Georgetown
University
Hospital
MedStar Health

J-6
MS-16

August 4, 2011

Dennis Lawyer
Health Physicist
Division of Nuclear Materials Safety
U.S. Nuclear Regulatory Commission, Region 1
475 Allendale Road
King of Prussia, PA 19406-1415

03035409

Reference: License #08-30577-01 License Amendment
Supplemental Information for Request Dated 26-July-2011

Dear Mr. Lawyer,

This correspondence is written to address additional questions in relation to the request for an amendment to the Medstar Georgetown Medical Center, Inc. (MGMC) radioactive material license (#08-30577-01) dated 26-July-2011. The information related to each query is attached in sequence.

Thank you for your continued guidance and support of the programs and initiatives promoting the safe use of radiation at Georgetown University Hospital. Please feel free to contact me for questions or concerns related to this correspondence.

Sincerely,

mla

Maxwell Amurao, PhD
Radiation Safety Officer
Medstar Georgetown Medical Center, Inc.

The following narrative and attachments are being written in response to the question below (italicized) describing attempts to decontaminate the concrete building.

1) *10 CFR 20.1402 gives the radiological criteria for unrestricted release. One part of the criteria is that the levels have been reduced to levels that are as low as reasonably achievable (ALARA).*

   What activities have been completed to attempt decontamination of the area exhibiting radioactive contamination listed in Table 4 or the "Final Status Survey Report" dated July 19, 2011 to demonstrate ALARA?

Weekly area surveys were being performed for the concrete building up to the week of June 20, 2011 to ensure acceptable radiation levels within the structure as well as the absence of contamination from stored radioactive material. An example of a weekly area survey for the week of June 20, 2011 is shown in attachment 1.

In the week of June 27, 2011 disposal of several DIS containers took place. Containers that were "not indistinguishable from background" and could not be disposed of were transferred to the Radiation Safety Lab in the Gorman Building room GL027. The decay-in-storage log reflecting these containers are shown in attachment 2.

After transferring the DIS containers, area surveys were performed on three separate days (June 28, June 29, and June 30) following transfer of the DIS containers to verify whether further decontamination is needed. See attachments 3, 4 and 5 respectively. Area surveys were performed with a calibrated scintillometer. Area survey results reflected that ambient radiation levels were at background on each of the three days.

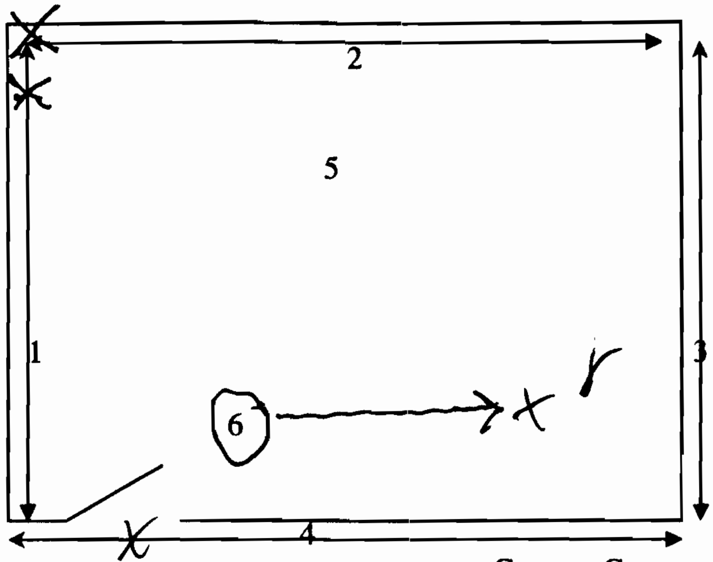
On June 30, area surveys and wipe tests were performed on four quadrants of the floor, the center of the floor, and each of the four walls of the structure. Results reflected that radiation levels were also at background as seen in attachment 6. Surface count rates were also taken on miscellaneous items in the room, such as a fire extinguisher, a steel cart, boxes, totes, etc. All surface count rates were "indistinguishable from background" (see attachment 7 and 8).

Weekly area surveys and wipes were still performed for two weeks after the DIS containers were removed from the concrete building to provide continued evidence for radiation levels being "indistinguishable from background". Measurements from July 8, 2011 and July 15, 2011 are included as attachments 9 and 10 respectively. A final status survey was performed by a third-party vendor on July 19, 2011.

In summary, DIS containers stored in the concrete building were either disposed of or transferred to the Radiation Safety Lab to reduce radiation levels in the structure that satisfies ALARA. Ambient radiation levels as well as removable contamination were measured and documented on several different occasions to verify that remaining radiation levels are "indistinguishable from background".

RSO Weekly Survey
Concrete Building

Attachment 1



Location of Measurement	Gross mR h^{-1}	Gamma Count CPM	Gamma Count DPM	LSC CPM	LSC DPM
Background	0.005	278	—		
1. Int. Wall	0.33	—	—		
2. Int. Wall	0.22	—	—		
3. Ext. Wall	0.00	—	—		
4. Ext. Door	0.01	—	—		
5. Ext. Tl-201 Sharps	0.04	354	79		
6. Ext. Tl-201 Sharps	0.01	323	47 (A)		

Notify the RSO when the restricted area exceeds 2.0 mrem h^{-1}, an unrestricted area exceeds 0.2 mrem h^{-1}, or removable contamination exceeds 200 dpm ·100 cm^{-2} (iodine-131), and document any action taken.

Survey Meter: Minispec S/N: 1560

Calibration Date: 02/10/2011 Check-source reading: 157 μR h^{-1}

Gamma Counter: Capintec Captus-600 S/N: 600277

Liquid Scintillation Counter: Malisa Star S/N: 35003

Comments: (A) Eff. = 96% Tl-201 (presumed)

Surveyor: [signature] Date: 06/23/2011

Reviewed by: [signature] Date: 24-Jun-2011
RSO or designee

Decay-in-Storage Log

Attachment 2

Storage Information					Disposal Information				
Storage Date 2011	‡Nuclide	‡Estimated Activity in µCi	‡Projected Disposal Date 2011	Surveyor's Name	Gross Reading at Surface in µrem/h	Bkgd. in µrem/h	Labels Defaced (Y/N)	Disposal Date or Return-to-Storage 2011	Surveyor's Name
04/21	Tl-201	<1	04/30	C. Alston	6	6	y	06/20	C. Alston
04/25	Tl-201	400	05/25	↓	5	5	y	06/16	C. Alston
04/25	Tl-201	3600	06/25	↓	6	6	y	06/~~16~~ 28	C. Alston CJA
04/25	Tl-201	67	05/19	↓	6	6	y	06/20	C. Alston
04/25	Tl-201	17	05/10	↓	6	6	y	06/28	C. Alston
05/02	In-111	62	06/05	C. Alston	5	5	y	06/16	C. Alston
01/29+01/28	I-131	6300	05/29	C. Alston	6	6	y	06/20	C. Alston
02/17	I-131	1000	05/07	↓	5	5	y	06/16	C. Alston
02/18	I-131	950	05/16	↓	5	5	y	06/16	C. Alston
05/09	Tl-201	34	05/20	C. Alston	6	6	y	06/20	C. Alston
05/09	Tl-201	510	06/18	↓	6	6	y	06/28	C. Alston
05/09	Tl-201	45	05/30	↓	6	6	y	06/20	C. Alston
05/16	Tl-201	1100	06/30	C. Alston	6	6	y	06/28	C. Alston
05/16	In-111	8	06/06	↓	6	6	y	06/20	C. Alston
05/16	Tl-201	110	06/16	↓	6	6	y	05/20	C. Alston CJA
05/18	Tc-99m	33	05/20	C. Alston	7	7	y	05/20	C. Alston CJA
05/18	Y-90 Sir.	2000	07/08	C. McDowell	7	7	y	07/15	C. Alston
05/18	Y-90 Sir.	2300	07/08	↓	7	7	y	07/15	C. Alston
05/18	Y-90 Sir.	32,500	07/08	↓	7	7	y	07/20	C. Alston
CJA ~~05/18~~	FE014V06								
05/18	Y-90 Sir. FE015V19	86,400	07/08	C. McDowell	7	7	y	07/20	C. Alston
05/23	Tc-99m	10	05/29	C. Alston	6	6	y	06/20	C. Alston CJA
05/23	Tl-201	3400	07/23	C. Alston	7	7	y	07/30	C. Alston
05/26	Tc-99m	3	06/01	C. Alston	6	6	y	06/20	C. Alston CJA
05/27	Tl-201	340	06/17	C. Alston	6	6	y	06/20	C. Alston CJA
06/08	Tl-201	340	07/23	C. Alston	7	7	y	07/30	C. Alston

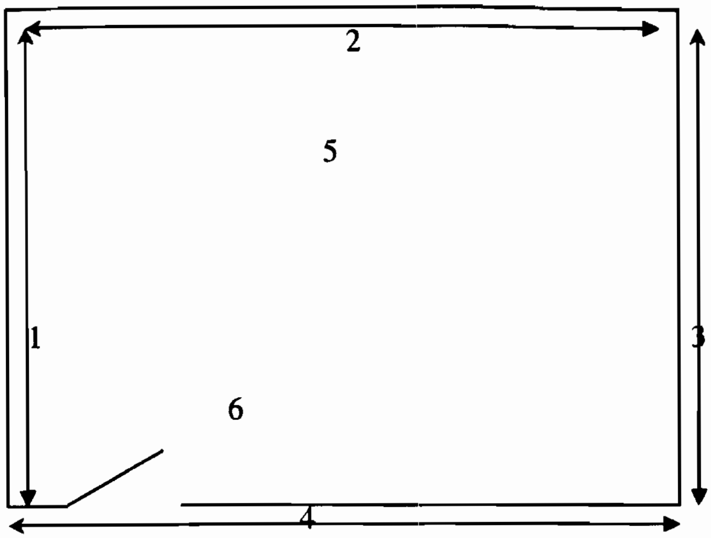
Survey Meter: Model GR-130M SN 1560 Calibrated 02/10/2011

‡ Information to be written on the waste label or tag.

† These columns completed after decay. Note: Sirspheres stock vials discarded in sharps box in Nuclear Medicine. — CJA

RSO Weekly Survey
Concrete Building

Attachment 3



Location of Measurement	Gross mR h^{-1}	Gamma Count CPM	Gamma Count DPM	LSC CPM	LSC DPM
Background	0.007	B			
1. Int. Wall	0.003				
2. Int. Wall	0.003				
3. Int. Wall	0.003				
4. Interior Areas A	0.003				
5. ↓ ↓	0.003				
6. ↓ ↓	0.003				

Notify the RSO when the restricted area exceeds 2.0 mrem h^{-1}, an unrestricted area exceeds 0.2 mrem h^{-1}, or removable contamination exceeds 200 dpm ·100 cm^{-2} (iodine-131), and document any action taken.

Survey Meter: Minispec S/N: 1560

Calibration Date: 02/10/2011 Check-source reading: 158 μR h⁻¹

Gamma Counter: Capintec Captus-600 S/N: 600277

Liquid Scintillation Counter: Malisa Star S/N: 35003

Comments: A Includes all equipment and materials. See attached listing. B See alternate survey form.

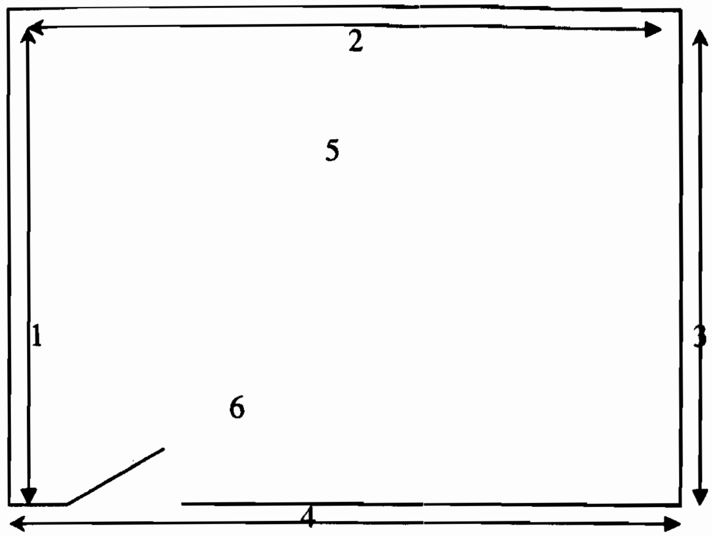
Surveyor: O. J. Ott Date: 06/28/2011

Reviewed by: MK Date: 8 July 2011
RSO or designee

Draft Form 0.44 01/04/2011 cja
RADSHARE\Section 0 - Forms\Radiation Safety 0.3 thru 0.5\Form 0.44_Survey_Waste_File12.2.doc

RSO Weekly Survey
Concrete Building

Attachment 4



Location of Measurement	Gross mR h^{-1}	Gamma Count CPM	Gamma Count DPM	LSC CPM	LSC DPM
Background	0.005	B			
1. Int. Wall	0.003				
2. Int. Wall	0.003				
3. Int. Wall	0.003				
4. Entire Interior A	0.003				
5.	0.003				
6.	0.003				

Notify the RSO when the restricted area exceeds 2.0 mrem h^{-1}, an unrestricted area exceeds 0.2 mrem h^{-1}, or removable contamination exceeds 200 dpm ·100 cm^{-2} (iodine-131), and document any action taken.

Survey Meter: Minispec S/N: 1560

Calibration Date: 02/10/2011 Check-source reading: 157 μR h⁻¹

Gamma Counter: Capintec Captus-600 S/N: 600277

Liquid Scintillation Counter: Malisa Star S/N: 35003

Comments: A Includes all equipment + materials. See attached listing. B See alternate survey report.

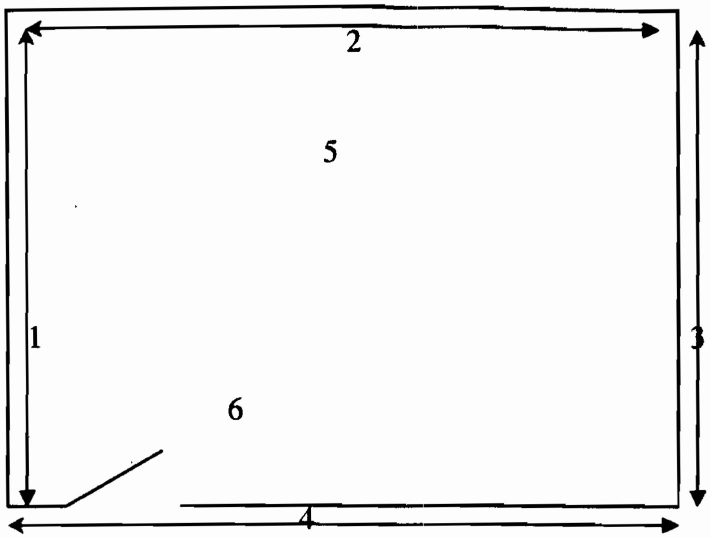
Surveyor: C. J. [signature] Date: 06/29/2011

Reviewed by: [signature] Date: 8-July-2011
RSO or designee

Draft Form 0.44 01/04/2011 cja
RADSHARE\Section 0 - Forms\Radiation Safety 0.3 thru 0.5\Form 0.44_Survey_Waste_File12.2.doc

RSO Weekly Survey
Concrete Building

Attachment 5



Location of Measurement	Gross mR h^{-1}	Gamma Count CPM	Gamma Count DPM	LSC CPM	LSC DPM
Background	0.006	B			
1. Int. Wall	0.002				
2. Int. Wall	0.002				
3. Int. Wall	0.002				
4. Entire Interior A	0.002				
5.	0.002				
6.	0.002				

Notify the RSO when the restricted area exceeds 2.0 mrem h^{-1}, an unrestricted area exceeds 0.2 mrem h^{-1}, or removable contamination exceeds 200 dpm ·100 cm^{-2} (iodine-131), and document any action taken.

Survey Meter: Minispec S/N: 156

Calibration Date: 02/10/2011 Check-source reading: 154 μRh^{-1}

Gamma Counter: Capintec Captus-600 S/N: 600277

Liquid Scintillation Counter: Malisa Star S/N: 35003

Comments: A Includes all equipment and materials. See attached listing. B See alternate survey form.

Surveyor: C.J. [signature] Date: 06/30/2011

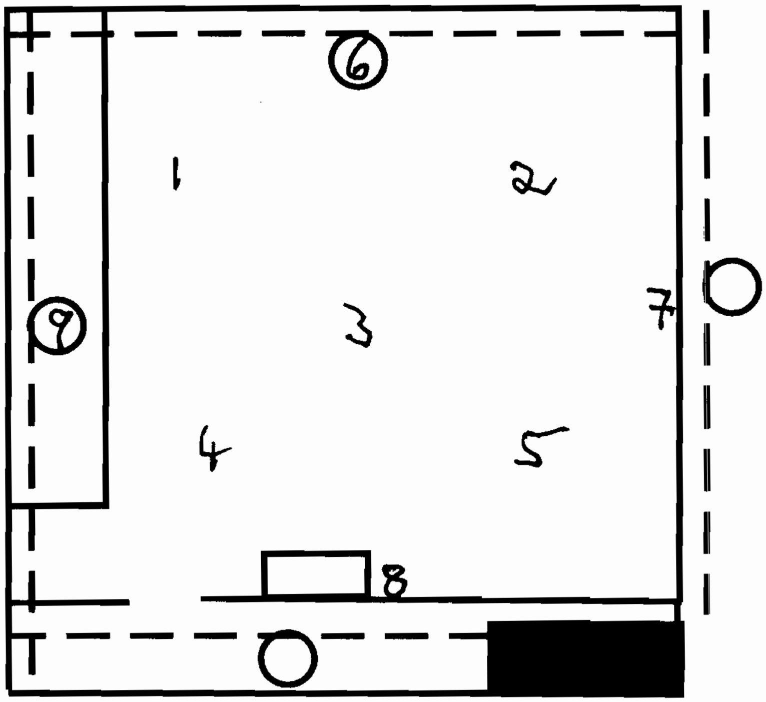
Reviewed by: [signature] Date: 8 July 2011
RSO or designee

GEORGETOWN UNIVERSITY HOSPITAL
RADIATION SAFETY ROUTINE SURVEY REPORT

DATE: 06/30/2011
PAGE: 1 of 1

Attachment 6

Sketch of Laboratory Area (A): Concrete Building



Wipe/ Area ID	Location note	Gross Wipe Test Results (~~NET DPM~~) CPM Ch 1	Beta Ch 2	56-1024	Area ID	Location note	Meter Reading(Gross) CPM	mR/hr
	BACKGROUND			420 / /	//////	BACKGROUND	140	D
////////	ACTION LEVEL			/ /	1		140	
1	Floor			416 406	2		140	
2	Floor			397 /	3		140	
3	Floor			440 /	4		140	
4	Floor			429 /	5		140	
5	Floor			416 /	6		140	
6	Center North Wall			445 /	7		140	
7	Center East Wall			442 /	8		140	
8	Center South Wall			428 /	9		140	
9	Center West Wall			425 /				
				/ /			*	
				/ /				
				/ /				
				/ /				
				/ /				
				/ /				
				/ /				
				/ /	*	ALL OTHER AREAS AND EQUIPMENT SURVEYED WERE LESS THAN OR EQUAL TO ~~TWICE~~ BACKGROUND.		
				/ /				
				/ /	**	ALL OTHER AREAS AND EQUIPMENT SURVEYED WERE LESS THAN OR EQUAL TO 0.1 mR/hr.		
				/ /				

CJA

CJA

Portable Survey Instruments Used: A Count Rate Meter B Dose Rate Meter (~~Ion Chamber~~)
Wipe Analysis Instruments Used: C Gamma Counter ______ Beta Counter (Liquid Scintillation)

Comment: A Ludlum Model 12/44-3 #223403/PR210721 Gamma Scintillometer. B GR 130 M #1569 Wikispec, C Captus 600 #600277. D See RSO Weekly Survey for 06/28, 06/29, 06/30.

60701/00327.08

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GEORGETOWN UNIVERSITY HOSPITAL
RADIATION SAFETY ROUTINE SURVEY REPORT

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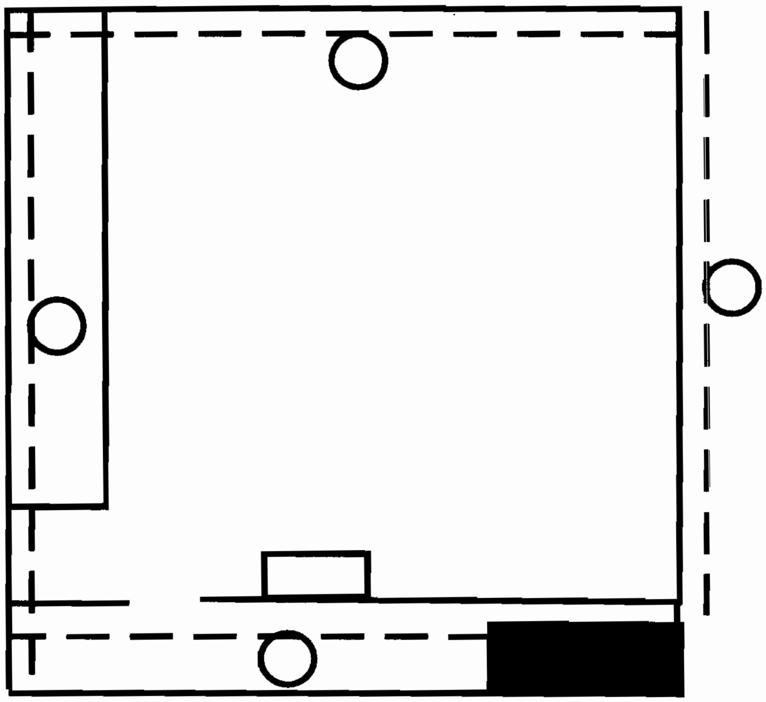
~~Attachment~~ 7

Sketch of Laboratory Area (A): Concrete Building



Wipe/ Area ID	Location note	Wipe Test Results (NET DPM) Ch 1	Beta Ch 2	
	BACKGROUND			/ /
////////	ACTION LEVEL			/ /
				/ /
				/ /
				/ /
				/ /
				/ /
				/ /
				/ /
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Area ID	Location note	Meter Reading(Gross) CPM	mR/hr
///////	BACKGROUND	140	0.006
	Steel Cart	140	
	Brooms (3)	140	
	Dustpans (2)	140	
	60 Gal. Containers	140	
	with Lids (6)	140	
	95 Gal. Totes (4)	140	
	Steel Shelves (15)	140	
	Fiber Drums (2)	140	
	15 Gal. Spill Kit	140	
	Drum	140	
	Steel Pails w/ Lid (2)	140	
	Milk Crate	140	
	Plastic Bottles (3)	140	
	Hand Tools (5)	140	
	Phantoms (6)	140	
	Wooden Box	140	
	Fire Extinguisher	140	

* ALL OTHER AREAS AND EQUIPMENT SURVEYED WERE LESS THAN OR EQUAL TO TWICE BACKGROUND.

** ALL OTHER AREAS AND EQUIPMENT SURVEYED WERE LESS THAN OR EQUAL TO 0.1 mR/hr.

Portable Survey Instruments Used: A Count Rate Meter B Dose Rate Meter (~~Ion Chamber~~)
Wipe Analysis Instruments Used: ______ Gamma Counter ______ Beta Counter (Liquid Scintillation)

Comment: A) Ludlum Model 12/44-3 #22403/PR210721 Gamma Scintillometer.
B) GR-130M #1560 minispec.

GEORGETOWN UNIVERSITY HOSPITAL
RADIATION SAFETY ROUTINE SURVEY REPORT

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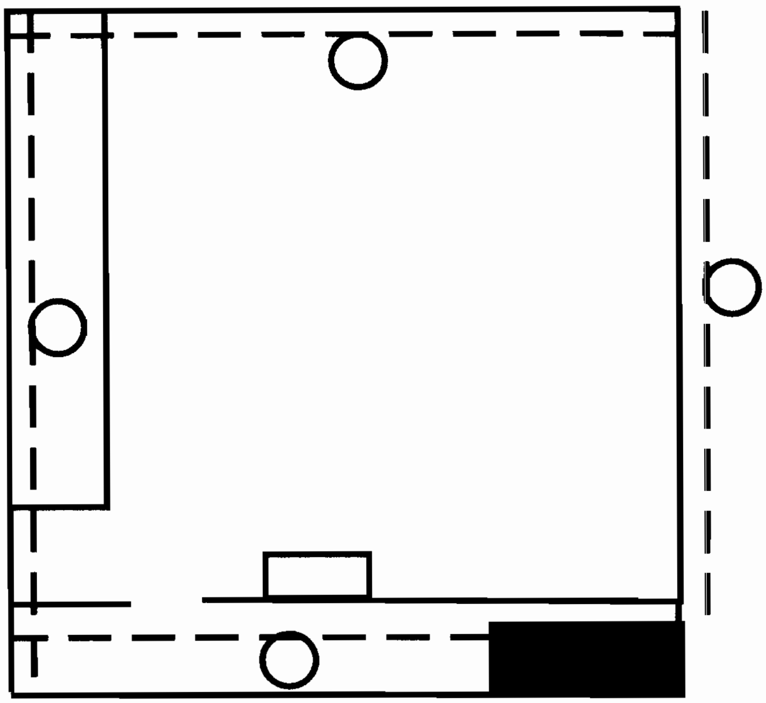
Attachment 8

Sketch of Laboratory Area (A): Concrete Building



Wipe/ Area ID	Location note	Wipe Test Results (NET DPM) Ch 1	Beta Ch 2	
	BACKGROUND			/ /
////////	ACTION LEVEL			/ /
				/ /
				/ /
				/ /
				/ /
				/ /
				/ /
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				/ /

Area ID	Location note	Meter Reading(Gross) CPM	mR/hr
//////	BACKGROUND	140	0.006
	Small Pieces Pb (22)	140	
	Pb Aprons (2)	140	
	Pb Thyroid	140	
	Large Pb Pig	140	
	Pb Lid	140	

* ALL OTHER AREAS AND EQUIPMENT SURVEYED WERE LESS THAN OR EQUAL TO TWICE BACKGROUND.

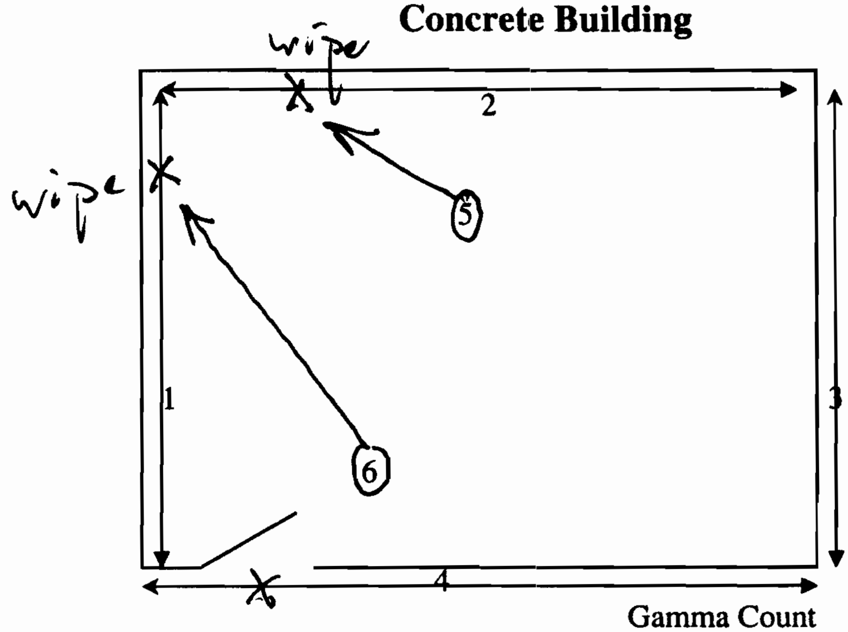
** ALL OTHER AREAS AND EQUIPMENT SURVEYED WERE LESS THAN OR EQUAL TO 0.1 mR/hr.

Portable Survey Instruments Used: A Count Rate Meter B Dose Rate Meter (~~Ion Chamber~~)
Wipe Analysis Instruments Used: ____ Gamma Counter ____ Beta Counter (Liquid Scintillation)

Comment: ____

RSO Weekly Survey
Concrete Building

Attachment 9



Location of Measurement	Gross mR h^{-1}	Gamma Count CPM	Gamma Count DPM	LSC CPM	LSC DPM
Background	0.006	149	—		
1. Interior Wall	0.00	—			
2. Interior Wall	0.00	—			
3. Interior Wall	0.00	—			
4. Interior Wall	0.00 (A)	148	-1		
5. 2nd Shelf from Floor	0.00	157	8 (B)		
6. 3rd Shelf from Floor	0.00	140	-9		

Notify the RSO when the restricted area exceeds 2.0 mrem h^{-1}, an unrestricted area exceeds 0.2 mrem h^{-1}, or removable contamination exceeds 200 dpm ·100 cm^{-2} (iodine-131), and document any action taken.

Survey Meter: Minispec S/N: 1567

Calibration Date: 02/10/2011 Check-source reading: 149 μR h^{-1}

Gamma Counter: Capintec Captus-600 S/N: 600277

Liquid Scintillation Counter: Malisa Star S/N: 35003

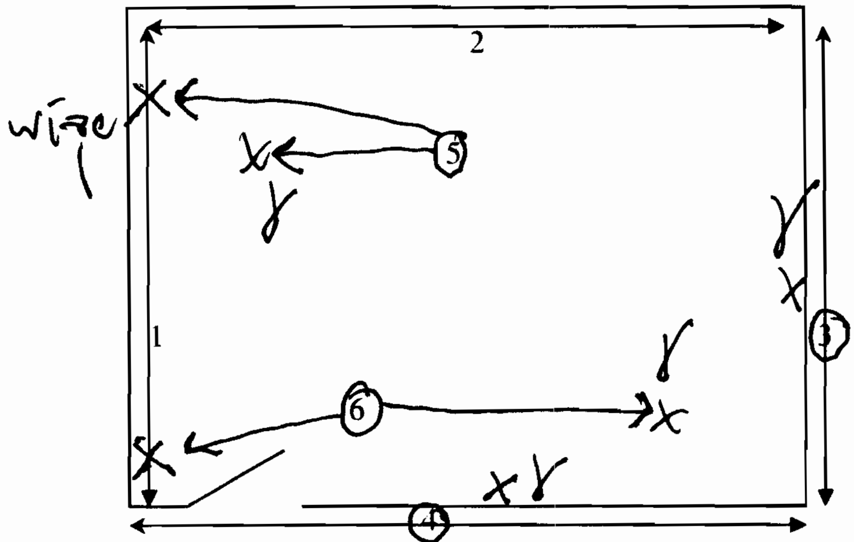
Comments: (A) Wipe taken of sharps box in G1027.
(B) Eff. ≈ 76% Tl-201 (presumed).

Surveyor: C.J. [signature] Date: 07/08/2011

Reviewed by: [signature] Date: 12 July 2011
RSO or designee

RSO Weekly Survey
Concrete Building

Attachment 10



Location of Measurement	Gross mR h^{-1}	Gamma Count CPM	Gamma Count DPM	LSC CPM	LSC DPM
Background	0.005	158	—		
1. Base of Wall	0.006	—			
2. Int. Wall	0.003	—			
3. Int. Wall	0.003	—			
4. Interior Wall	0.003	187	30 (A)		
5. Uppermost Shelf	0.003	167	9 (B)		
6. Cinderblock at Base of Wall	0.003	153	-5		

Notify the RSO when the restricted area exceeds 2.0 mrem h^{-1}, an unrestricted area exceeds 0.2 mrem h^{-1}, or removable contamination exceeds 200 dpm ·100 cm^{-2} (iodine-131), and document any action taken.

Survey Meter: Minispec S/N: 1560

Calibration Date: 02/10/2011 Check-source reading: 159 µR h⁻¹

Gamma Counter: Capintec Captus-600 S/N: 600277

Liquid Scintillation Counter: Malisa Star S/N: 35003

Comments: (A) Wipe #4 taken on Tl-201 sharps box in GL027.
(B) Eff. = 96% Tl-201 (assumed).

Surveyor: C.J. [signature] Date: 07/15/2011

Reviewed by: [signature] Date: 18 July 2011
RSO or designee

Draft Form 0.44 01/04/2011 cja
RADSHARE\Section 0 - Forms\Radiation Safety 0.3 thru 0.5\Form 0.44_Survey_Waste_File12.2.doc

Dade Moeller & Associates®

Analytical & Calibration Services
438 N. Frederick Avenue, Suite 220
Gaithersburg, Maryland 20877
800.871.7930
www.MoellerInc.com
www.RadiationSafetyAcademy.com

August 4, 2011

Dr. Maxwell Amurao
Radiation Safety Officer
Georgetown University Hospital
3800 Reservoir Road
Gorman Building, Suite 2040, Room 2047A
Washington, DC 20007

Re: Request for Additional Information – Reference Control 575679

Dear Dr. Amurao:

This letter is in reference to the Final Status Survey (FSS) Report created by Dade Moeller & Associates, Inc. (Dade Moeller) on July 19, 2011, and the request for additional information from the Nuclear Regulatory Commission (NRC). The format below is question and answer based on the email request for additional information from the NRC.

1) 10 CFR 20.1402 gives the radiological criteria for unrestricted release. One part of the criteria is that the levels have been reduced to levels that are as low as reasonably achievable (ALARA). What activities have been completed to attempt decontamination of the area exhibiting radioactive contamination listed in Table 4 or the "Final Status Survey Report" dated July 19, 2011 to demonstrate ALARA?

    A. To be answered by Georgetown University Hospital (Georgetown).

2) In the "Final Status Survey Report" dated July 19, 2011", It is stated that the areas exhibiting radioactive contamination is assumed to be carbon 14 and is a radionuclide with beta emission only. Based on the provided information, the conclusion is not fully supported. No physical evidence provide confirms nor conflicts with the statement. The collection of samples showed no evidence as presented in the report. No explanation is given by the licensee to why the contamination would be present. Please obtain physical measurements that would show that the contaminate is carbon 14, or further justify why carbon 14 limit should be used.

    A. See Attachment 1 – "Final Status Survey Work Plan." All evidence gathered during the Historical Site Assessment indicates that the primary radiological contaminate will most likely consist of tritium (Hydrogen 3 or H-3) or Carbon 14 (C-14). As stated in the attached Work Plan:

    Examination of historical isotope usage and licensed isotopes, quantities, allowed uses and physical forms support the following conclusions:

    - *The majority of usage involved C-14, a low energy beta particle emitter with a 5,730 year half-life and tritium (H-3), a low energy beta particle emitter with a 12.35 year half-life.*

- *Usage quantities and decay kinetics indicate that any residual contamination remaining in the building will most likely consist of tritium or C-14. These two radionuclides, therefore, constitute the* ***radionuclides of potential concern*** *for this project.*

Evidence gathered onsite as well as the historical data provided support the conclusion reached in the FSS Report that the radiological contaminate was most likely from C-14.

Other nuclides of concern with long lived half-lives include the following:

- H-3 ($T_{1/2}$ – 12.3 years) – Primary mode of decay:
  - Beta – 18.6 keV (max)
  - Nuclide **undetectable** with hand-held instrumentation used.

- Co-57 – ($T_{1/2}$ – 271.8 days) – Primary mode of decay:
  - Gamma – 122 keV (87%)
  - Gamma – 136 keV (11%)
  - Gamma – 14 keV (9%)

- Na-22 – ($T_{1/2}$ – 2.61 years) – Primary mode of decay:
  - Beta – 545 keV (max)
  - Gamma – 1,275 keV (100%)

Given the above list of nuclides, H-3 can be immediately ruled out as the equipment used onsite will not detect this low energy beta emitter; furthermore, no evidence of H-3 was seen with the samples collected onsite.

The decay of Co-57 and Na-22, as indicated above, includes the emission of a various gamma components. Surveys of the areas exhibiting elevated count rates with the scintillator would have yielded an elevated count rate with that probe as well. Additionally, the areas of elevated count rates were only visible upon direct scan of the floor. The count rate was considerably less when the probe (Meter #1) was elevated from the floor at a distance greater than 5 cm. At distances greater than 30 cm (12 inches) the count rate was approximately equal to the background rates listed in Table 2 of the FSS Report. The estimated range of the beta emission in air for C-14 and Na-22 is 24 cm and 1.4 meters, respectively.

Calcium 45 (Ca-45) was questioned as a possible contaminate based on evidence that this nuclide is listed on Georgetown's license and was historically utilized by Georgetown. In speaking with the Dr. Amurao, RSO, the last time Ca-45 was utilized and/or stored within the facility surveyed was around the year 2000. Based on the decay of Ca-45 ($T_{1/2}$ – 162.7 days), approximately 24 half lives have elapsed since the year 2000 ruling out Ca-45 as the possible radiological contaminate.

Together with the information provided in the Historical Site Assessment, decay kinetics, type of radiological emission seen onsite, and the distances that the beta emissions were seen at, Dade Moeller concludes that the radiological contaminates are from C-14.

3) In the "Final Status Survey Report" dated July 19, 2011", a counting efficiency of 8.5% was used for carbon 14. No basis for the efficiency was given. Please provide the basis of the efficiency for the instrument used.

   A. Please find the Calibration Certificate for Meter #1 (S/N 187246) as Attachment 2 to this letter detailing the calibration conditions of the instrument used to collect the final data from each area exhibiting elevated count rates. Additionally, please find the Calibration Certificate for Meter #2 and #3 in Attachment 2.

4) In the "Final Status Survey Report" dated July 19, 2011", scans of the area was performed. However, the rate of speed being used and the minimal detectable activity that could be found using the survey protocol was not provided. Please provide information on scan rates and minimal detectable activity.

   A. Please find Attachment 1 – "Final Status Survey Work Plan." The work plan details all aspects of the surveys plan including those questions listed above.

5) In the "Final Status Survey Report" dated July 19, 2011", it was stated that survey meter 3 was used for floor and lower wall scans. No survey information was provided. Please provide the survey information associated with the low energy gamma scintillator.

   A. No areas of elevated count rate were observed with the scans of the low energy gamma scintillator. Please find the Calibration Certificate for this instrument in Attachment 2 to this report. The results section of the FSS Report indicates "No other areas of radioactive contamination were found during the scans of the building surfaces and shelving units (other than those listed in Table 4 above)." The statement includes the scan with Meters 1-3 included in Table 2 "Radiation Detection Equipment" of the FSS Report.

   Additionally, each area exhibiting elevated count rates was scanned with the low energy gamma scintillator yielded results equal to normal background rates as indicated in Table 3 "QC Meter Information" of the FSS Report. Table 4 "Areas Exhibiting Radioactive Contamination" of the FSS Report has been recreated (below) to include the results of the low energy gamma scintillator for the survey of areas exhibiting elevated count rates.

Table 4 – Areas Exhibiting Radioactive Contamination

Smear # (Location)	Description	Count Rate (max cpm)[1, 2]	dpm/100cm^2	Count Rate (max cpm)[3]	Smear Sample Results (dpm & nuclide)[2]	Comments
28 & 29 S1	Location A – Floor Area - 0.1332 m^2	1,072 / 1,109	12,612 / 13,047	~ Bkgd (1,000-2,000 cpm)	n/a	Picture 1
30 S2	Location B – Floor Area - 0.091 m^2	6,514	76,635	~ Bkgd (1,000-2,000 cpm)	n/a	Picture 2

Smear # (Location)	Description	Count Rate (max cpm)[1,2]	dpm/100cm^2	Count Rate (max cpm)[3]	Smear Sample Results (dpm & nuclide)[2]	Comments
31 S3	Location C – Floor Area (C/D/E Total) - 0.9571 m^2	2,981	35,071	~ Bkgd (1,000-2,000 cpm)	n/a	Picture 2
32 S4	Location D – Floor Area (C/D/E Total) - 0.9571 m^2	2,922	34,376	~ Bkgd (1,000-2,000 cpm)	n/a	Picture 2
33 S5	Location E – Floor Area (C/D/E Total) - 0.9571 m^2	1,576	18,541	~ Bkgd (1,000-2,000 cpm)	n/a	Picture 2

[1] Meter #1 used for all cpm values.
[2] Values corrected for background.
[3] Meter #3 results.

6) In the "Final Status Survey Report" dated July 19, 2011", it did not describe any surveys in additional areas of potential contamination collection. Please provide surveys of the drain, ventilation exhaust area, or areas with previous equipment exhaust.

    A. The building structure consisted of a concrete floor and concrete/cinder block walls. The structure did not include any drains (sink, floor, or similar type drains). The structure did include a small exhaust fan (for room cooling purposes) that was inoperable at the time of survey. Scans with the equipment listed in Table 2 "Radiation Detection Equipment" of the FSS Report were used to scan the accessible areas of the exhaust fan. Additionally, smear number 15 (see Attachment 1 of the FSS Report for location) included the wall and exhaust fan area. No elevated counts were seen with the scan and or sampling of the area adjacent to and including the exhaust fan.

Sincerely,

Mike Jedlicka
Health Physicist
Analytical & Calibration Services
Dade Moeller & Associates, Inc.

Attachment 1

FINAL STATUS SURVEY
WORK PLAN ~ Former Radioactive Waste Storage Facility

Georgetown University Hospital
Washington, D.C.

July 2011

Prepared by:

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I. INTRODUCTION

As described in the proposal dated July 14, 2011, Dade Moeller & Associates (Dade Moeller) has been retained to perform a final status survey (FSS) of a 200 square foot former radioactive waste storage facility at the Georgetown University Hospital (GUH) in preparation for building demolition. The methodology followed in designing, conducting, and evaluating the FSS will be consistent with the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM). This Work Plan incorporates all of the elements defined in the "Scope of Services" section of the Dade Moeller proposal including a Historical Site Assessment (HSA), or review of activities involving radioactive materials, a description of the data collection activities to be conducted in the building, a discussion of data quality objectives (DQOs) established for the project, and the development of derived concentration guideline limits (DCGLs) to be used to evaluate the levels of residual radioactivity found in the building.

II. HISTORICAL SITE ASSESSMENT (HSA)

The former radioactive waste storage facility is a small concrete structure located in Washington, D.C. GUH was licensed by the United States Nuclear Regulatory Commission (NRC) to possess and use radioactive materials to perform clinical therapeutic treatment and diagnostic studies, and to support basic research and development. Several years ago, the license was transferred to Medstar (license number 08-30577-01).

Work done with radioactive materials in posted areas at GUH have been subject to the radiation safety program implemented by the radiation safety officer at GUH for decades. The program has included radioactive waste disposal protocols which included temporary storage of radioactive waste containers in the radioactive waste building. Contamination surveys in the building were included as part of the GUH radiation safety program. Therefore, it is unlikely that significant areas of building surfaces would be contaminated and undiagnosed for any extensive period of time. Historically, any contamination found during a survey should have been responded to immediately. As necessary, radiation workers and/or EHS personnel would have been responsible for performing any decontamination and follow-up survey activities in response to these types of events.

Over the years, several radioisotopes used at GUH became incorporated into radioactive wastes moved to the radioactive waste storage building. These included numerous short-lived radionuclides as well as carbon-14 (C-14 – $T_{1/2}$ = 5,730 years), cobalt-57 (Co-57 – $T_{1/2}$ = 272 days), sodium-22 (Na-22 – $T_{1/2}$ = 2.61 years), and hydrogen-3 (H-3 or tritium – $T_{1/2}$ = 12.3 years) in microcurie to low millicurie quantities.

The RSO at GUH has reported that there have been no documented "incidents" involving radioactive materials, contamination, and/or significant worker dose in the former radioactive waste storage building. Therefore, no historical incidents have any impact on the design for the upcoming final status survey.

Examination of historical isotope usage and licensed isotopes, quantities, allowed uses and physical forms support the following conclusions:

- The majority of usage involved C-14, a low energy beta particle emitter with a 5,730 year half-life and tritium (H-3), a low energy beta particle emitter with a 12.35 year half-life.
- Usage quantities and decay kinetics indicate that any residual contamination remaining in the building will most likely consist of tritium or C-14. These two radionuclides, therefore, constitute the **radionuclides of potential concern** for this project.

The NRC has published screening levels related to 25 mrem per year dose equivalents for several isotopes in NUREG 1556 Volume 11, including C-14 and H-3. The screening level values are provided in Table 1 below.

Table 1 - Surface Radioactivity Screening Levels (dpm/100 cm^2)

Radionuclide	25 mrem per year Screening Value	Source
H-3	1.3 E +07	NUREG 1556 Vol. 11
C-14	3.7 E +06	NUREG 1556 Vol. 11

The dose per level of contamination for these two nuclides is greatest for C-14. Establishing a Derived Concentration Guideline Level (DCGL) based on the assumption that measured beta-

emitting contamination is due to C-14 is most appropriate for the vast majority of survey units to be surveyed.

III. DATA QUALITY OBJECTIVES

The Data Quality Objectives (DQO) process is a tool that may be used to improve the quality of the data collection process by generating data that support defensible decisions. The DQO process addresses study objectives, data collection and limits on decision errors. It is an iterative process, allowing for re-evaluation of preliminary decisions as more information and analytical data are gathered. At the HSA stage of the site survey life cycle, the DQO process has as its goals the following:

- Identify an individual or list of planning team members (and decision-maker)
- Concisely describe the problem(s)
- Initially classify the facility and survey unit(s) as impacted or non-impacted

Planning Team

Dade Moeller has put together a team of professional health physicists to plan, conduct, and evaluate the final status survey of the GUH facility. The individual responsible for decision making is Dr. Maxwell Amurao, Radiation Safety Officer at GUH. Other members of the team include certified health physicist Dr. Alan Fellman, field team leader Mr. Mike Jedlicka, and health physics technician Mr. Craig Harris of Dade Moeller.

Problem Description

For several decades, operations at the facility utilized numerous sources of radioactive materials for medical and research and development studies. The majority of materials were unsealed sources, or liquids, although other forms of radioactive materials were possessed and used for various purposes. The facility operated under a radioactive materials license and was subject to a radiation safety program.

Initial Classification

Dade Moeller personnel have established a preliminary survey unit classification for the radioactive waste storage building of MARSSIM Class II impacted area.

Full implementation of the DQO process is warranted for the purpose of completing a statistically defensible FSS. The DQO process involves a seven-step data life cycle that generates a set of quantitative and qualitative statements pertaining to data collection activities and is based on the methodology described in the MARSSIM document. The multi-step data life cycle is presented below.

State the Problem

The objectives of this step are the following: identify the planning team members and decision maker(s), summarize available resources and relevant deadlines, and formulate a description of the problem. The problem is to demonstrate that the level of radioactive contamination present in the radioactive waste storage facility is compliant with levels suitable for unrestricted use. Mr. Jedlicka, in consultation with Dr. Fellman, will determine sample and measurement locations and will ensure adherence to all relevant quality assurance elements as they relate to radiation measurements. Data will be evaluated in the field as appropriate to guide additional data collection efforts. The FSS data sets will be evaluated via the statistical testing methodology described in MARSSIM to determine the status of each survey unit with regard to suitability for release for unrestricted use.

Identify the Decision

This field investigation, combined with a review of data evaluated during the HSA will provide an answer to the following question: *"Are building areas suitable for unrestricted use? If not, what is the type and extent of contamination?"* The answer to these questions will be provided in the FSS Report.

Identify the Inputs to the Decision

The required data that need to be generated are discussed in detail in various sections throughout this document. In general, the characterization and survey activities will consist of:

- Collection of wipe samples in each room for liquid scintillation and gross gamma counting analysis
- Collection of static measurements of surface alpha and beta levels with a 100 cm^2 gas flow proportional detector
- Scan of floors with a large area floor monitor
- Scan of lower walls with gas proportional detectors
- Exposure rate measurements, if appropriate, with ion chambers and/or microrem meters
- Survey of interior surface of hoods (wipe samples from hood surfaces and ducts)
- Collection of wipe samples in sinks, sink drains, and floor drains

Wipe samples collected to delineate levels of removable surface radioactivity are inherently non-quantitative and therefore not suitable for MARSSIM-type hypothesis testing. Wipe test data are used to qualitatively describe conditions of building surfaces; they are not used for non-parametric statistical tests to determine compliance with release criteria. Samples of sinks and hoods are evaluated with respect to suitable release criteria, but are not subject to statistical testing. The appropriateness of releasing each survey unit is based on static measurements of surface radioactivity.

Define the Boundaries of the Study Area

The study area consists of approximately 200 square feet of floor space. Any sinks, fume hoods, duct work, and other surfaces potentially contaminated are included in the study area.

Develop a Decision Rule

This step requires that (1) a statistical parameter, such as the arithmetic mean or median of a group of measurements, be selected as the parameter of interest, (2) a quantitative action level, such as the instrument response, be established for further investigation, and (3) decision rules be

generated based on the statistical parameters and the quantitative action levels. The objective of the investigation is to estimate the range residual radioactivity in the building and generate sufficient data for final status survey testing. Experience in other similar facilities where licensed radioactive materials were used in medical and research/animal studies and a review of existing data suggest that the primary contributor to potential residual contamination is from surface contamination due to work with beta-emitting radionuclides. As discussed in the HSA, consideration of the suite of radionuclides utilized historically at GUH indicates that only a select few radionuclides (C-14 and H-3) have half-lives of suitable duration combined with recent usage patterns such that they may still be present on surfaces; therefore, total surface radioactivity in units of dpm per 100 cm^2 is the primary parameter of interest. All gross beta measurements will be assumed to be measures of C-14 activity. Analytical data will be generated via liquid scintillation and gross gamma counting to quantify the levels of removable surface activity, if present. The liquid scintillation analyses will be able to distinguish between C-14 and H-3.

Statistical Parameter

Ultimately, nonparametric statistical tests will be utilized if necessary to determine if the level of residual radioactivity in each survey unit exceeds the maximum allowable limit. Since data sets exhibiting a severely skewed (to the right) distribution could result in an arithmetic mean exceeding the limit, MARSSIM recommends using the mean as the statistical parameter of interest.

Investigation Level

The entire 200 square feet of the total building area is considered as Class II impacted area. No areas will initially be established as Class I survey units because a finding of average residual radioactivity throughout a survey unit in excess of the release limit, or $DCGL_w$ is unexpected. The $DCGL_w$ is discussed in section IV of this Work Plan. For a Class II survey unit, an investigation level of one half the $DCGL_w$ will be used. Any static or scanning measurements exceeding this level will be flagged for further investigation.

Decision Rules

The radionuclides used at GUH historically do not appear in background. Therefore based on MARSSIM guidance, the Sign Test was selected as the appropriate test for this FSS. MARSSIM recommends that for situations where the contaminant is not present in background or is present at such a small fraction of the $DCGL_W$ as to be considered insignificant, a background reference area is not necessary. The contaminant levels, should they exist, are compared directly to the $DCGL_W$ value. The Sign Test is considered a one-sample nonparametric statistical test. To implement the decision rules, the FSS must provide sufficient data to allow for a statistically valid estimate of the mean concentration and enough sample data to generate a test with the required statistical power.

Specify Limits on Decision Errors

The results of the survey will be used to evaluate if residual radioactivity in the survey units are compliant with the appropriate release limit. A yes or no decision on suitability for release will be drawn based on an evaluation of the Null Hypothesis. In these final status surveys, the Null Hypothesis tested will be that the residual radioactivity in the survey unit exceeds the release criterion. Therefore, evidence to the contrary is necessary to release the survey unit for unrestricted use. In such a situation, it is possible for two incorrect decisions to be made: (1) deciding the answer is "yes" when the true answer is "no", and (2) deciding the answer is "no" when the true answer is "yes." The approach to the surveys in this Work Plan has been designed to minimize the possibility of decision errors at acceptable levels.

There are two types of decision errors associated with hypothesis testing. A Type I error (false positive) would result in release of a survey unit when the level of residual radioactivity is actually greater than the release criterion. Establishing a low Type I error could result in a rejection of the Null Hypothesis in fewer survey units; it also increases the Type II error. A Type II error (false negative) would result in expending resources to clean up residual contamination when the true level of residual radioactivity is compliant with the release criterion. Establishing a low Type II error will minimize the probability of having to perform additional work in survey units that are in fact compliant with the release criterion. The probability of

making a Type I decision error is called alpha (α) and the probability of making a Type II decision error is called beta (β). Typically, when α increases, β decreases. In some cases, it may be necessary to increase the number of samples or measurements collected in each survey unit in order to minimize α and β to acceptable values. What is clear from the consideration of Type I and Type II errors is that there are trade-offs that must be considered in designing a FSS.

To be protective of worker (and possibly public) health, a value of 0.01 will initially be used for α. Fortunately, as shown in Section IV below, given the anticipated range and standard deviation of surface beta activity in the building, it will be possible to set α and β at 0.05 without requiring an unreasonable number of measurements. If conditions found in the former radioactive waste storage building are not found as assumed, it is possible that a reevaluation might lead to a conclusion that the alpha value should be larger to keep the Type II error and sample size requirements acceptable.

Limits on Analytical Data Decision Errors

Quality control checks on hand-held survey instruments will include measures of battery strength, background response, and response to a check source. More rigorous decision error limits are not warranted for hand-held survey instruments.

There are several types of decision errors associated with analytical data. The data can be biased high (false positive), biased low (false negative), or completely invalid (rejected). The amount of error associated with the data will be minimized through the implementation of methods that produce precise, high-quality data. As part of the data generation process, appropriate quality control measures and samples (e.g., backgrounds, replicate analysis, etc.) will be included. During the decision making process, the bias of the data, if any, will be considered.

Optimize the Design for Collecting Data

This step is used to produce the most resource efficient investigation design that will meet the DQOs. The approach to final status survey data collection incorporates biased scanning of potentially contaminated surfaces and random sampling and measurement locations necessary for performing the required nonparametric statistical tests to evaluate the Null Hypothesis.

IV. MARSSIM IMPLEMENTATION

The building surfaces targeted for final status surveys consist primarily space used to store low level radioactive wastes. The entire building is considered to fall under the category of "Impacted Area" as defined by MARSSIM.

Radionuclides of Concern

Residual contamination with short- and intermediate-lived nuclides is no longer a concern due to radioactive decay. Therefore, the survey measurements and sample analyses will be focused on quantifying the nuclides potentially present on building structural surfaces. Most likely, this would be restricted to C-14. To ensure the most conservative approach to evaluating residual contamination levels, gross beta measurements conducted with survey instruments will assume that all counts are from C-14, and wipe sample activity as determined via liquid scintillation counting will be assumed to be entirely from C-14 and H-3, depending on the spectrum.

Determination of Derived Concentration Guideline Levels (DCGLs)

The surface area concentration of radionuclides of concern equivalent to the release criterion, or regulatory limit expressed as an annual dose, is identified in MARSSIM as the DCGL. Exposure pathway modeling is necessary to relate a concentration of a specific radionuclide to an annual dose. Averaged over a large area, the DCGL is expressed as $DCGL_{W}$. Since it is possible that residual radioactivity will be found in the form of small areas of elevated activity rather than throughout a survey unit, MARSSIM also establishes and evaluates survey data relative to a DCGL for elevated measurement comparisons, or $DCGL_{EMC}$.

In support of license termination, NRC has published screening levels for surface radioactive contamination relating activity level to an annual dose of 25 mrem per year. GUH has determined that compliance with the aforementioned portions of 10 CFR 20 and the 25 mrem per year standard is best attained by utilizing screening levels that are equivalent to doses estimated at **one mrem per year**. The appropriate activity levels equivalent to this minimal annual dose

rate have been determined by dividing the NRC 25 mrem per year screening levels by a factor of 25, maintaining the NRC default assumption of 10 percent removable activity. These values appear in Table 3 below.

Table 3 - $DCGL_W$ Values For Total Surface Radioactivity		
Radionuclide	NRC Screening Value[1] ($dpm/100\ cm^2$)	$DCGL_W$[2] ($dpm/100\ cm^2$)
H-3	1.2 E +08	4.8 E +06
C-14	3.7 E +06	1.5 E +05

[1]From NUREG 1556 Volume 11, Table 11.1; equivalent to 25 mrem per year.
[2]Equivalent to 1 mrem per year.

Historical Co-57 and Na-22 use was minimal. It is highly unlikely that any measured gross beta activity in the building would be due to contamination from either of these radionuclides. Any remaining beta-emitting residual radioactivity on building surfaces is most likely to be from either H-3 or C-14. Therefore, the most restrictive of the values for these radionuclides, **1.5 E +05 dpm/100 cm^2**, will be utilized for this project as the $DCGL_W$ for gross beta activity in all survey units. In addition, gross gamma measurements will be collected to evaluate the presence of unknown gamma-emitting radioncuclides.

Identify Survey Units

MARSSIM methodology includes segmenting impacted areas into survey units based on the probability of finding contamination. The entire 200 square feet of the total building area and the lower walls (up to 2 m above the floor) will be evaluated as a survey unit.

A survey drawing will be prepared and will include the unit dimensions, classification, comprehensive scan completion date, MARSSIM based grid structure, area(s) of contamination (as necessary), sample locations for removable radioactive material contamination, and other information as necessary.

Number and Location of Measurements

To properly implement MARSSIM final status survey nonparametric statistical testing via the Sign Test, a determination must be made of the number of measurements required in each survey unit. Sign p is the probability that a random measurement from the survey unit is less than the $DCGL_W$ when the survey unit median is equal to the lower bound of the gray region (LBGR). Sign p values are provided in Table 5.4 of MARSSIM.

N can be estimated as follows:

$$N = \frac{(Z_{1-\alpha} + Z_{1-\beta})^2}{4(signp - 0.5)^2};$$
$$\alpha, \beta = 0.05; Z_{0.95} = 1.645$$

The shift (Δ), is equal to the $DCGL_W$ - LBGR. For the hand-held gas proportional detectors, assuming the detection efficiency is 0.14 and the source efficiency is 0.25, the count rate related to the 1.5 E +05 dpm $DCGL_W$ is 5,250 cpm. The LBGR may be set at one-half the $DCGL_W$, so it is equal to 2,625 cpm. Assuming a value of 875 cpm for σ results in a relative shift (Δ/σ) of 3.0. From Table 5.4 in MARSSIM, the appropriate value for sign p is approximately 0.998650. Using the equation shown above, N is determined to be 10.9. The value for N is then increased by 20 percent and rounded upwards to obtain the desired power level with the statistical tests used to evaluate compliance with the $DCGL_W$. This results in 14 measurements per survey unit. Since there are no Class 1 survey units, additional evaluation of the measurement number with respect to finding small areas with elevated activity is not warranted.

A Random-Start Triangular Grid Measurement Pattern will be used to determine measurement locations in the Class II survey unit. As per MARSSIM, the number of calculated measurement points, N (N=14), is used to determine the spacing, L, of a systemic triangular grid pattern by:

$$L = \sqrt{\frac{A}{(0.866 \times N)}}$$

The first location is determined via a random number generator and a row of points are identified, parallel to the X-axis, at intervals of L. Survey points are then identified along

parallel rows at distances of 0.866 x L from each adjacent row midway between the points on the preceding row. Locations which are found to be either outside of the survey unit or inaccessible are replaced by locations generated with random numbers.

Minimum Detectable Concentrations (MDC)-Static Measurements

Two important measures of instrument sensitivity must be established to ensure that data quality objectives are met. One is the minimum detectable concentration (MDC) of the static surface beta activity measurements to be used in the Sign tests for compliance with the release level, and the second is the scan MDC attained when surveying walls and floors.

Equation 6.7 in MARSSIM provides the method to calculate the MDC for static measurements when α and β are equal to 0.05:

$$MDC = C \times (3 + 4.65\sqrt{B})$$

Where:
B = number of background counts expected to occur while performing an actual measurement
C = factors to convert from counts to units of concentration

For a 0.5 minute measurement, B is equal to approximately 85 counts (standard background, conditions of background subject to survey conditions and building materials). Given a C-14 detector efficiency of 0.0825 and a source efficiency of 0.25, the 100 cm^2 gas proportional detectors have a MDC approximately equal to **3887 dpm/100 cm^2**, which makes this instrument appropriate for generating data to be utilized to determine compliance with a $DCGL_w$ of 1.5 E+05 dpm/100 cm^2.

Minimum Detectable Concentrations (MDC)-Scanning

Several factors influence the minimum concentration detectable when scanning surfaces for total activity levels. These include the detector characteristics (efficiency and probe area), the scan rate, surveyor characteristics, type of radiation (energy and type), and material being scanned (potential loss of efficiency due to absorption of radiations). The methodology used to estimate the minimum detectable count rate (MDCR) and scan MDC for beta emitters is

described in section 6.7.2.1 of MARSSIM. Equations for each parameter and appropriate

$$MDCR = s_i x \frac{60}{i}$$

$$s_i = d' \sqrt{b_i}$$

$$ScanMDC = \frac{MDCR}{\sqrt{pe_i e_s} \frac{probearea}{100cm^2}}$$

parameter values are:

where:

- s_i = minimum detectable number of net source counts in the interval
- i = observation interval = 2 seconds
- d'= index of sensitivity representing the distance between the means of the background and background plus signal (Table 6.5 MARSSIM, providing 95% correct detection rate or 5% false negative rate and 60% false positive rate[1]) = 1.38
- b_i = number of background counts in the two-second interval = 5.7 and 32.7 for the Model 43-68 hand-held gas proportional detector and Model 43-37 floor monitor, respectively.
- p = surveyor efficiency = 0.5
- e_i = instrument efficiency = 0.0825 (43-68) and 0.09 (43-37)
- e_s = surface efficiency = 0.25

Probe areas:

Gas Proportional = 126 cm^2
Floor Monitor = 584 cm^2

The appropriate MDCR and Scan MDCs are provided in Table 3.

Table 3 – Scan MDC Values

Detector	Background (cpm)	MDCR (net cpm)	Scan Sensitivity (gross cpm)	Scan MDC (dpm/100 cm^2)
43-68	170	98.7	269	**5,371**
43-37	980	236.7	1,218	**2,548**

[1] Note that while scanning, it is preferable for the surveyor to have a liberal willingness to decide that a signal is present based on minimal evidence. The "cost" of the false positive is only a small increment of time incurred while taking a closer look at the area in question with the detector.

As with the MDC for static measurements, the scan MDCs for both detectors provide sufficient sensitivity to investigate survey units with a $DCGL_w$ of 1.5 E+05 dpm/100 cm^2.

V. FIELD INVESTIGATION

The approach to the field activities is based on the findings of the HSA and an understanding of the radiation safety program in place at GUH. The survey will incorporate several types of measurements and sample analyses to fully characterize any residual radioactive contamination and generate the final status survey data necessary to test each survey unit for suitability to be released for unrestricted use. A combination of surface scanning, static measurements, and sample collection and analysis will ensure that all impacted areas in the building are fully evaluated compliant with MARSSIM. The following steps will be completed during the FSS of the former radioactive waste storage building.

Survey Drawings

The survey unit will be drawn in a two-dimensional format, including the floor and all wall areas. It will detail the survey unit dimensions, MARSSIM classification, comprehensive scan completion date, MARSSIM based grid structure detailing the location of static measurements and sample locations for removable radioactive contamination, area(s) of contamination (as necessary), sample locations for removable radioactivity other than grid points, floor and wall areas found to be inaccessible due to fixed building structures should there be any, and other information as necessary.

MARSSIM Grid

Each survey unit will be "gridded" following the techniques and examples laid out in the MARSSIM. Fourteen (14) samples/direct measurements will be completed in each survey unit.

Comprehensive Floor and Wall Scans

A comprehensive scan of not less than 90% of all accessible floor and wall areas up to two (2) meters will be completed with the equipment identified below. In addition, a gamma survey

will be completed on all floor and wall surfaces to eliminate the possibility of contamination from gamma-emitting isotopes. Radiation detection equipment to be used consists of:

- Ludlum Model 2224-1 Alpha/Beta Scaler/Ratemeter with Model 43-68 Gas Proportional Detector.
- Ludlum Model Floor Monitor Model 239-1F with Model 2224 Alpha/Beta Scaler/Ratemeter and Model 43-37 Gas Proportional Detector.
- Ludlum Model 3 Survey Meter with Model 44-3 1" x 1mm NaI Low Energy Gamma Scintillator.

Comprehensive Fixed Item Scans

Additional comprehensive scans will be performed on fixed materials found, as appropriate. Fixed materials are defined as, but not limited to the following:

- Bench and Counter Tops
- Under Bench/Counter Drawers and Cabinets
- Over Bench/Counter Cabinets and Shelves
- Hood and Sink Surfaces

Exposure Rate Measurements

External exposure rate measurements will be taken in any areas found exhibiting elevated surface activity levels. These will be taken with a suitable hand-held instrument such as an ion chamber, pressurized ion chamber, or plastic scintillator (micro rem meter). Exposure rate readings will be documented on the survey unit drawing.

Total Surface Activity Measurements

Compliance with the appropriate DCGL will be evaluated based on a comparison of static total surface measurements (30 second measurement interval) collected with a 100 cm^2 Ludlum Model 43-68 Gas Proportional Detector to the applicable DCGL. A static measurement will be taken at each grid point established in the MARSSIM grid structure for each survey unit.

Removable Surface Activity – Smear Sample Surveys

Smear samples will be collected at each grid point within each survey unit to assess each unit for the presence of removable radioactive contamination. Additional smears will be collected on the other surfaces as appropriate. Smear samples may be taken on bench/counter tops, sink surfaces, and interior hood surfaces, based on past nuclide usage, survey data gathered onsite, and professional judgment.

VI. FINAL STATUS SURVEY REPORT

All data collected during the FSS will be used to complete a FSS Report for presentation by GUH to the NRC. The report will include in detail all aspects of the survey, including methods, instrumentation, survey unit drawings, scan/analytical data/results, and quality control data. The report will include all radiological data generated in each survey unit with an evaluation of the surface activity data with respect to the $DCGL_W$. All data gathered will be used to answer the question, *"Are areas suitable for unrestricted use? If not, what is the type and extent of contamination?"* Recommendations for additional work in any areas found not suitable for release will be included.

VII. ANALYTICAL LABORATORY

All samples collected will be analyzed by Dade Moeller's Radio-Analytical & Calibration Laboratory (RACL) for gross gamma and via liquid scintillation counting for the presence of alpha and beta emitting radionuclides. The RACL follows a rigorous Quality Assurance program that meets and follows the standards provided under ISO/IEC 17025:2005, NELAC Standard, ANSI NQA-1, ANSI N323A, MDE License MD-31-244-01, and other applicable standards. All RACL personnel are trained in key aspects of the quality system.

VIII. HEALTH AND SAFETY

Level D protective clothing will be utilized during the field activities. This includes:

- Work clothes
- Work boots

- Disposable gloves (to be worn during sample collection)

The following radiation detection devices will be utilized:

- Personal dosimeters
- Bicron micro-rem meter, or
- Hand-held ion chambers
- Ratemeter with pancake GM, NaI, and gas flow proportional detectors
- Floor monitor

All pieces of equipment will be checked for contamination with the pancake GM detector prior to removal from restricted areas. Items not indistinguishable from background will be decontaminated. If levels remain elevated, the items in question will be placed in plastic bags and labeled with "Caution, Radioactive Material" tape.

Workers will frisk their hands and feet prior to exiting any restricted area. Any items or pieces of clothing identified as contaminated will be placed in plastic bags labeled with "Caution, Radioactive Material" tape.

August 4, 2011

Attachment 2

Dade Moeller & Associates®

Analytical & Calibration Services
438 N. Frederick Avenue, Suite 220
Gaithersburg, Maryland 20877
800.871.7930
www.MoellerInc.com
www.RadiationSafetyAcademy.com

Certificate of Calibration

Client Name:	Dade Moeller & Associates	**Instrument Make:**	Ludlum
Client Address:	Mike Jedlicka	**Instrument Model:**	2224-1
	438 N. Frederick Ave.	**Instrument S/N:**	187246
	Suite 220		
	Gaithersburg, MD 20877	**Calibration Date:**	**Nov. 29, 2010**
Phone Number:	(301) 990-6006	**Calibration Due Date:**	**Nov. 29, 2011**

CALIBRATION DATA*

Scale	Expected (cpm)	Observed (cpm)	Delta % (CF)	Observed (cpm)**	Delta % (CF)**
x1	100	100	0.00	-	-
x1	250	250	0.00	-	-
x10	1,000	1,000	0.00	-	-
x10	2,500	2,500	0.00	-	-
x100	10,000	10,000	0.00	-	-
x100	25,000	25,000	0.00	-	-
x1000	100,000	100,000	0.00	-	-
x1000	250,000	250,000	0.00	-	-

*Calibration Data using Wm. B. Johnson & Associates Inc. Varipulser; Model VP-CC; Serial Number 1135; Calibrated 2/19/2010.
**Observed (cpm) and Delta % after adjustment

Probe Data

Probe Type	Model	S/N	Geometry	Isotope 1	Eff.	Isotope 2	Eff.	Isotope 3	Eff.
A/B Prop	43-68	PR216244	Contact	C-14	8.5%	Tc-99	32.0%	Tl-204	21.5%
				Sr-90	29.5%	Pu-239	19.0%		

(C-14 1369-68-1) (Tc-99 2220-90) (Tl-204 1369-68-3) (Sr-90 1369-68-2) (Pu-239 1369-60-1)

Instrument Checks

Voltage[1]:	1650V
Voltage[2]:	n/a
Batteries:	OK
Cable:	OK
Sound:	OK
Check Source (Tc-99)	4 kcpm (w/ 1 hr charge)
Comments:	n/a

Calibrated By: Mike Jedlicka, HP

Reviewed By: Alan Fellman, CHP

~ Dade Moeller & Associates, Inc. is licensed by the State of Maryland (MD-31-244-01) to perform instrument calibrations. ~

[1]- Initial voltage reading
[2]- Voltage after adjustment

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Certificate of Calibration

Client Name:	Dade Moeller & Associates	**Instrument Make:**	Ludlum
Client Address:	Mike Jedlicka	**Instrument Model:**	2224-1
	438 N. Frederick Ave.	**Instrument S/N:**	187275
	Suite 220		
	Gaithersburg, MD 20877	**Calibration Date:**	**Jan. 14, 2011**
Phone Number:	(301) 990-6006	**Calibration Due Date:**	**Jan. 14, 2012**

CALIBRATION DATA*

Scale	Expected (cpm)	Observed (cpm)	Delta % (CF)	Observed (cpm)**	Delta % (CF)**
x1	100	100	0.00	-	-
x1	250	250	0.00	-	-
x10	1,000	1,000	0.00	-	-
x10	2,500	2,500	0.00	-	-
x100	10,000	10,000	0.00	-	-
x100	25,000	25,000	0.00	-	-
x1000	100,000	100,000	0.00	-	-
x1000	250,000	250,000	0.00	-	-

*Calibration Data using Wm. B. Johnson & Associates Inc. Varipulser; Model VP-CC; Serial Number 1135; Calibrated 2/19/2010.
**Observed (cpm) and Delta % after adjustment

Probe Data

Probe Type	Model	S/N	Geometry	Isotope 1	Eff.	Isotope 2	Eff.	Isotope 3	Eff.
A/B Prop	43-68	PR285682	Contact	C-14	8.0%	Tc-99	25.5%	Tl-204	14.5%
				Sr-90	17.0%	Pu-239	17.0%		

(C-14 1369-68-1) (Tc-99 2220-90) (Tl-204 1369-68-3) (Sr-90 1369-68-2) (Pu-239 1369-60-1)

Instrument Checks

Voltage[1]:	1550V
Voltage[2]:	n/a
Batteries:	OK
Cable:	OK
Sound:	OK
Check Source (Tc-99)	3.5 kcpm (w/ 1 hr charge)
Comments:	n/a

1- Initial voltage reading
2- Voltage after adjustment

Calibrated By: Mike Jedlicka, HP

Reviewed By: Alan Fellman, CHP

~ Dade Moeller & Associates, Inc. is licensed by the State of Maryland (MD-31-244-01) to perform instrument calibrations. ~

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www.RadiationSafetyAcademy.com

Certificate of Calibration

Client Name:	Dade Moeller & Associates	**Instrument Make:**	Ludlum
Client Address:	RACL	**Instrument Model:**	3
	438 N. Frederick Ave.	**Instrument S/N:**	164026
	Suite 220		
	Gaithersburg, MD 20877	**Calibration Date:**	**Apr. 20, 2011**
Phone Number:	(301) 990-6006	**Calibration Due Date:**	**Apr. 20, 2012**

Calibration Data*

Scale	Expected (cpm)	Observed (cpm)	Delta % (CF)	Observed (cpm)**	Delta % (CF)**
x0.1	100	100	0	-	-
x0.1	250	250	0	-	-
x1	1,000	1,000	0	-	-
x1	2,500	2,500	0	-	-
x10	10,000	10,000	0	-	-
x10	25,000	25,000	0	-	-
x100	100,000	100,000	0	-	-
x100	250,000	250,000	0	-	-

*Calibration Data using Wm. B. Johnson & Associates Inc. Varipulser; Model VP-CC; Serial Number 1135; Calibrated 2/09/2011.
**Observed (cpm) and Delta % after adjustment

Probe Data

Probe Type	Model	S/N	Geometry	Isotope 1	Eff.	Isotope 2	Eff.	Isotope 3	Eff.
				C-14	5.0%	Tc-99	32.0%	Tl-204	33.0%
PanGM	44-9	PR168846	Contact	**Isotope 4**	**Eff.**	**Isotope 5**	**Eff.**	**Isotope 6**	**Eff.**
				Sr-90	37.0%	Sr/Y-90	74.0%	n/a	n/a
Probe Type	**Model**	**S/N**	**Geometry**	**Isotope 1**	**Eff.**	**Isotope 2**	**Eff.**	**Isotope 3**	**Eff.**
NaI	44-3	PR168781	Contact	I-125	23.0%	Cs-137	6.0%	Co-60	1.5%
Probe Type	**Model**	**S/N**	**Geometry**	**Isotope 1**	**Eff.**	**Isotope 2**	**Eff.**	**Isotope 3**	**Eff.**
ZnS	43-2	PR171007	Contact	Pu-239	26.5%	n/a	n/a	n/a	n/a

(C-14-S/N 1369-68-1; 5/1/2009) (Tc-99-S/N 2220-90; 4/13/2006) (Tl-204-S/N 1369-68-3; 5/1/2009) (Sr-90-S/N 1369-68-2; 5/1/2009)
(I-129 - S/N 1369-68-8; 5/1/2009) (Cs-137-S/N 1120-45-3; 7/1/2005) (Co-60-S/N 1369-68-6; 5/1/2009) (Pu-239-S/N 1362-60-1; 5/1/2009)

Instrument Checks

Voltage (initial):	900 V
Voltage (final):	n/a
Batteries:	OK
Cable:	Replaced
Sound:	OK

Check Source: Tl-204 PanGM (45 kcpm)
Check Source: I-129 NaI (28 kcpm)

Calibrated By: Mike Jedlicka, HP

Reviewed By: Sean Austin, CHP

~ Dade Moeller & Associates, Inc. is licensed by the State of Maryland (MD-31-244-01) to perform instrument calibrations. ~
~ The instrument above was calibrated with NIST traceable sources. ~