area of information analysis for safeguards, nonproliferation, and arms control verification. Panel members were:

- Joseph Kielman (Science Advisor, Cyber Security Division, Science and Technology Directorate, U.S. Department of Homeland Security)
- Peter Sprunger (Physical Scientist, Bureau of International Security and Nonproliferation, Office of Multilateral Nuclear and Security Affairs, U.S. Department of State)
- Daniel Wurmser (Office of Verification and Transparency Technologies, U.S. Department of State).

The panel members underscored the importance of continued research in data integration. They also highlighted challenges to the international verification community, including staff turnover, high politicization of organizations such as the IAEA, and the slow rate of organizational adoption of new technologies that could support nonproliferation objectives. They noted that technology is the driver for where we go next and called for an extension of the international community's outlook from "What can we do now?" to "What might be possible ten to thirty years from now?" The panel also discussed the need for data integration and other information analysis issues to be better understood by policymakers.

Conclusions and Recommendations

The opening and closing plenary sessions of the workshop served as a call to action for participants as the speakers highlighted the current challenges and future research directions of their respective organizations. The opening plenary speaker was Thomas Lorenz, who spoke on behalf of Jacques Baute, director of the IAEA Department of Safeguards Division of Information Management. Closing plenary speakers included Joseph Kielman of the U.S. Department of Homeland Security's Science and Technology Directorate, Peter Sprunger of the U.S. Department of State's Office of Multilateral Nuclear and Security Affairs, and Daniel Wurmser of the U.S. Department of State's Office of Verification and Transparency Technologies.

Lorenz opened the workshop by describing the current state of information analysis in the IAEA Department of Safeguards. He raised a series of questions to workshop participants, which highlighted areas of pressing need, ranging the full spectrum of information analysis from data collection and storage, to analytical tools and processes. The identified needs were:

- Where should the IAEA Department of Safeguards be looking for potentially relevant information?
- How can the Agency best store and protect its safeguards information to ensure the highest quality, maintain internal consistency, and provide access to institutional knowledge?
- How can the Agency optimize protection of information and proper dissemination of information to support effective analysis?
- How can the IAEA find and implement better expertise, processes, and tools in order to enhance safeguards findings and conclusions?
- How can the IAEA constantly improve the effectiveness and efficiency of the consistency analysis of all safeguards-relevant information?

These questions framed the presentations and discussion throughout the remainder of the workshop. In the closing plenary, panel members touched on many of the same issues while describing their agencies' research agendas.

In the closing plenary presentations, the panel members incorporated ideas and themes they heard throughout the workshop, and formed cohesion between the expected direction of research at their respective organizations with the needs and lessons presented by the researchers, analysts, and policymakers during the technical sessions.

Kielman focused on the needs for visual and predictive analytics. Kielman stressed that data analysis needs are not unique to the safeguards, nonproliferation, and arms control community, and are well documented in the field of information technology. While Kielman's focused more on national security than nonproliferation and arms control, the research needs he highlighted were highly relevant for both communi-

ties: data provenance (especially dealing with open source data and data coming from inspected parties), situational awareness, the need for rapid indicators of anomalous behavior, and socio-behavioral considerations.

Sprunger focused on IAEA information technology needs, based on his role on the Subgroup on Safeguards Technical Support (SSTS), which provides guidance on the U.S. Support Program to IAEA Safeguards (USSP). Sprunger identified data integration as the primary information technology challenge for IAEA safeguards. According to Sprunger, the IAEA has sufficient data analysis software, but the integration of the analyses in those tools will provide the most useful gains. However, he noted that data integration at the IAEA is not likely for a long time due to the difficulty in implementing software at large international organizations such as the IAEA.

Wurmser described the U.S. State Department's Verification Technology R&D Needs Document. He called out nuclear arms control and transparency as two challenging issues, and indicated that the U.S. State Department's information technology interest lies in innovative, "paradigm shifting" technical solutions that can support more traditional arms control activities. His presentation specifically highlighted technologies that could support intrusive inspection regimes, ubiquitous sensing, mobile devices, and the internet of things, and analytic support software (information discovery, data mining and advanced search support, visual analytics, and temporal analysis). He also mentioned the importance of ethical and legal issues, especially related to privacy issues and risk to users of advanced concepts such as societal verification.

In addition to their agency-specific research needs, the plenary speakers also emphasized challenges to the verification community as a whole, including staff turnover and the slow rate of organizational adoption of new technologies that could support nonproliferation objectives. The plenary speakers described technology as the driver for future verification regimes, and encouraged the verification community to extend their priorities from "What can we do now?" to "What might be possible in ten to thirty years from now?" The panel also discussed the need for data integration and other information analysis issues to be better understood by policymakers.

Key observations from the opening and closing plenary speakers' comments are:

1. Information collection and analysis are at the heart of modern nuclear verification. However, safeguards, nonproliferation, and arms control verification agencies are still in need of the appropriate methodologies, techniques, tools, skills, and methods to effectively leverage information collection and analysis.
2. Current inspection regimes and national technical means alone are insufficient for treaty verification. In addition to having more intrusive inspection regimes, new tools to support future verification efforts are required for:
   - Dispersed and ubiquitous sensing;
   - Enhanced human observation enabled by the information revolution; and
   - Improved data analysis and exploitation.
3. Transparency and confidence building measures may increasingly supplement traditional arms control and nonproliferation agreements—information analysis can play a significant role here. To do this, transparency regimes need to be carefully designed with the following in mind:
   - Protect sensitive or proprietary information;
   - Technical approaches that are robust and resistant to manipulation; and
   - Institutions designed to engender high degree of professionalism and impartiality.
4. Data integration is a major information technology challenge for IAEA safeguards. Focusing on furthering software that will allow for the integration of different types of data is needed, whether it be for in-field activities to allow the inspector to be able to use one platform to collect and assemble all of the data, or at IAEA headquarters where analysts and inspectors are analyzing large and diverse data sources and condensing their findings into a single concise report.

Though many of the information analysis challenges described by Baute were echoed as research needs for the agencies represented in the closing plenary, there were some differences. For example, Baute's first question regarded additional potential data sources for IAEA safeguards analysis, while Sprunger emphasized that there is sufficient data and emphasis should be placed on mechanisms to integrate existing data. This raises the question: Is there a divide between the IAEA's (or other verification bodies') information analysis needs and the international verification community's priorities for information analysis technologies to support the Agency? Or rather, does this difference point to a communication breakdown? The authors offer the following recommendations:

Recommendation 1: Conduct a survey of existing information analysis and information technology that could be used for safeguards, nonproliferation, and arms control analysis, with consideration given to each verification body's analytical needs and technical requirements. For capabilities that closely align with a verification body's needs and requirements, additional research should be conducted to determine what it would take to transfer that capability to the verification agency, rather than the development of costly "boutique" software solutions.

Recommendation 2: Recent activities, such as participation in the Information Analysis Workshop, indicate an opening of IAEA dialogue regarding information analysis technology. The international verification community should take advantage of this opening and engage the IAEA in discussions regarding information analysis for safeguards, and readjust research and development priorities to be more in line with IAEA needs.

Recommendation 3: Priority should be given to data integration for IAEA safeguards. Data integration remains a critical information and communication technology need for the IAEA, but because the Agency has been hesitant to discuss information and communication technology issues with external partners, funding for these projects has diminished.

These themes were also reflected in a closing discussion with workshop participants, who identified six main areas of greatest opportunity for information analysis in support of safeguards, nonproliferation, and arms control verification:

1. *Data.* Given that transparency and confidence-building measures may increasingly supplement traditional arms control and nonproliferation agreements, the international verification community requires a better understanding of what data is available, how trustworthy or accurate it may be, how the data could potentially support their respective verification missions, and the legal mechanisms required to incorporate that data into its analysis.
2. *Information security.* Data security issues abound, especially regarding how to manage the use of open source information when those sources sometimes contain more sensitive details than those held as confidential or otherwise classified by an organization. The information security challenge is different between safeguards and arms control, in part because there is more trust between parties in arms control. Gaining that level of trust by the IAEA with every member state would be exceedingly difficult, and the Agency doesn't have the benefit of some of the joint technology development opportunities that create trust in arms control.
3. *Timing.* There is a critical need to continue information and communication technologies research. The information technology development and acquisition challenge exemplified by rapidly evolving information analysis capabilities and data access can result in technology for verification organizations becoming obsolete before its completed implementation. Though a time lag in deployment of new technologies (as well as international politics) may have slowed research into this area, there is a strong need for innovative solutions to support information analysis for nonproliferation, safeguards, and arms control verification. Incremental deployment of small or modular applications might offer a benefit and provide more flexibility.
4. *Communication and collaboration.* Policymakers, developers of analysis techniques and tools, and analysts using those techniques and tools tend to work on their challenges independently. However, these groups need to be communicating their needs, potential solutions, and special considerations for solutions to be fruitful. The conversations between policymakers and information analysis developers and users do not happen as often as needed because of the organizational barriers between the groups, limited communication channels, and diverse professional vernaculars (with different priorities).
5. *Societal verification.* Signature identification techniques are needed in societal verification along with an understanding of how to use the lessons and technologies from the marketing industry to benefit international verification efforts. For the verification community to do societal verification, help is needed from the outside to manage that volume of data (and potentially hardware). The discussion emphasized cultural factors to consider in societal signatures (such as use of cell phones and the internet).
6. *Human resources.* Skilled developers of information analysis techniques and tools are getting lured away from the international verification community by the information technology industry and companies like Amazon and Facebook. The international verification community needs to develop a cadre of technique and tool developers who are highly skilled and who also understand the workflow and

requirements involving nonproliferation and safeguards. The international verification community is currently not competitive to recruit and retain such developers; rather the community relies on the developers' own interests to keep them in the field.

It is apparent from the closing plenary remarks and the participant discussion that the workshop achieved its objective: to bring together experts from the policy and technology communities to exchange ideas, current work, and lessons learned. While the purpose of the workshop was not to identify a path forward for the information analysis or verification communities, the identification of a set of challenges and research needs may provide an informal understanding among policymakers, analysts, and technical experts on information analysis for safeguards, nonproliferation, and arms control verification. Despite its apparent success, the Information Analysis Technologies, Techniques and Methods for Safeguards, Nonproliferation and Arms Control Verification Workshop alone was not sufficient to solve the information analysis challenges facing the verification community. Funding for future research should be prioritized based on the verification community's needs, and should be done through a coordinated approach between funding agencies.

Zoe Gastelum is a nonproliferation research scientist at Pacific Northwest National Laboratory. Gastelum has experience in developing computational models and analytical approaches to support nuclear nonproliferation analysts. Her work has included developing a mobile information platform for enhanced situational awareness for safeguards inspectors, modeling terrorist groups' intent to use weapons of mass destruction, analyzing the feasibility of social media and other open source data to support nonproliferation and arms control treaty verification, and developing strategies for information and communication technology deployment to international safeguards organizations. Gastelum first joined PNNL in 2008. From October 2009 to October 2011, she served as an Open Source Information Analyst in the International Atomic Energy Agency's Department of Safeguards. She has a BA in political science from the University of New Mexico, an MA in international security from the University of Texas–Austin, and a graduate certificate in international nuclear law from the University of Montpellier-1.

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Global Views of Spent Fuel Storage and Disposal

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In mid-January 2015 about 100 nuclear industry professionals met in Washington, DC, USA, for the 30th Annual Spent Fuel Seminar, sponsored by the Institute of Nuclear Materials Management (INMM), in partnership with the U.S. Nuclear Infrastructure Council (NIC). This year the conference had a heavy international flavor, including updates on the successful disposal program in Sweden, progress being made toward a consolidated interim storage facility in Spain, perspectives of the European Union on spent fuel management, updates on spent fuel storage in Germany and Japan, and what the U.S. Department of Energy (DOE) is doing to lay the groundwork for its spent fuel strategy that was announced two years ago in January 2013. Regulatory perspectives, legislative predictions in the U.S., advances in storage research and development, transportation, and the growing decommissioning market were also covered as part of the packed two-and-a-half-day agenda.

International Overview

Nigel Mote, executive director of the U.S. Nuclear Waste Technical Review Board (NWTRB) offered a global overview of spent fuel management. Mote pointed out the important fact that the disposal of spent fuel and/or high-level radioactive waste (HLW) in a deep underground repository is an internationally accepted concept. Many countries with nuclear power programs have plans for repository disposal, even though several countries with repository programs have had "resets" in their programs. Still, despite significant advances over the last few years, most notably in Sweden, Finland, and France, no spent fuel or HLW has been disposed of to date, and no country has a repository for this material licensed, much less in operation. As a result, long-term storage of this material, mainly at reactor sites, continues to be the reality that electricity generators face. Some countries do reprocess spent fuel, but a disposal facility is still needed for the waste resulting from that process.

Mote presented a brief summary of the spent fuel/HLW policies of a number of major nuclear countries, including any progress on siting or licensing a disposal facility. Beginning with the countries most advanced in the process, his overview is summarized below:

- **Sweden** and **Finland** have several elements of their programs in common: both countries have repository license applications under review by their respective regulators; both repositories will be located in granite; both will use the KBS-3 concept with small capacity spent fuel canisters; and both locations are sited near nuclear facilities where the population is familiar and comfortable with nuclear power and the industrial activity surrounding it.
- **France** concluded public debate on its proposed repository in 2014 and a license application is scheduled to be submitted to the French Parliament in 2017. Since France reprocesses all of its spent fuel, the repository will contain vitrified HLW containers, although PWR mixed-oxide (MOX) spent fuel may also be emplaced in it.
- **Canada**, through its Adaptive Phased Management siting approach, has had twenty-two communities express an interest in hosting a repository, nine of which have now been eliminated from consideration for technical reasons.
- **Germany** passed the Repository Siting Act in 2013 and established a Commission to develop a siting process. In contrast to the U.S., when Germany reset its program all previous potential sites remained open for consideration.
- **Belgium** has not officially adopted a spent fuel disposition strategy, but is completing performance assessments on data from an underground research laboratory in Boom clay near the Dessel/Mol site.
- **China** is planning an underground repository laboratory in granite near Beishan, and also is considering an underground lab at a clay site.
- **Japan** has moved away from the consent-based process and will rely on the central government to identify potential sites.
- South Korea is planning to have a centralized interim storage facility operational by 2024.
- **Russia** has centralized wet and dry spent fuel storage facilities at Zheleznogorsk, and is planning an underground laboratory at an intended repository site, for which a decision is expected by 2025 with operations starting in 2035.

- **Spain's** long-term strategy includes spent fuel disposition in a repository at Villar de Cañas, which was selected using a consent-based siting process. Spain is also building a centralized interim storage facility.
- **Switzerland** has had an away-from-reactor centralized storage facility operating since 2000; underground laboratories are operating in granite and in clay.
- The **United Kingdom** experienced a reset when one level of government rejected the plan to build a repository at a proposed site, so a new approach was outlined in a 2014 government white paper.

The United States also has reset its waste management program, but needs congressional authorization to move forward with site-specific siting efforts.

Some common elements in the three most successful countries thus far are that all have used a consent-based site selection process; all have single-purpose implementers; all have long-term, multi-year assured budgets; all have stable political support; all have high staff retention rates; all have focused on the demonstrating long-term safety and establishing and maintaining public acceptance. None of those common elements guarantees success, however, and Mote noted that both Sweden and France have experienced initial resets.

Other speakers elaborated on the specifics of their countries' or regions' plans and progress. The **European Union** (EU) has twenty-eight member states, each of which can decide whether or not it wants to include nuclear power in its energy mix. Sixteen countries have nuclear power plants, which account for more than one quarter of the electricity generation, according to Christos Koutsoyannopoulos of the European Commission. The nuclear power plants generate about 3,200 metric tons of heavy metal (MTHM) of spent fuel each year.

The European Commission is the executive arm of the EU. Its Council Directive 2011/70/*EURATOM* of July 19, 2011, established a community framework for the responsible and safe management of spent fuel and radioactive waste. Member states are required to present national programs that indicate when, where, and how they will construct and manage final repositories and how they will guarantee the highest safety standards. The safety standards become legally binding and enforceable in the EU. Member states have to submit the first report on the implementation of their national programs this year. The directive promotes transparency and public information, and includes strict conditions on exports of radioactive waste outside the EU.

Sweden's program is often touted as the model of a consent-based siting process. Anders Sjöland from SKB said that Sweden's twelve operating reactor units, located at three sites, provide about 45 percent of Sweden's electricity. Sweden currently has a central interim storage facility for spent nuclear fuel, known as Clab, and is the furthest along in developing a final repository for spent fuel and HLW of any country in the world.

The license application for a spent fuel repository at Forsmark and separate application for an encapsulation plant in Oskarshamn was submitted to the national regulator in March 2011 — the day the earthquake and tsunami destroyed the Fukushima Daiichi Nuclear Plant in Japan. The review takes place under both the Environmental Code and the Nuclear Act. The Environmental Court has had many questions; 350 questions were sent to the court by stakeholders. SKB submitted a 600-page response document in April 2013, and provided additional responses in September 2014. SKB is waiting for the Court to declare the application is complete, which is a very important step. The Radiation and Nuclear Safety Authority (STUK) also must declare the application is complete (on February 11, 2015, STUK notified Finland's Ministry of Employment and the Economy that its review is complete and that the "spent nuclear fuel encapsulation plant and final disposal facility designed by Posiva can be built to be safe"). SKB needs five approvals to start construction — the STUK, the Environmental Court, the communities of Östhammar and Oskarshamn, and the final decision made by the Swedish government, basically by the prime minister. A decision could be made by late 2015.

The repository will have a capacity of 6,000 spent fuel canisters, which correspond to 12,000 metric tons of spent fuel. The facility will operate for sixty years, followed by decommissioning and closure. SKB hopes to begin construction in early 2019.

Public opinion on the facility has been very stable and supportive. Dr. Sjöland stated that about 85 percent of the population actively wants a repository in their community In Sweden, the community that was not chosen to host the repository (Oskarshamn) will receive two-thirds of the government incentive, and the community that was chosen will get one-third. The main prize is the repository itself and all the economic benefits that will ensue as a result.

Sweden also has a centralized wet storage facility, Clab, which has a capacity of 8,000 metric tons of spent fuel. An application has been submitted to increase that capacity to

11,000 metric tons since the facility will reach its limit before the repository is in operation. The expansion will likely be increased by installing higher density storage racks and removing non-fuel items from the pool.

Germany is in the process of phasing out nuclear energy; after the Fukushima disaster, eight plants were ordered to shut down immediately, and the remaining nine will be shut down by the end of 2022. Dr. Holger Volzke, of Germany's Federal Institute for Materials Research and Testing (BAM), said that as of December 31, 2013, German nuclear power plants had accumulated about 53,600 spent fuel assemblies; by the end of 2022 when the last German reactor is permanently shut down, the reactors will have discharged about 60,000 spent fuel assemblies. Currently, about 1,000 dual-purpose storage casks of various types are in use for dry storage of spent fuel and HLW at sixteen storage sites, twelve of which are at-reactor facilities.

Germany's policy calls for interim storage of spent fuel followed by disposal in a deep geological repository. Germany has had a total reset of its repository siting procedure. The Repository Site Selection Act entered into force on July 27, 2013, which called for a final repository site selection to be done by 2031. The site selection procedure will be prepared by a thirty-three-member commission made up of eight scientists, eight representatives from social stakeholder organizations, eight members from the German parliament, eight ministers representing the states, and one chair. The commission is scheduled to present proposals and geological selection/exclusion criteria by the end of 2015. All areas of the country can be considered. Because of the delay in repository operations, extended storage of spent fuel in dry casks of up to eighty years or even more is likely.

Masumi Wataru of Japan's Central Research Institute of Electric Power Industry (CRIEPI) said that according to Japan's new energy plan, nuclear power is still considered an important baseload power source, and the government will lead the efforts for the final disposal of HLW. Construction of a new dry storage facility at the Mutsu site was completed on December 22, 2014. Built by the Recyclable Fuel Storage Company (RFS) the capacity will be about 3,000 tU, or about 288 dry storage casks. Final storage capacity is planned to be 5,000 tU. RFS was established in November 2005 by Tokyo Electric Power Company (80 percent) and Japan Atomic Power Company (20 percent).

David Garrido, Design and Licensing Manager at Spain's Equipos Nucleares, S.A. (ENSA) said that current spent fuel inventory in Spain is about 4,600 tU, most of which is in spent fuel pools. Three independent spent fuel storage installations (ISFSIs) are in operation – one at the Trillo Nuclear Power Plant, one at the José Cabrera plant, and one at the Ascó plant. An ISFSI at the Santa Maria de Garoña plant is planned for 2016. The Almaraz and Cofrentes plants use wet storage only, and will ship spent fuel to the ATC (Almacén Temporal Centralizado) interim storage facility that is under construction at Villar de Cañas. The first shipment is scheduled to come from the Almaraz 1 pool in 2018. Since the vault storage will not be ready by that time, an ISFSI is being built at the ATC to store the casks until the vault storage is ready to operate. The ISFSI will be ready to receive the loaded casks in 2018.

The total estimated amount of spent fuel generated in Spain, assuming each reactor operates for forty years, will be 6,700 tU, or about 20,000 fuel elements.

Spanish regulations require that all spent fuel that is loaded into a cask should be able to be transported just after loading or after a storage period. Classifying the spent fuel for transportation is important. The transport of intact and/or undamaged fuel is not a concern in Spain due to the relatively short storage period of about ten years, even for high burnup fuel. The transport of damaged fuel, however, is different. Several variables must be considered, including leakage, cladding corrosion, mechanical damage, or fuel assemblies with defects (so that the fuel assembly is altered such that it cannot perform its fuel-specific or system-related functions).

At the ATC, fuel from the transportation casks will be transferred into welded canisters inside a dry hot cell. The canisters will be able to accommodate either intact, undamaged, or damaged fuel (with its damaged fuel cans). Canisters are designed for 100 years of storage, which includes a renewal period. The canisters used at the ATC will accommodate a small number of fuel assemblies; if these canisters will eventually be used for transportation to a permanent disposal facility, design adjustments will be needed to meet the transport requirements. A huge number of shipments would be required to ship these small-capacity canisters to a repository, which could have a possible social impact. Other transport combinations are under evaluation.

Finally, in the United States, Dr. Peter Lyons, Assistant Secretary for Nuclear Energy, reaffirmed the Obama Administration's political decision that the Yucca Mountain site for a deep geologic repository in the U.S. is "unworkable." Lyons said he "absolutely agrees" with this position because the

country needs "new solutions." Lyons reviewed the now three-year-old recommendations of the Blue Ribbon Commission (BRC) on America's Nuclear Future, which called for prompt efforts to develop one or more geologic disposal facilities and one or more consolidated storage facilities, and to prepare for the large-scale transportation of spent fuel and HLW to one of these facilities. The BRC also recommended a new consent-based siting approach, a new organization dedicated solely to implementing the U.S. waste management program, access to the Nuclear Waste Funds, support for continued U.S. innovation in nuclear energy technology, and active U.S. leadership in international efforts. Lyons also reviewed the administration's waste management strategy, which calls for a consent-based siting process that includes agreement at "multiple jurisdictional levels," a pilot interim storage facility by 2021, a larger interim storage facility by 2025, and a geologic repository somewhere other than Yucca Mountain by 2048. The strategy also endorsed the BRC recommendation of a new organization outside of government that is "empowered with the authority to succeed."

Lyons emphasized that, "Full implementation of the administration strategy will require new legislation; however, in the meantime the department is taking action on the strategy to the extent possible within existing authorities." DOE is therefore "laying the groundwork" for implementing consolidated storage, working on transportation issues, and moving forward with a non-site-specific repository program, as well as developing a research and development (R&D) roadmap for deep borehole disposal.

DOE has two areas of focus: the Used Nuclear Fuel Disposition (UNFD) R&D Campaign, and the Nuclear Fuel Storage and Transportation Planning Project (NFST), which was established in FY 2013 to plan for interim storage and transportation of spent fuel from the reactor sites to a storage facility.

Ken Sorensen of Sandia National Laboratories (SNL) elaborated on the UNFD work, which has three objectives:

- Contribute to the technical bases to demonstrate spent fuel integrity for extended storage periods;
- Contribute to the technical bases for fuel retrievability and transportation after long-term storage;
- Contribute to the technical bases for the transportation of high burnup fuel.

This work is supporting the High Burnup Storage Cask Research Project (also known as the Cask Demonstration Project), for which the final test plan was published in February 2014, and the conceptual design review was finished in June 2014. John Kessler, who manages the Electric Power Research Institute's (EPRI's) used fuel and high-level waste programs, elaborated on the Cask Demonstration Project, which EPRI is leading.

The data from this project is needed to support license extensions for Certificates of Compliance (CoC) and for site-specific ISFSI licenses. The renewal application for AREVA TN's CoC 1004, which is in use at multiple locations, has been submitted to the NRC for review and approval, and several more renewals will be needed in the next ten to fifteen years. The NRC wants an "industry commitment" to high burnup R&D.

Kessler pointed out that EPRI's work supports "four pillars of integrated used fuel management": storage (now); transport (later); disposal (eventually); and recycling (maybe). EPRI is continuing two decades of work on spent fuel criticality issues during wet storage, such as neutron absorber degradation, and is also working on extended storage issues. EPRI's Extended Storage Collaboration Program (ESCP) is intended to "provide the technical bases to ensure continued safe, long-term used fuel storage and future transportability." This program has three phases, the first of which is complete. Phase 2 involves conducting experiments, field studies, and additional analyses to address gaps, and Phase 3 will confirm long-term dry storage performance using a full-scale dry storage system loaded with high burnup fuel.

Mark Nutt, NFST National Technical Director, elaborated on the NFST project. NFST is organized into four key elements: consent-based siting, storage, transportation, and "strategic crosscuts," such as project management, systems analyses, data and document access, etc. Integration among these elements is important.

Using existing funds and operating within the current legislative authority of the Nuclear Waste Policy Act (NWPA), NFST is making progress in several key areas. Some, not all, of NFST's work is listed below:

- Has developed and maintains a database of prior siting efforts; this database can be found at http://curie.ornl.gov/SED/pages/sed-homepage.
- Reviewing and evaluating lessons learned from prior domestic and international siting efforts, and gaining insights on consent-based siting of waste management facilities.
- Evaluating interim design concepts with input from industry contractors, including a report prepared by CB&I titled "Generic Design Alternatives for Dry Storage of Used

Nuclear Fuel." The pilot facility will be "flexible, adaptable, and expandable," to be able to handle a wide range of systems, Nutt said. Dry storage alternatives include vented concrete systems at grade in horizontal and vertical systems currently in use, vaults for dry storage canisters, universal storage overpacks, and universal underground systems.

- Evaluated costs and impacts of opening non-disposable storage canisters, and is preparing to develop a generic pilot interim storage facility design and Topical Safety Analysis Report to submit to the NRC. Revision 2 of the shutdown sites report was completed on August 30, 2014, but it is not currently available to the public. It provides information on site inventory, site conditions and infrastructure, and near-site transportation infrastructure. The report includes a site-by-site description for each of the twelve sites that were shutdown before publication. NFST visited all twelve of the sites, and after the report was completed NFST visited the Kewaunee site.
- "Laying the groundwork" for standardization of casks, which would mean the currently loaded systems into much smaller canisters.

Transportation

Without transportation, "don't waste time building a nuclear power plant," said Henry-Jacques Neau, secretary general of the World Nuclear Transport Institute (WNTI). Neau also pointed out that radioactive material transportation is the only nuclear energy related activity that takes place in the public domain, thus requiring interaction among many stakeholders. Back-end materials are transported by sea, air, road, and rail.

Key principles of transportation of radioactive material include: safety, security (the more hazardous or sensitive the content, the more robust the transportation package must be), safety in depth, and defense in depth. A number of regulations govern the transport of radioactive materials, some of which are international and some are country-specific; there are regulations for transport by sea and regulations for transport by air. A single transport may need to follow international regulations, the regulations of one country, then, if the transport crosses several countries, the transportation package must also meet the regulations of each country through which it passes. Neau said, though there are efforts underway to harmonize the regulations, much work remains.

The transport of large objects, such as the one pictured

Figure 1.

in Figure 1, can be very difficult because these large objects might not meet IAEA transport regulations (SSR-6), such as undergoing a drop test. As a result, many large components have been transported under "special arrangements" approved by competent authorities, which may be difficult and time-consuming to obtain. With the support of WNTI, a guidance document, "Guidance for Transport of Large Components under Special Arrangements," was incorporated into an appendix of the "IAEA Regulations for the Safe Transport of Radioactive Material" (SSG-26). This document provides basic concepts, recommended criteria, and specific examples. WNTI also published a fact sheet that can be found on its website at: http://www.wnti.co.uk/media/61396/FS9_EN_Nov13_V1.pdf

Neau addressed the use of dual purpose casks, noting that one issue with these casks is that the approval of a cask system design is usually relatively long for storage (decades), but relatively short for transportation (three to five years). A potential problem could arise if the transport approval cannot be renewed, either because of a revision in the transport regulations or because of a new safety review.

The IAEA has a working group that is focused on integrating the safety cases for storage and transport in a holistic manner. Some WNTI Back-End Transportation Working Group (BETWG) managers have participated in these activities, and in developing the TECDOC (Technical Documentation), "Dual Purpose Cask Safety Case for Transport/Storage Casks Containing Spent Fuel," which was finalized in April 2013 and updated in March 2014.

Waste characterization and inventory forecasting is also important because all waste streams will eventually have to be

transported, and the forecasts determine the number of packages, transport vehicles, personnel, etc., that will be required in the coming years. Forecasting may also influence capital investment requirements and programs for related facilities. Wide discrepancies between the forecasting and actual demand are reported in some countries, causing inefficiencies in the process. The BETWG has a project underway to collect, collate, and share the knowledge and experience in these areas, and a good practice guide on inventory forecasting may be developed.

WNTI is looking to determine future trends and issues for back-end transportation. Issues such as waste characterization and forecasting, repository and disposal, and harmonization of regulatory frameworks could be major challenges for worldwide back-end radioactive materials transportation. WNTI held a workshop to discuss these issues in December 2014. The workshop covered waste streams (characterizing and forecasting waste), global future prospects and challenges for nuclear waste and spent fuel transportation and storage, and country-specific examples of future prospects, challenges, and the current situation of each country.

In the United States, Lisa Janairo, program director of the Midwestern Office of the Council of State Governments (CSG), explained why states should be involved in transportation planning for shipping to a spent fuel storage or disposal facility, and what state involvement should look like.

State involvement is important because the states are responsible for the health and safety of its citizens and the environment. The states are co-regulators of transportation activities along with the NRC and the U.S. Department of Transportation, and state officials have expert knowledge of their jurisdiction in terms of the political landscape, state laws and regulations, etc.

State involvement should include: identifying and prioritizing issues, developing plans, procedures, and schedules, and public outreach to local governments and other stakeholders. Janairo said, "DOE needs all the help it can get (on public outreach), and the states can help."

More specifically, states can identify the shipping mode, evaluate and select transportation routes, train local emergency responders and state personnel, decide on plans for inspecting and escorting shipments, and identify travel restrictions. States could also help identify the site selection criteria, help develop the site selection process, and oversee the facilities.

Several obstacles to gaining state involvement are apparent. The biggest and most obvious obstacle is the fact that no spent fuel management program exists right now; accordingly, securing state participation and money on a program that does not exist is difficult. Another obstacle is that the states clearly articulate what their expectations are to DOE, but DOE, as the GAO pointed out in its analysis, does not let the states participate to the extent allowed by the National Environmental Policy Act (NEPA). Janairo asserted that DOE needs to better define state cooperation. One example of the gap between states and DOE is the development of a draft transportation plan. Submitting comments on a draft plan is not enough, Janairo said. States are expecting and needing to participate more, but all they are offered is a review and comment process.

The last obstacle Janairo mentioned is that DOE puts institutional issues on the back burner until technical issues are settled. Although technical issues are important, DOE will not be able to build trust and consent if DOE waits until all the technical issues are resolved. By then it could be too late for state input, meaning state involvement is essentially insignificant, and DOE will not be able to build the essential levels of trust and consent that will be needed on multiple levels.

The NRC's David Pstrak, senior project manager in the Division of Spent Fuel Management (DSFM), provided the regulatory boundaries of spent fuel storage and transportation. He noted that storage is governed by the regulations in 10 CFR Part 72, and transportation in 10 CFR Part 71. Spent fuel transportation requirements are in place to ensure the contents of the package remain subcritical during transportation, and to gain NRC approval, analysis of multiple packages, including consideration of potential water in-leakage, must be considered. For radiation protection purposes, the transport limit is 10 mrem per hour at 2 meters from the outer lateral surfaces/vertical plans of the vehicle, including the top and underside. The temperature of the accessible surfaces may not exceed 185˚F as prepared for transport.

In January 2014 the NRC published the *Spent Fuel Transportation* Risk Assessment (NUREG-2125), which documented the staff's analysis that radiation emitted from a cask during transportation is a fraction of the natural background radiation. The report contained the conclusion that the risk from an accidental release is extremely low, and that the regulations are adequate to protect the public against unreasonable risk.

Transporting high burnup fuel, defined as fuel that achieved at least 45 GWd/MTU, is being studied to ensure that this fuel can be safely transported after a period of storage. DSFM es-

tablished a taskforce, which has developed a Regulatory Issue Summary (RIS) that provides background and guidance for possible licensing approaches for high burnup fuel for both storage and transportation. A draft version of the RIS was expected to be issued for public comment by the end of February 2015.

Decommissioning

More than 140 reactors worldwide have been decommissioned or are undergoing decommissioning to date, and about one-third of the currently operating reactors are projected to be decommissioned by 2030, according to Larry Camper, director of Decommissioning, Uranium Recovery, and Waste Programs at the U.S. NRC, who discussed the past, present, and future of power reactor decommissioning. Out of the 100 operating reactors in the United States, seventy-two are operating with renewed licenses, and twenty-eight are operating under their original licenses. The next major wave of license terminations is expected to occur between 2025 and 2050.

In the U.S., stakeholder concerns about decommissioning plants cover several areas, including spent fuel management; high burnup fuel; long-term storage and disposal of the spent fuel; the Post-Shutdown Decommissioning Activities Report (PSDAR); and the appropriate role of the state in the decommissioning process. Programmatic challenges for the reactor operators and the NRC also covers several areas, including maintaining the decommissioning program while coping with uncertainties; maintaining knowledgeable staff throughout the process; ensuring adequate funds exist; ensuring adequate decommissioning documentation is available and provided to the NRC; ensuring stakeholders are appropriately involved in the process; lessons learned; and decommissioning timeliness.

Camper identified several keys to decommissioning project success:

- Plan for decommissioning before the facility is even built and consider decommissioning throughout operations;
- Have early and frequent consultations between regulators and owner/operators throughout the decommissioning process;
- Ensure flexibility and transparency in regulatory implementation, emphasizing safety and environmental protection;
- Use realistic approaches in selecting post-termination land use scenarios and parameters;
- Involve stakeholders throughout the process;
- The unexpected will happen — be ready to accept it and address the problem.

Representatives from two companies that are conducting decommissioning services — Robert Quinn of Energy*Solutions* and David Jones of AREVA — offered their perspectives on the decommissioning market.

In the U.S., Energy*Solutions* (ES) has decommissioned the Big Rock Point, Yankee Rowe, Connecticut Yankee, Fermi 1, and Rancho Seco plants, and is currently decommissioning Exelon's Zion Nuclear Station. ES has its own waste management infrastructure, including two major disposal facilities, two comprehensive processing facilities, and a logistics operation.

Quinn highlighted the work ES, through its wholly owned subsidiary Zion*Solutions* (ZS), has done and is doing at the Zion station. ES acquired the plant on September 1, 2010. The operating licenses were transferred from Exelon to Zion, and the Nuclear Decommissioning Trust (NDT) funds were transferred to ES/ZS as the Trustee. ZS originally estimated ten years to decommission the plant, but the current plan is to finish the project in eight years (2018), which would be fourteen years earlier than Exelon's schedule, which would have seen the site restoration complete in 2032.

The Zion model has many advantages, Quinn noted. Utilities can focus on their core business of generating electricity rather than on decontamination and decommissioning (D&D); the risk for D&D is transferred from the utility to Energy*Solutions*; accelerated cleanup can be safely achieved at a much lower cost than the *SAFSTOR* approach, which carries its own risks; and for ES, the model allows the company to book long-term projects and schedule the waste management and disposal operations earlier than under the utility model.

Specifically, the Zion project is being done in three phases. Phase 1 was the transfer of the 2,226 spent fuel assemblies from the pool into sixty-one MAGNASTOR cask systems. This phase was completed on January 8, 2015. Phase 2 is the license termination (D&D) phase, which involves the removal of the reactor vessel, removal of SSCs (structures, systems, and components), building dismantlement, and license termination. Phase 3 is non-rad site restoration, which includes landscaping, grading, etc.

When ES is done with the project, all Class A, B, and C waste will be disposed of offsite, the spent fuel and greater-than Class C (GTCC) waste will be in dry storage on the ISFSI, the land will be restored to allow unrestricted use of the site with the exception of the *ISFSI*, the NRC license will be modified to reflect ISFSI only and transferred back to Exelon, and the land will be returned to Exelon for other beneficial use.

David Jones, senior vice president of AREVA's Back End Business Group, pointed out that in the next twenty years, more than 200 nuclear power plants are expected to be closed, primed for, or begin the decommissioning process. This is more than one-half of the plants that are currently operating. Nearly three-quarters of these 200 plants will be shutdown because they have reached the end of their operating lives or are no longer economically justified to continue operations. About 18 percent of the 200 will be closed prematurely due to political or regulatory reasons, and the remaining 8 percent will be closed following an accident or serious incident. Europe, Jones said, is on track to decommission 150 reactors in the next twenty years. A 2012 estimate of the market value of these decommissioning efforts, he said is $81.5 billion.

In France, Russia, and the United Kingdom the main driver for plant closures will be that the reactors reached the end of their operating lives. The market value in these countries is estimated to be $21.5 billion in France, $13.5 billion in Russia, and $18.7 billion in the UK. In Germany, the main driver for plant closures is the political decision to phase out nuclear power following the 2011 accident at the Fukushima Daiichi plant. Germany has already closed down eight units that could be dismantled in the medium term. Final costs may exceed $32.5 billion when long-term storage costs are considered. Germany is currently dismantling four units, and nine more will eventually be shutdown. The market value of decommissioning reactors in the Asia Pacific region over the next twenty years is estimated to be over $20 billion, driven mostly by the Fukushima accident.

In the U.S., Jones said the current forecast for known D&D work is estimated to be about $8 billion. By 2034, thirty-eight reactors are expected to be shutdown, all due to reaching the end of their operating lives. Market conditions, however, have resulted in several premature plant closings, including five in 2013. More plants are expected to close early due to economic performance pressures, Jones said. Political implications and site-specific situations have also influenced decisions to close plants early. Plants most at risk are small, single unit sites in deregulated electrical markets, but technical issues are responsible for half of the reactors recently shut down. Over the next five years, an estimated five to ten plants are at risk of early closure due to adverse economic conditions, he said.

The decision to take the SAFSTOR or near-term D&D is influenced by several factors, including: the status and confidence in the nuclear decommissioning trust (NDT) fund and the decommissioning cost estimate; local and state politics; and public pressure. Most plants that shutdown prematurely have underfunded decommissioning funds and opt for SAFSTOR to allow the funds to grow, although to make this decision, the operator must assume that the costs over time will increase less than the fund will grow, which may not always be true. When projected out over several decades, there is a good deal of uncertainty and risk related to waste disposal costs, future regulatory requirements, etc.

Jones concluded by noting that early shutdown projections have bended to underestimate the actual rate of closures, and urging the industry to "safely and cost effectively deal with our retiring fleet to sustain the ability to build new plants."

U.S. Dry Storage

Mark Lombard, director of the NRC's Division of Spent Fuel Management (DSFM), provided an overview of the NRC's work in dry storage over the last year and looking forward. The overview represented much the work the NRC staff, cask vendors, utilities, and industry as a whole through organizations such as the Nuclear Energy Institute (NEI), and EPRI have done to continue to ensure the safe storage of various types of spent fuel, including high burnup spent fuel.

In 2014, the NRC staff completed seventy-one transportation cases including AREVA TN's NUHOMS MP-197 transportation package, which is the first package that is authorized to transport canisterized high burnup fuel after storage. In addition, the staff completed its review of seventeen storage cases, including Holtec International's new underground storage system, the HI-STORM UMAX, two AREVA-TN NUHOMS amendments, and a revision to NAC International's NAC-LWT Certificate of Compliance (CoC) to authorize the transport of liquid highly enriched uranium. The staff also issued a renewed site-specific license for the Calvert Cliffs ISFSI, began the review of the NUHOMS CoC renewal, and finished the first phase of transportation testing on actual fuel samples.

NRC staff, with industry participation, developed and used a revamped license renewal process for CoCs and site-specific ISFSIs. NEI submitted for the NRC's endorsement NEI 14-03, which is a document intended to provide license renewal guidance for utilities to use when preparing license renewal applications. Staff also revived the Spent Fuel Regulatory Conference (RegCon), and began a risk initiative.

During the rest of Fiscal Year (FY) 2015, the staff intends to complete its review of Energy*Solutions'* VSC-24 CoC renewal,

and the Environmental Assessment (EA) and Safety Evaluation Report (SER) for the Prairie Island ISFSI license renewal. Staff also will respond to NEI 14-03, which will feed into the draft Managing Aging Process for Storage (MAPS) NUREG report and the updated NUREG 1927, the "Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance" (will present to the Advisory Committee on Reactor Safeguards in March). Staff intends to issue a draft Regulatory Issue Summary (RIS) on high burnup fuel for public comment in the February/March timeframe, and complete the feedback on NEI's guidance for 72.48 evaluations by the second quarter of this fiscal year.

The staff's additional plans for this fiscal year include completing technical reports on functional monitoring of dry storage systems and on stress analysis of fuel cladding in dry storage systems, complete the transportation fire studies, continue with licensing process improvements (including retrievability), establish a concrete expert panel, conduct a workshop, and complete a technical report. Other activities are planned for this fiscal year as well.

After the current fiscal year, the NRC expects to receive an application for a consolidated interim storage facility. (Update: Waste Control Specialists has announced it plans to submit an application for an interim storage facility to the NRC in April 2016.)

The dry cask storage business is thriving in the United States and many other countries, due largely to the political and societal hurdles that must be overcome before a permanent disposal facility can be sited, licensed, constructed, and begin operations. With more than 2,100 dry storage systems safely deployed in the U.S., and hundreds more planned for deployment in 2015, a lack of spent fuel storage will not impede the production of electricity.

DOE Spent Fuel and HLW Storage

U.S. policy is to reduce the amount of highly enriched uranium (HEU) available in the world to help eliminate the potential for the material to be used for an improvised nuclear device, a radiological dispersal device, or other radiological exposure devices. To that end, the DOE's Office of Environmental Management (EM) partners with the U.S. Global Threat Reduction Initiative (GTRI) to secure and consolidate HEU and plutonium materials to prevent these materials from falling into the hands of terrorists, and to disposition these materials in a manner that renders these nuclear materials nonproliferable. DOE supports GTRI's Gap Removal Program through the receipt, storage, and disposition of high-risk, vulnerable nuclear materials primarily of non-U.S. origin, including plutonium from Sweden, Belgium, and Italy, and spent fuel from Chile.

Several speakers, including EM's Edgard Espinosa and David Rose of Savannah River Nuclear Solutions (SRNS) gave participants a picture of the DOE's work in securing these materials, and storing, transporting, and eventually disposing of spent fuel and HEU from defense activities and research reactors.

Rose described the Foreign Research Reactor (FRR) spent fuel acceptance program, noting that all of the spent fuel received in the U.S. as part of this program is U.S.-origin spent fuel. It is received and consolidated at the Savanah River Site (SRS) and the Idaho National Laboratory (INL), with the aluminum clad fuel stored at SRS and the non-aluminum clad fuel stored at INL. Since 1996, SRS has received approximately 9,500 spent fuel assemblies and INL has received about 2,100 spent fuel assemblies. The material received accounts for about 1,200 kilograms of HEU and about 3,640 kilograms of low-enriched uranium (LEU). The HEU and the LEU were used in foreign research reactors from countries that include Australia, Brazil, Indonesia, and South Africa. The FRR program ends in 2019, although DOE is considering a hardship exemption. Shipments are expected to be received from Canada, Switzerland, Japan, Finland, Australia, and others. Exceptions to the 2019 deadline will be made for spent fuel from Austria in 2025, and Japan in 2029. About 2,000 more assemblies are expected to be sent to SRS in about ninety-seven casks. The Domestic Research Reactor (DRR) fuel shipments to SRS have no defined end date. About 40-100 assemblies are received per year, in about five to twenty casks.

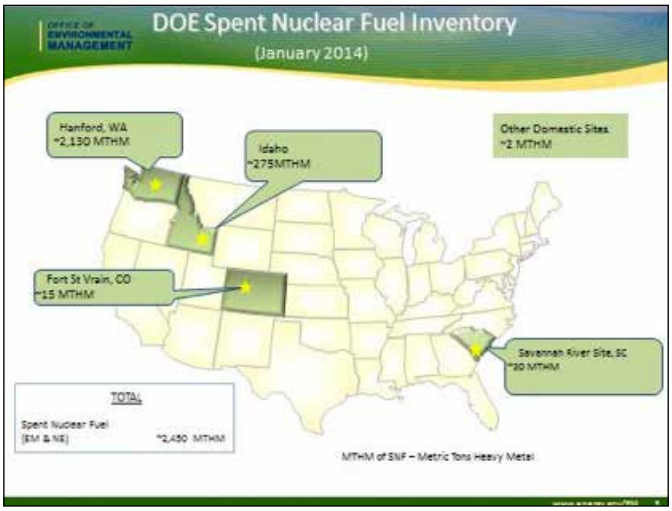
Some country-specific arrangements are detailed below:

- **Canada** — The DOE and the Atomic Energy of Canada Limited (AECL) signed a contract in March 2012 to receive HEU fuel assemblies from National Research Universal (NRU)/National Research Experimental (NRX) reactors. This HEU fuel will be sent to SRS over a period of about four years. Shipments are scheduled to begin in 2015. In September 2012, DOE and AECL signed a contract to receive Target Residual Material liquid HEU from medical isotope production. This will also be sent to SRS in a one to two year shipping campaign that is scheduled to begin in 2016. HEU from liquid HEU will be processed in H-Canyon at SRS, downblended to LEU, then shipped to Tennessee Valley Authority for fabrication into commercial fuel.

- **Germany** — EM is exploring the possible acceptance and disposition of German pebble-bed research reactor fuel that contains U.S.-origin HEU, which was provided for purposes of peaceful uses and the development of nuclear energy. This fuel was used in two research reactors in Germany, the AVR reactor — the first high-temperature reactor in Germany to test the technology of graphite spheres, and the THTR-300 reactor — a demonstration reactor built to prove the AVR concept design to produce electricity. The fuel under consideration is approximately one million graphite spheres stored in Jülich and Ahaus, Germany. These spheres contain about 900 kilograms of HEU from the U.S.. This fuel is currently stored in 455 CASTOR casks — 152 casks are from the AVR reactor and are stored at Jülich and 303 casks are from the THTR-300 reactor and are stored at Ahaus. These CASTOR casks are certified in Germany by the German regulator, and are being reviewed for acceptance as DOE/U.S. Department of Transportation certified casks.

  At the request of the German government, EM is conducting a feasibility evaluation for possible acceptance, return to SRS, and alternatives for disposition of this fuel. Research and development at Savannah River National Laboratory (SRNL), in collaboration with Forschungszentrum Jülich (FZJ), provides a means for graphite removal from the fuel kernels without the development of graphite fines as seen in mechanical graphite removal methods. FZJ provided funding for the R&D, which focused on the chemical digestion of the graphite; results to date have been "very successful." The next research steps are to maturate and validate the technology, and work the scale-up of technology and optimize the conceptual process. A large-scale production rate of execution is projected to digest 1,000 units per day. The environmental analysis of accepting this fuel is ongoing, with several options for disposition of the uranium after receipt, storage, and chemical digestion of the graphite matrix; no decision has been made. If this material is returned to the U.S., and the R&D is successful, the graphite fuel cycle would be closed, with a disposition path identified and developed. The technology would close the back-end of the fuel cycle for graphite-based fuels.
- **Japan** – In 2014, DOE and Japan reached an agreement aimed at reducing proliferation risks whereby Japan will send its HEU and plutonium to the U.S. by 2019. The U.S. will assist Japan in research reactor spent fuel management, and will cooperate on upcoming R&D projects. EM is working with GTRI on receipt and disposition options for the plutonium, as well as extending the receipt of FRR fuel until 2029. Accepting this material is also subject to completion of the appropriate environmental analyses.

Figure 2. DOE Spent Nuclear Fuel Inventory



Espinosa reviewed the status of DOE-owned spent fuel storage, and the storage of HLW at DOE sites. As of January 2014, Espinosa said, DOE has 2,450 MTHM in storage at several sites, as pictured in Figure 2. Most of this spent fuel is from DOE production reactors, but it also includes core debris from the Three Mile Island Unit 2 reactor, commercial power demonstration projects, domestic research reactors, and FRRs from forty-one countries.

In October 2014, DOE published a report, *Assessment of Disposal Options for DOE-Managed High-Level Radioactive Waste and Spent Nuclear Fuel,* that analyzed three disposal options:

- Disposal of all HLW and spent fuel regardless of origin in a single repository;
- Disposal of some DOE-managed HLW and spent fuel in a separate mined repository; and
- Disposal of smaller waste forms in deep boreholes.

The results of that indicated that "multiple disposal options are technically feasible and have the potential to provide excellent long-term isolation of DOE-managed HLW and SNF (spent nuclear fuel), and that there are programmatic advantages to a phased strategy that allows for flexibility in disposal pathways for some DOE-managed HLW and SNF." The authors thus

recommended that DOE "begin implementation of a phased, adaptive, and consent-based strategy with development of a separate mined repository for some DOE-managed HLW and cooler DOE-managed SNF, potentially including some portion of the inventory of naval SNF." Other DOE-managed HLW and SNF, including that of commercial origin and naval SNF with relatively higher heat output, would be disposed of with commercial SNF and HLW. The report also recommends that DOE "retain the flexibility to consider options for disposal of smaller DOE-managed waste forms in deep boreholes rather than in a mined geologic repository."

The conference included a number of other valuable presentations, including updates about research on spent fuel storage casks during extended storage periods, more specific information about ongoing work at Idaho National Laboratory, cask vendor technology updates, additional details on aging management programs for spent fuel casks and ISFSIs, and more. The 31st Annual Spent Fuel Seminar will be held in Washington, DC, USA, January 12–14, 2016. Based on the success of the first thirty seminars, this is one conference not to be missed.

Carlyn Greene is the executive director, Back-End Publications, at the Ux Consulting Company (UxC), where she contributes to consulting projects and reports related to spent fuel management issues. She is also responsible for the overall management of two newsletters, SpentFUEL *and* StoreFUEL, *which cover developments in spent nuclear fuel storage, disposal, transportation, and decommissioning. Greene joined UxC in May 2008 and has more than thirty years of nuclear industry experience, including six years with Washington Nuclear Corporation and nearly twenty years at NAC International. Prior to joining UxC, Greene was associate editor of* SpentFUEL *and* StoreFUEL *for Washington Nuclear Corporation, and also assisted with various consulting studies related to spent fuel storage, decommissioning, and nuclear fuel costs. With UxC's purchase of these two backend newsletters in 2008, Greene assumed the managing editor responsibilities for those products. From 1980 to 1999, Greene was employed at NAC International, where she assumed various responsibilities, including supervisor of data analysis for the Fuel-Trac database, and conducted research for the uranium and enrichment markets. She also assisted with the Uranium Price Information System, the Uranium Supply Analysis (USA) System, and the Worldwide U3O8 Producer Profiles. Ms. Greene graduated from Mercer University in Macon, Georgia, USA, with a bachelor of arts degree in English and communication in 1978, and completed graduate work in English at Georgia State University.*

Spent Fuel Management in Spain

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Abstract

Transportation of the spent nuclear fuel is one of the current important matters in spent nuclear fuel management strategy. Not only technical challenges (fuel-specific or system related safety functions), but political, social, and economic decisions are involved, such us, where and when the spent fuel will be transported.

The purpose of this paper, developed based on a presentation the author made at the 30th INMM Spent Fuel Mangement Seminar, is to show, in particular, Spanish spent fuel transport strategy from the industry point of view and some of the possible decisions to be considered by Spain. A quick update of the current spent nuclear fuel (SNF) management scenario in Spain is provided, describing the inventory, the estimation of the total amount of spent fuel over a period of forty years of nuclear power plant operation, and different spent nuclear fuel dry storage independent installations (ISFSI).

As is well known, the dry storage lifetime of spent fuel casks is limited to forty to fifty years, so what does Spain do with spent fuel afterwards? The answer to this question is directly linked to the strategy recently decided and approved for the near future, the ATC (Almacén Temporal Centralizado), the Centralized or Consolidated Interim Storage Facility. Spent fuel will be transported to the ATC a few years from now. Transportation of all the spent fuel in Spain has to be considered, this means intact, undamaged, and damaged fuel. Classification of the spent fuel is critical, as all of the spent fuel that is being loaded in a cask should be able to be transported. Intact or undamaged fuel is not a concern, but damaged fuel is different. In this case, fuel pathologies will be considered to establish a criterion for classification. This criterion will allow the industry to choose from different options in the design of the damaged-fuel cans. These damaged-fuel cans will be compatible with the transportation casks and the ATC canisters.

Quick Review of the Spanish Current Spent Fuel Scenario

A total of 4,600 tU of spent nuclear fuel is stored in Spanish nuclear power plants. Most of that is stored in spent fuel pools. Three spent nuclear fuel dry storage independent installations (ISFSI) are in operation at the moment, located at Trillo Nuclear Power Plant (NPP), Jose Cabrera NPP (being decommissioned), and the Ascó NPP. A fourth ISFSI is currently in the licensing and construction phase at Santa María de Garoña plant. The best estimation of the total amount of spent fuel in Spain, assuming each reactor operates for forty years, is around 20,000 fuel elements, equivalent to 6,700 tU.

A quick review is presented here showing the dry storage and ISFSI technology selected at these sites and the status of the spent fuel pools in the rest of the plants.

Trillo NPP is currently using the ENSA-DPT dual purpose metal cask technology. The ENSA-DPT cask is licensed for both storage and transport. Spent fuel transportation may be done immediately after loading or after a long period of storage. The ISFSI is a concrete building with air inlets and outlets, and a capacity for at least eighty ENSA-DPT metal casks. To date, twenty-eight ENSA-DPT casks have been successfully loaded in Trillo (the first ENSA-DPT cask was loaded in 2002).

José Cabrera NPP and Ascó NPP use the HI-STORM system from Holtec International. The HI-STORM is only licensed for storage, so transfering to a transportation cask (HI-STAR) is required. The HI-STAR casks were licensed in Spain in 2009 and 2012 for José Cabrera NPP and Ascó NPP, respectively. The ISFSI design at both sites is an open concrete pad with a capacity for twelve and sixteen HI-STORM units (for spent nuclear fuel), respectively.

Santa María de Garoña NPP will use ENSA's newest design technology, ENUN 52B. The ENUN 52B is a dual-purpose (storage and transport) metal cask. The storage application was approved in November 2014 by the Spanish Nuclear Safety Council (CSN) and MINETUR (Ministry of Industry, Energy, and Tourism). Transport license approval is expected before summer 2015. The ISFSI at Santa María de Garoña consists of two open concrete pads, each with a capacity of sixteen ENUN 52B metal casks. The first loading campaign is expected to begin by 2016.

Figure 1. Almaraz NPP Units 1 and 2 (left), Vandellós II NPP (center), Cofrentes NPP (right)

Figure 2. Spent Nuclear Fuel Transport Strategy in Spain (left); ATC Conceptual Design[2] (right)

The spent nuclear fuel in the rest of the plants in Spain, as mentioned before, is stored in spent fuel pools. Almaraz NPP (see Figure 1, left) is a Westinghouse PWR design with two units (unit 1 and 2) located in southwest Spain. All of the spent fuel discharged from the cores is currently stored in the spent fuel pool of each unit. Unit 1 spent fuel pool will be full by 2018, while unit 2 spent fuel pool will be full by 2021. Vandellos NPP unit 2 (see Figure 1, center) is a Westinghouse PWR design located on the northeast Mediterranean coast of Spain. The spent fuel pool will be full by 2021. Finally, Cofrentes NPP (see Figure 1, right) is a GE BWR design located in the east Mediterranean coast of Spain. The spent fuel pool will be full by 2019.

Spent Nuclear Fuel Transport in Spain

In 2004, as a result of the resolutions of the Congressional Commission for Industry, the government of Spain was urged to create a new General Radioactive Waste Plan (6th). This new plan includes, after the evaluation of different options, the start-up of a centralized temporary storage facility (called in Spanish *ATC or Almacén Temporal Centralizado*) for spent fuel and high-level waste generated in Spain and the dismantling of the nuclear power plants that reach the end of their service lifetimes.

This decision answers at least one of the next two questions: where and when will the spent nuclear fuel (SNF) be transported? Spent fuel management strategy in Spain considers the ATC as the centralized storage facility to receive and store all of the SNF from all the NPPs in Spain (see Figure 2).

The ATC design is based on a vault system (see Figure 3). This concept will safely store and temporarily solve the spent fuel and other waste storage problems for at least sixty years, with the potential to last up to 100 years. An ISFSI attached to the main building will be constructed to receive loaded casks from the plants and to store the fuel while conditioning the ATC canister for transferring and final disposal.

Although there is no official schedule for the first shipment, it is expected by the Spanish industry that this could hap-

pen sometime around 2018. For this reason, and based on the Spanish SNF inventory currently stored in the spent fuel pools and different dry storage systems, the transportation of spent fuel will consider the transport of spent fuel immediately after loading and transport of spent fuel after a long storage period. In the latter case, the maximum storage estimated time before the first shipment is around twenty years for low burnup fuel and around ten years for high burnup fuel.

All SNF will be shipped from the plants to the ATC in transportation casks. SNF will then be transferred to welded canisters inside a dry hot cell. These canisters will be able to accommodate intact, undamaged, and damaged fuel (with its damaged-fuel cans) and are designed for 100 years of storage, in accordance to the Spanish IS-20[3] and IS-29[4], IAEA SSG-15[5] and 10CFR72 [6],NUREG-1567[7] and 1536[8] regulations. If these canisters are to be used for transportation, ENSA considers that design adjustments are required to meet transport regulations. Additionally to this, ATC canisters will accommodate a small number of fuels, and as a consequence, it is presumed that a huge number of shipments should be required, with a possible social impact as it is happening in other countries. ENSA, as a cask vendor, is looking for other transport solutions to optimize the designs and minimized the operational and social impact.

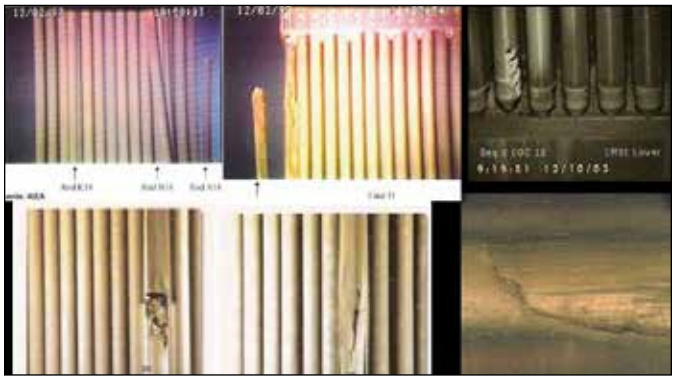
Classifying SNF for Transportation

Interim Staff Guidance ISG-1[9] is a worldwide recognized guidance on classifying the spent nuclear fuel (SNF) and is being used in Spain for this purpose. SNF may be classified according to ISG-1 as damaged, undamaged, or intact. The latest revision of this guidance defines the SNF in terms of the characteristics needed to perform the fuel-specific and system-related functions.

ISG-1 defines damaged SNF as any fuel rod or fuel assembly that cannot fulfill its fuel-specific or system-related functions, undamaged SNF as any fuel that can meet all fuel-specific and system related functions (undamaged fuel may be breached and may have assembly defects), and intact SNF as any fuel that can fulfill all fuel-specific and system-related functions, and that is not breached.

It is established in Spanish regulations that all of the spent fuel that is being loaded in a cask should be able to be transported just after loading or after a storage period. Transportation of intact and/or undamaged SNF is currently not a concern in Spain for two main reasons: a) the dry storage time of the SNF prior to any shipment will not be long, and b) compliance with cladding considerations per ISG-11, Rev. 3[10] during loading, draining, and drying activities.

Figure 3. Examples of breached SNF



The main concern in Spain, as in any other country, is damaged SNF. Several aspects on the SNF have been considered by ENSA to evaluate the relationship between damaged SNF and transport casks and are presented here.

Leakage

A fuel assembly can be considered Unbreached (leak tightness) when evidence of no leakage in all of its fuel rods can be proved, based on the radiochemistry data, inspections (i.e., UT, Eddy Current, etc.) or any other visual inspection performed. Based on the results, the following classification could be established:

- Breached (see Figure 3) or Unbreached: if there is consistency between the results of the radiochemistry and the inspections
- Unclear: inconsistency between both results. It is a standard procedure that in this case, all other fuels irradiated in the same cycle will be considered as "Unclear" fuels.

Cladding Corrosion

Classification of fuel assemblies with this pathology will be based, basically, on the oxide that has the possibility to become detached or spalled from the cladding fuel rods (Figure 4).

Using the data from different inspections, a relation between the thickness of the oxide layer and the rod average burnup is obtained, defining the average burnup for potential spalling.

This criterion will be used to establish a design requirement to be implemented in the transport cask, to assure proper behavior of the fuel under normal and hypothetical accident conditions. Note that based on this design criteria, fuels with cladding oxide may or may be not classified as damaged fuel, directly dependent on the system (fuel-specific or system-related).

Figure 4. Oxide layer spalled from the fuel cladding surface material

Figure 5. Example of mechanical damage: (a) PWR fuel top nozzle, (b) BWR fuel top nozzle, (c) and (d) Guide tube thimble sleeves mechanical damage nozzle, (b) BWR fuel top nozzle, (c) and (d) Guide tube thimble sleeves mechanical damage

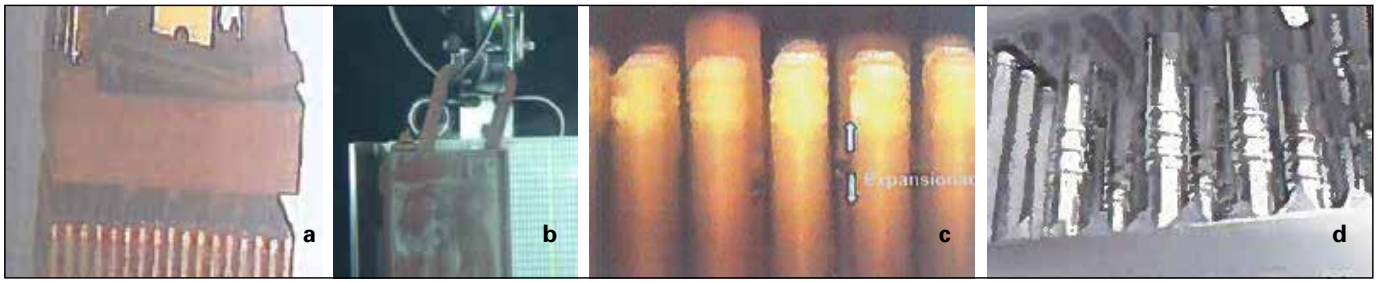
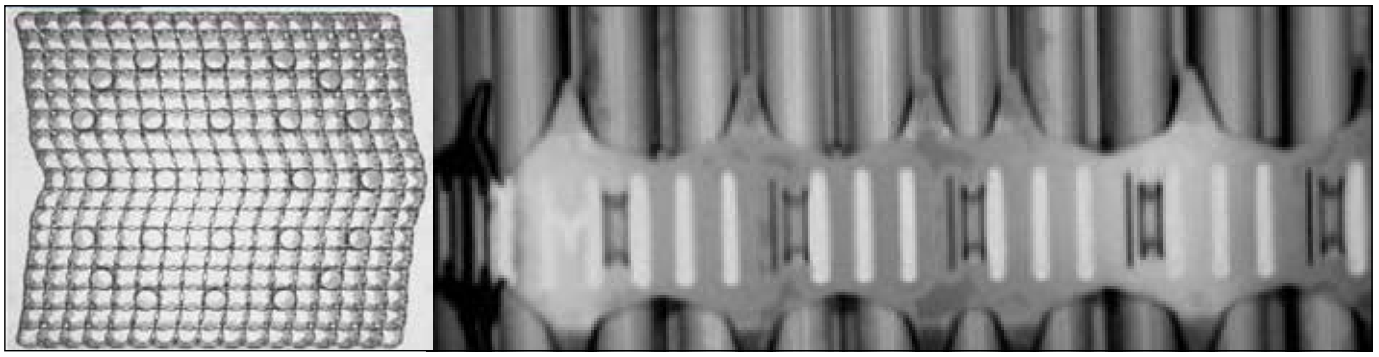


Figure 6. Example of mechanical damage in a fuel assembly grid



Mechanical Damage or Fuel Assembly with Defects

This type of pathology refers to alterations of the fuel assembly that prevent it from fulfilling its fuel-specific or system-related functions. Damaged SNF with this pathology is evaluated based on the following three fuel-specific functions:

Handling. The ability to handle individual spent fuel assemblies will be assured by the use of normal means. Mechanical defects in the Spanish spent fuel inventory are basically due to:

- Top nozzle hold-down spring bolt failure (Figure 5 (a)) due to Primary Water Stress Corrosion Cracking (PWSCC), identifying as the root cause the susceptibility of the Inconel 600 alloy to this phenomenon.
- Top nozzle-guide tubes thimble sleeves failure (Figure 5 (c) (d)) due to Intergranular Stress Corrosion Cracking (IGSCC) in the expansion area. Fuel assemblies were in contact with pure water contaminated with sulfur compound.

Integrity. Intregrity basically refers to structural integrity of the grids, with problems such as broken or missing grids or grid straps (spacers), missing or broken grid springs, weld spots failure, missing parts (Figure 6).

Stability. Although this problem should not necessarily make a fuel a "damaged fuel," an excessive bow should have an impact on the fuel-specific or system-related functions. As an example, an excessive bow or even excessive irradiation growth may have an impact on the insertion of the fuel assembly in the basket cells, generating special requirements for being loaded in the transportation cask.

Options for Transportation of Damaged Fuel

According to the different pathologies listed above, ENSA considers that the following assessment can be made to manage damaged fuel in future transportation campaigns in Spain: ***All damaged fuels can be grouped into two families based on the fuel-specific or system-related functions***.

As a result of the above conclusion, damaged SNF with leakage (Breached) may be transported inside a special sealed damaged-fuel can. ENSA is currently evaluating the feasibility of this design, which, basically, is focused on the operational feasibility. Sealed damaged-fuel cans will not only assure confinement of gross particles, debris, and missing parts, but it will also simplify the fuel-specific or system-related design assumptions (i.e., criticality). On the other hand, damaged fuel with corrosion and/or mechanical damage will be transported in a standard damaged-fuel can, with proven operational experience. If the sealed damaged-fuel can is found not to be a feasible solution, Breached SNF may be allocated in the standard damaged-fuel cans. In this case, fuel-specific and system-related design assumptions will be more conservative with direct implications on the transportation cask loading scenario.

ENSA Experience in Spent Fuel Management

Equipos Nucleares, S.A. (ENSA), a leading company in the manufacturing of nuclear components, has been active in the spent fuel management area since the mid-1980s. This first incursion was the development of the CENTAURO cask together with the Spanish Center for Energetic, Environmental, and Technological Investigations (CIEMAT).

During the 1990s and in order to accommodate the SNF from Trillo NPP, the ENSA-DPT dual purpose metal cask was developed based on the model NAC STC-26, already approved by the U.S. Nuclear Regulatory Commission. ENSA was responsible for the design, analysis, manufacturing, scale model for drop testing, fabrication thermal tests, and cask loading. Enresa (National Radioactive Waste Company) was in charge of the licensing. NAC supported ENSA on the nuclear analyses during the design phase. At present, there are twenty-eight ENSA-DPT casks loaded at the Trillo NPP. This cask is able to accommodate twenty-one KWU 16x16-20 fuel assemblies.

A new conceptual design for a BWR dual-purpose metal cask was developed for Hitachi Ltd. in 1998. This new cask had to combine both a maximum capacity and a competitive cost within the Japanese market thinking on Mutzu site. ENSA developed the conceptual design, manufacturing feasibility, structural and, thermal analyses, material evaluation, design criteria and, shared responsibility in the performance and further evaluation of drop tests on a one-third scale model and a thermal test on a full scale cask. Hitachi Ltd. has sold fifty units of this cask for Mutsu project and thirty for Hamaoka. The delivery of the first cask is planned for 2015.

With the application of new standards and regulations, beginning in 2000 and for a decade, ENSA carried on the investigation of new materials, analysis methodology, etc., so as to be applied to a new metallic cask design on a more competitive basis than those of the ENSA-DPT. As a result of this investigation, new different R&D projects arose culminating in a new design of a dual purpose metallic cask called ENUN (ENSA UNiversal). The R&D projects have been focused on a cask design for both PWR and BWR spent fuel and with a new impact limiters design, using new materials that keep the lowest G loads in a drop event even with a big size cask.

These projects are of the sole responsibility of ENSA, which is acting as designer and licensee at the same time. Such responsibility gathers the complete design, analyses, manufacturing feasibility studies, material analysis and evaluation, scale model for drop testing, regulator's relations, licensing, etc. Currently the BWR design version for Sta. Mª de Garoña fuel (ENUN 52B) has already been licensed and five units are being manufactured. The approval of the PWR design version (ENUN 32P) is foreseen for the fourth quarter of 2015. ENUN 32P has been designed to accommodate most of the Spanish PWR fuel types.

During these last years, ENSA has been making a commercial effort to potentiate the design, licensing, and manufacturing of spent fuel components, not only within the national market but also worldwide. Within this goal, at the end of 2013 an ENUN 24P (PWR AFA 2G/3G/3GAA fuel types version) was sold to Chinese Daya Bay Power Plant, fully designed and licensed by ENSA in Spain giving the necessary support to the customer for its licensing in China by the NNSA.

It is important mention is ENSA's participation in the Spanish campaigns of spent fuel loading of casks made on both its own designs and other designs for the Business Development Area of ENSA. This team will also give support and training in China for the ENUN 24P. ENSA is the leading, supervising, and controlling company throughout the spent fuel cask management cycle, from the beginning of the conceptual design of the

cask, through the analyses, final design, licensing, manufacturing, inspections and loading. Operating and maintenance procedures are handed over to the final customer.

In the area of Spent Fuel racks ENSA has performed the delivery/re-racking in all spent fuel pools of the Spanish nuclear power plants starting in the 1990s at Asco NPP and Almaraz units 1 & 2 with a Siemens design, and finalizing in 2009 with ENSA's own technology (Interlock Cell-Matrix with ENSA patent) for Cofrentes NPP. ENSA has also supplied Interlock Cell Matrix design racks to General Electric (USA), Lugmen and Kuosheng (Taiwan), Olkiluoto 2 (Finland), Yong-Gwang (South Korea), Ling Ao 3 y 4 (China).

ENSA is at present licensing in France the re-racking of spent fuel racks for 3 EDF French NPPs (Penly, Nogent & Catenon) with own design (patented Interlock Cell-Matrix). In the same way as for the spent fuel casks, ENSA is the leading, supervising, and controlling company throughout the whole cycle, including spent fuel handling.

The experience of leading, supervising and controlling the whole spent nuclear fuel management cycle allows ENSA to optimize our own cask and racks designs, as well as all the ancilliary equipment, making proprietary design highly competitive and with excellent quality.

David Garrido earned his mechanical engineering degree in 1997 and started his professional career in the nuclear business in 1998 as structural and thermal analyst.

A few years later, he became project manager of all the spent fuel cask fabrication at ENSA's facility. After six years of manufacturing experience, he has been leading the design and licensing team of the new ENSA's Spent Fuel Cask design (ENUN) since 2008.

ENSA is a globally recognized multi-system supplier of NSSS components also providing a variety of services, casks and racks to plants. ENSA has a complete proven capability from design thru operation, including proprietary cask designs for PWR and BWR fuel and non-fuel hardware, both for storage and transportation purposes. ENSA has manufactured and loaded all of the used fuel casks in Spain used by Enresa, both of proprietary designs and of other designs. ENSA is fully owned by the SEPI group, a holding with major ownership of sixteen public companies and more than 75.000 employees in 2013.

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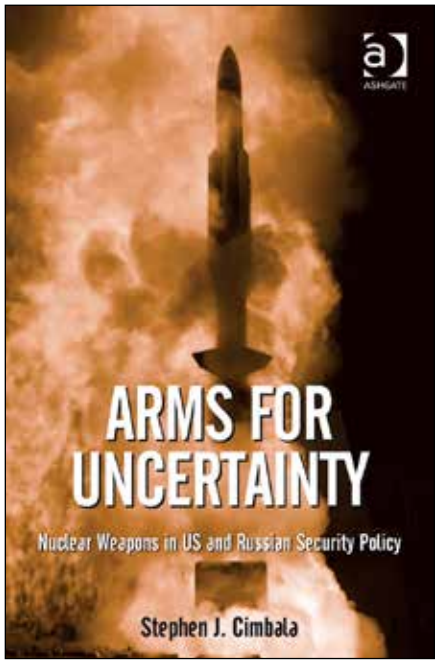
Book Review

By Mark L. Maiello, PhD
Book Review Editor

Arms for Uncertainty Nuclear Weapons in U.S. and Russian Security Policy

Stephen J. Cimbala
Hardcover, 230 pages
ISBN 978-1-472-409850
Ashgate Publishing Company
Burlington, VT, 2013



The somewhat baffling title supported as it is by a lengthy subtitle does little to inform the reader of the fascinating subject matter of this book. Herein resides a look backward and forward into the policy, strategy, and arms control of nuclear weapons. Authored by the formidable talent of Penn State University's distinguished professor of political science, the book is a serious analysis of Cimbala's three "nuclear ages." The first ended with the demise of the Soviet Union. The second age is the one the world finds itself currently in. The third is estimated to arrive in 2020 if nuclear arms spread to state and perhaps non-state actors. A firm belief in the permanence of nuclear weapons underscores the discussion. Nuclear abolition is however, given its due.

In ten chapters, the discussion frequently considers the modern spectrum of warfare that now includes the increased efficiency of precision targeted conventional weapons, more effective defensive measures against attacking weapons, and the parallel offensive in cyberspace that will undoubtedly accompany an offensive against a computer-dependent adversary such as the United States. How these new technologies affect theoretical deployment and use of nuclear weapons and the nonproliferation discussion are some of the major contributions of this work.

The considerations of the author begin with the current situation, itself very fluid as the adversaries against that nuclear weapons might be used are no longer as well defined as they were in the Cold War era. Future nuclear worlds are postulated in which potential adversaries and allies may emerge. Minimum deterrence as reflected in the number of nuclear warheads in this new world of modern warfare is another result of technological change affecting policy and strategy. As mentioned, the equally modern idea of nuclear abolition is accommodated with a chapter of its own. Discussions of the relevancy of first-use in today's regime, the nuclear arsenals of Asia including the problem posed by the nuclearization of North Korea and, NATO/Russia missile defense issues are devoted to separate chapters. The book is replete with novel discussions but none so much as the chapter given to stopping a nuclear war initiated using the computer controlled armaments and command structures of the 21st century.

Professor Cimbala is as well-armed as the weapons he analyzes. This book reveals his enviable depth of knowledge. Each chapter is heavily footnoted with reference sources that illustrate the infinite extent of the author's research. He has left the curious a generous assortment of breadcrumbs leading to further enlightenment. A healthy ten page select bibliography for further reading is testament to his underlying intent to teach. But many books by many authors take such an approach. A notable difference here lies in the analysis. The author, a political scientist — not a physicist or engineer — shows no fear in his use of computer models to support much of his research, particularly that of his estimates of the minimum nuclear deterrent that the United States and Russia currently require. Forty-one tables of modeling results and other data pepper his chapters. Rest easy for the narrative does not bog down under mathematical analysis nor does he divert the discourse to the largely tangential inner workings of the models. However, one cannot have it all: the author does assure that

the political science message remains on course but to the detriment of a discussion of where the models might go wrong (and we know that models can go wrong). In short, there are no error terms on the results. Although this pool of analytical quicksand is avoided, the reader is advised to pay strict attention to the text. This is heavy-weight material — reader prerequisites include consciousness and the ability, or, at minimum, the desire to self-enlighten. Indeed, it is somewhat unfortunate that the author's lead chapter is largely a model-based deterrence analysis that may temporarily dissuade a student or newly initiated reader from continuing onward. The subsequent chapters are where the intellectual candy is hidden.

Cimbala's analyses show that if not for the harsh reality of politics and due to the unfortunate lack of a comfortable alternative to deployment, Russia and the United States could position between 500 and 1,000 warheads under most proposed force structures and operational regimes to allow either nation a reasonable retaliatory capability. For now, such relatively paltry levels remain the chess pieces of academicians rather than the sought after goals of politicians. Such reductions await a true pan-European/NATO/Russo security establishment to be constructed that obviates the need for deterrence. Additionally, a broader "grand bargain" between other nuclear weapons states would be necessary. Such a deep draw down from the 2,000 or so long-range devices that remain operational in both the United States and Russia seems ambitious. However, as the author implies, international relations have long since reached the limits of the first nuclear age and they must evolve. A combination of conventional forces, now augmented by precision guidance ordnance, international agreements, and diplomacy may lead the way to what he calls the potential "consolation prize" of nuclear abolition.

The author excels at marrying the current state of electronic warfare to the discussion of nuclear arms control. This revealing discussion has many layers. Nuclear weapons are to a limited extent superseded by these computer-assisted precision-targeting conventional forces. They can be as effective as nuclear weapons under some circumstances.

Cyber warfare has also changed the nuclear war game plan. Interfering with the command and control structures of nations adds frightening aspects to the initiation — or the potential termination of nuclear war. Cyber attacks are part-and-parcel of warfare, being so routine now that only the most sensational are reported such as the December 2014 attack on the Sony Corporation apparently by the irate, illiberal and "interview"-adverse North Korean government. In the chapter titled "Controlling Nuclear Crises in Digital Times," Cimbala asserts that cyber attacks do not merely disrupt communications between policy makers and military operatives. Intentions and capabilities are clouded by false information rendering data unreliable such that false detection of attack and detection of false attacks could become nightmare realities for military and government leaders. Cyber assaults wreak havoc by inserting misinformation into the data-collection stream rendering for example, damage estimates inaccurate. Such data corruption creates environments that diminish options for government officials and in so doing, narrow the problem solving abilities of those that may seek creative solutions to a nuclear crisis. In short, they increase the probability that a graceful "way out" for either opponent will not be successful. Not as theoretical as you may think, the results of military hacking are supported by analog examples from the Cuban missile crisis. This chapter upgrades our understanding of nuclear warfare from the perspective of the digital environment — a frightening but necessary education.

In much the same manner as the preceding discussion, Cimbala presents a detail-rich discourse on nuclear war termination, citing the unpredictable affects of cyberwar on de-escalation of a nuclear conflict. But, the chapter goes much further, analyzing the many challenges to escalation control. Over decades and mainly by trial and error, American and Soviet command and control operators and analysts established a reliable system that kept intruders out while still rendering accurate responses to their respective command authorities. It is unclear whether in the new nuclear weapons states, e.g., North Korea, that such a control structure exists. India, Pakistan, North Korea, and Israel do not regularly advertise state secrets such as the delegation of authority between government officials and military field commanders regarding the launching of nuclear weapons. Furthermore, we do not know for sure whether deterrence in these states is based on preemptive considerations or second-strike capabilities. This is important because surviving nuclear forces may tip the balance of war-termination from surrender to negotiated peace (many states will desire a face saving option if presented).

But, will the leaders survive the initial attack, remain in contact with military commanders, be capable of restricting targeting and if decided, terminating the

attack? Cimbala points out that rogue commanders may be bent on revenge, failing to put their training and their professional obligations to the state in the forefront of their duties. As he points out, some may and some may not—the latter resulting in the genie never returning to the bottle. To make matters worse, mature nuclear states like Russia cannot be ruled out as escalators of nuclear conflict. This was especially true in the 1990s when its failing economy began deteriorating its conventional forces and putting pressure in its nuclear capabilities to defend the state. The latter also deteriorated forcing the United States to become invested in Russian nuclear material security. Smaller states with limited nuclear arsenals may instigate conflict and perhaps be less willing to end it. The philosophy of "use the nukes or lose them" may prevail. Yet, Cimbala warns that larger states may suffer similar syndromes under certain circumstances. The response of a nation to a nuclear attack is dependent not only on technology but on a state's national policy of decision making. He includes the disturbing thought that the personalities of a nation's leadership and the mood of its public are also factors when considering the prolonging or ceasing of nuclear war. Weren't all terrorists and states aiding terrorists targeted for vengeance by the United States after 9/11?

Cimbala concludes with practical advice. Nuclear abolition is a laudable goal but difficult to achieve when everybody else says "you go first." But policy makers should not assume that this means the reduction of arms is not worthy of time and effort. Cimbala quips that the fewer nuclear weapons, the better for all of us. With 21st century terrorism abounding and the occasional cross-talk with states such as North Korea a high probability, the nuclear ambitions will be a source of concern and frustration for the established nuclear powers. The United States, NATO and Russia must conclude that their mutual interests including security policy, outweigh their disagreements. Peace must be negotiated, nurtured, and backed by force if need be. It is not an automatic default position.

This particular component of the discourse put forth by Professor Cimbala is one of the most current and thought provoking to be had. To drive home the effects of indifference to current arms control and nuclear engagement issues Cimbala in a "what if" scenario, describes in a mere two pages the escalation of a middle east conflict into a Pakistan/India nuclear exchange that leaves the U.S. and its allies economically devastated and Russia with the means to further its ambitions in world politics. The nuclear agenda will always need attention lest as the author puts it, a "future history making event" such as this comes true to the detriment of all.

Taking the Long View in a Time of Great Uncertainty

Making Sausage — A View into Creating the Annual Meeting Technical Program

By Jack Jekowski
Industry News Editor and Chair of the Strategic Planning Committee

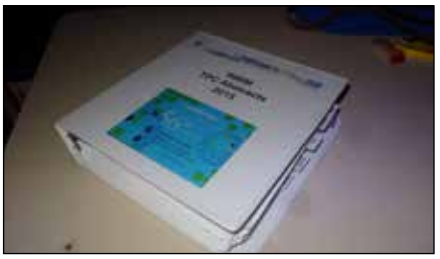
Three years ago I attended the Technical Program Committee (TPC) meeting for the first time. The TPC is usually held the day prior to the March Executive Committee meeting each year to make it easier for those attending both gatherings to participate. The TPC meeting brings together more than fifty experts from the Institute's Technical Divisions to construct the complex agenda for the Annual Meeting. In 2012 the meeting location was Orlando, Florida, USA, and there was a sadness that hung over the paricipants as they reflected upon the perennial leader of the activity, Charlie Pietri, who had passed away just the month before in Feburary.[1] Steve Mladineo, pictured here, volunteered that year to lead the effort, as many others "chipped in" to make sure it was a success, including long-standing participants Paul Ebel and Teressa McKinney, who is now the TPC Chair.

Preparing for the Technical Program Committee Meeting

As the new chair of the INMM Strategic Planning Committee in 2012, I felt a need to have a better understanding

of how the TPC process worked. Many described it as "making sausage," so it was intriguing to think that I could get a glimpse into the inner workings of the Institute. Little did I know that this annual event would become addictive, and one that I have participated in ever since. Not only do I get to preview the Annual Meeting and network with the leadership and "movers and shakers" of the Institute, but also the meeting provides insight into critical strategic issues and topical areas of interest for our membership.

That year I had diligently downloaded, printed, and organized more than 500 abstracts that had been submitted for consideration into a three-ring binder, and sorted them under Technical Division Tabs (the photo shows this year's binder).



I started reading the abstracts the weekend before the meeting, and continued on the flight to Orlando, as I ran into other participants coming in from across the country, also diligently reading abstracts on the airplane, and putting colored sticky tabs on groupings of papers that could potentially align to be a technical session. I had no idea what I was in store for the next day — I was supposed to be an "observer" but was called in to help organize the new education and training sessions. It was a trial by fire.

Technical Division Discussions and the Art of Horse-Trading

The six technical division chairs and the chair of the Education and Training Committee have a table in the meeting room, and at 7:30 a.m. folks begin to gather and talk about the abstracts that have been pre-identified for this year's sessions — and in some cases, they immediately start the process of horse-trading with other divisions for abstracts they believe would be of benefit in one of their sessions or more appropriate for a particular technical track they are creating. Even with multiple discussions occurring simultaneously, there is a certain synergy

This column is intended to serve as a forum to present and discuss current strategic issues impacting the Institute of Nuclear Materials Management in the furtherance of its mission. The views expressed by the author are not necessarily endorsed by the Institute, but are intended to stimulate and encourage JNMM *readers to actively participate in strategic discussions. Please provide your thoughts and ideas to the Institute's leadership on these and other issues of importance. With your feedback we hope to create an environment of open dialogue, addressing the critical uncertainties that lie ahead for the world, and identify the possible paths to the future based on those uncertainties that can be influenced by the Institute. Jack Jekowski can be contacted at jpjekowski@aol.com.*

that is created among the groups, and by mid-morning flip charts begin to be populated and put up on every available inch of wall space in the room. It is not uncommon for other discussions to occur about special sessions, plenary speakers, and logistics for the meeting at the same time, including the size and availability of presentation rooms at the Annual Meeting hotel, and other activities that might need to be scheduled for the meeting. If you pay attention you can pick up a lot of interesting information. Throughout all of this, the INMM headquarters staff also assists and handles issues as they arise. For a short video of this year's morning session to get a sense for the activity, see http://youtu.be/UA44OsRr_II.

Roll Call, Orphans, and Dupes

As the morning rolls into lunch time, the TPC chair checks with each group to see if they are ready for the first roll call. The groups by now have populated flip charts of abstract numbers and hopefully have been able to capture every abstract they are responsible for (including the ones that have been traded) into technical session groupings. If they all agree, we are ready for the first roll call; if not we adjourn for lunch and take a new pass at the compilation after lunch. This year, partly because a new process was instituted to handle "orphans" (abstracts no one claimed) and duplicates (abstracts more than one group claimed — otherwise known as "dupes"). The first roll call did not happen until after lunch. The flip chart shown is from 2012 and shows how many orphans and dupes were discovered after the first roll call. This year, INMM staff used an automated system to cross-check for orphans and dupes prior to the first roll call. it was hoped that by doing this the first roll call would be (almost) perfect. The picture of Corey Hinderstein on the floor shows the anomalies the automated system identified that were corrected prior to the first roll call vote this year. Some in the audience found it hard to believe such a system would work well, and several others connived to yell out "got it" when the first abstract number was called as a joke (proving that even technical nerds have a sense of humor). It created a great laugh, but then we settled in with the first roll call of abstracts, and remarkably, only two or three anomalies were found through the

400+ abstracts. Two video clips of the afternoon session, including the call out by several people of "Got it" when the first abstract number was called, can be found at http://youtu.be/ajHC3Az7rbI and http://youtu.be/39Wa6O4r5zw. This was the first year most folks could remember that the alignment was accomplished on the first roll call!

Some Unexpected Side Benefits

In 2012, the participants were treated to a special presentation by Paul Ebel on his efforts over the years to help speakers and session chairs improve their presentation and session management respectively. Each year at the annual meeting, Ebel helps to orchestrate the technical sessions and provide guidance and support to the presenters. He has also done an online webcast for student chapter members to help them as they begin their journey in the field of nuclear materials management. Oh, and to reward all of the participants, once a good roll call vote has been completed in the late afternoon, an ice cream cart is rolled in and everyone gets to have a sugar boost to get them through the final phase of the meeting — identifying session chairs, room size, and scheduling. The final details are then collected on the flip charts by headquarters staff and the TPC chair and used to put the complete program together. It is not uncommon when that

process occurs for further discussions to take place to make sure the final Technical Program comes together seamlessly. A special computer program is used to complete that phase of the Program creation.

Capturing the Environment

It is hard to describe in words, or even pictures and videos the remarkable experience of a TPC meeting. It is an important part of the overall mission of the Institute and one that participants take very seriously, and put an extraordinary effort into to make sure the attendees of the INMM Annual Meeting have a rewarding and value-added experience. If you see anyone who participates in the TPC at the Annual Meeting, including our headquarters staff, be sure to thank them for their service and talk to them about the possibility of helping in the future — you will find it a very rewarding experience!

The Final Takeaway

The health of the Institute has never been better. The quality, diversity, and technical depth of the abstracts submitted for this year's Annual Meeting in Indian Wells, California, USA, are nothing short of amazing, with more students and international presenters than ever providing a strong and diverse technical program.

Reference

1. See the *Journal of Nuclear Materials Management*, 2012, Vol. 40, No. 3, for a Memoriam by Yvonne Ferris, and other remembrances of Charlie's long contributions to the Institute.

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