IN SITU NDA CONFORMATION MEASUREMENTS PERFORMED AT AUXILIARY CHARCOAL BED AND OTHER MAIN CHARCOAL BEDS AFTER URANIUM REMOVAL FROM MOLTEN SALT REACTOR EXPERIMENT ACB AT OAK RIDGE NATIONAL LABORATORY.

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ABSTRACT

The Molten Salt Reactor Experiment (MSRE) site (Figure 1) is located in Tennessee, on the U.S. Department of Energy (DOE) Oak Ridge Reservation (ORR), south of the Oak Ridge National Laboratory main plant across Haw Ridge in Melton Valley. The MSRE was run by ORNL to demonstrate the desirable features of the molten-salt concept in a practical reactor that could be operated safely and reliably. It introduced the idea of a homogeneous reactor using fuel salt media and graphite moderation for power and breeder reactors. The MSRE reactor and associated components are located in cells beneath the floor in the high-bay area of Building 7503.

The reactor was operated from June 1965 to December 1969. When the reactor was shut down, fuel salt was drained from the reactor circuit to two drain tanks. A "clean" salt was then circulated through the reactor as a decontamination measure and drained to a third drain tank. When operations ceased, the fuel and flush salts were allowed to cool and solidify in the drain tanks. At shutdown, the MSRE facility complex was placed in a surveillance and maintenance program.

Beginning in 1987, it was discovered that gaseous uranium (U-233/U-232) hexafluoride (UF_6) had moved throughout the MSRE process systems. The UF_6 had been generated when radiolysis in the fluorine salts caused the individual constituents to dissociate to their component atoms, including free fluorine. Some of the free fluorine combined with uranium fluorides (UF_4) in the salt to produce UF_6. UF_6 is gaseous at slightly above ambient temperatures; thus, periodic heating of the fuel salts (which was intended to remedy the radiolysis problems) and simple diffusion had allowed the UF_6 to move out of the salt and into the process systems of MSRE. One of the systems that UF_6 migrated into due to this process was the off-gas system which is vented to the MSRE main charcoal beds and MSRE auxiliary charcoal bed (ACB).

Recently, the majority of the uranium laden-charcoal material residing within the ACB was safely and successfully removed using the uranium deposit removal system and equipment. After removal a series of NDA measurements was performed to determine the amount of uranium material remaining in the ACB, the amount of uranium material removed from the ACB, and the amount of uranium material remaining in the uranium removal equipment due to removal activities.

INTRODUCTION/BACKGROUND

The MSRE (Figure 1) was an 8 MW reactor that was operated at Oak Ridge National Laboratory (ORNL) from 1965 through 1969 as a demonstration of the technology needed to develop a commercial Molten Salt Breeder Reactor. The reactor used a unique liquid fuel, formed by dissolving UF_4 fuel in a carrier salt composed of a mixture of LiF, BeF, and ZrF. The fuel salt circulated through a reactor vessel, a fuel salt pump, and a primary heat exchanger at temperatures above 600º C (1112º F). In the reactor, the salt was forced through channels of graphite to provide the geometry and moderation necessary for a nuclear chain reaction. Heat was transferred from the fuel salt to the secondary coolant salt in the primary heat exchanger. The coolant salt is similar to the fuel salt, except that it contains only LiF (66%) and BeF (34%). The coolant salt passed from the primary heat exchanger to an air-cooled radiator, a coolant salt pump, and then returned to the primary heat exchanger. Each of the salt loops was provided with drain tanks, located such that the salt could be drained out of either circuit by gravity. A single drain tank was provided for the nonradioactive coolant salt. Two drain tanks were provided for the fuel salt. The fuel salt drain tanks were provided with a system to remove the intense heat generated by radioactive decay immediately after an emergency reactor shutdown and fuel salt drain. A third drain tank connected to the fuel salt loop was provided for storing a batch of flush salt. This batch of salt, similar in composition to the coolant salt, was used to condition the fuel salt loop after it had been exposed to air and to flush the fuel salt loop of residual fuel salt and contaminants before accessing the reactor circuit for maintenance or experimental activities.

The MSRE was originally fueled by adding ~218 kg of uranium, consisting of 30% U-235 and 70% U-238, to the carrier salt. In 1968 this initial charge of uranium was stripped from the salt using a fluoride volatility process wherein the molten salt was sparged with fluorine gas to convert UF_4 to the volatile UF_6 which was subsequently recovered by chemisorption on sodium fluoride (NaF). The reactor was refueled with ~38 kg of uranium, consisting of - 83% U-233 (with ~250 ppm of U-232 at the time of refueling). When reactor operations were terminated in December 1969, the fuel, flush and coolant salts were drained into their respective drain tanks. The fuel salt was divided between the two fuel-salt drain tanks.

Following reactor shutdown, several activities were performed to evaluate the performance of the materials of construction and to improve containment of the stored salts. Prior to reactor operation, it had been discovered that irradiation of the solid salt produced fluorine gas, presumably by providing the energy to release fluorine radicals (or atomic fluorine) from the salt molecules. This atomic fluorine could then combine to form F_2. Experiments indicated that at sufficiently high temperatures the release of radiolytic fluorine did not occur. An annual annealing process was instituted in which the salt was heated to temperatures above 149º C (300º F), but below the melting point, for periods of about two weeks so that radiolytic fluorine would recombine with the lithium or beryllium in the salt. This procedure was carried out through December 1989. After this time annealing procedures were halted in part because discovery of the of Tl-208 a product of the decay of U-232, in the off-gas systems.

Fig. 1 Aerial view of the MSRE facility

In May 1996, a gas sample withdrawn from the drain tank off-gas was found to contain UF_6 near its saturation pressure, smaller but significant quantities of MoF_6, and about 50% fluorine. A review was promptly undertaken to assess the extent of uranium migration, and a deposit of uranium (~2 to 3 Kg) was identified on the ACB (a 6-in. diameter pipe ~40-ft. long filled with activated charcoal) and less than 1 Kg of uranium in four main charcoal beds which were connected to the drain tank off-gas piping. Since the identification of these conditions, a Remediation project has been undertaken to remove the uranium material from the off-gas system; remove the highly contaminated fuel and flush salts from fuel drain tanks and fuel flush tank; and uranium material from the ACB.

In April and May of 2001, MSRE successfully and safely removed the majority of the uranium-laden charcoal material residing in the ACB. Currently, the uranium-laden charcoal is being stored at MSRE under a controlled environment, awaiting final processing and disposal. The uranium material that was removed from ACB had been a source of concern at MSRE under original as-found condition in the ACB.

The U-233 isotope results in a particularly difficult to handle charcoal material because high levels of gamma radiation are associated with it. A small percentage of U-232 is included in U-233 as an impurity. The decay chain of U-232 includes Tl-208, which is a strong gamma radiation emitter (2.6 MeV gamma energy and 100% emission rate). This daughter product builds up rapidly, causing very high exposure rates (approximately 500 R/hour at the side of the ACB where the majority of uranium-laden charcoal is residing). Because of the high exposure rates (as well as fission products in the ACB), much of the MSRE ACB uranium removal activities were done remotely or with long-handled tools.

As mentioned briefly above due to radiolysis of the fuel salt it is estimated that ~2-3 Kg of the uranium had migrated to the MSRE ACB and another ~1 Kg among the other four charcoal beds via the MSRE

off-gas system. The majority of the uranium in the ACB was identified to reside in the top ~14 in. of the bed. To remove the uranium charcoal material from the ACB, an elaborate remote-handled charcoal removal system was designed, fabricated, tested, and installed at MSRE ACB. Using this system,long-handled tooling, and remote equipment, the ACB top was removed. Then the diffuser at the inlet to the charcoal bed was removed to gain access to the uranium-laden charcoal within the ACB. A solidified portion of uranium-laden charcoal was manually fractured, and ~14 in. of the material was vacuumed from the ACB into the charcoal shielded canister.

Subsequent to the removal activities, as part of material removal conformation activities, MSRE has deployed in situ nondestructive assay (NDA) systems (an in situ gamma spectroscopy system and in field exposure rate measurement instruments) and performed a series of measurements from the uranium laden charcoal material and variety of other material removed from the ACB. Additional measurements were performed on the remaining material in the ACB as well as other main charcoal beds. In addition, NDA measurement of the charcoal canister in its shielded container was performed to estimate the amount of material removed from the ACB.

Configuration of the Auxiliary Charcoal Bed

A schematic of the configuration of the ACB and maintenance shield as it was for this series of NDA measurements it is shown in Figure 2. As noted above, the top of the ACB had been cut off, approximately 14" of highly contaminated charcoal had been removed, a lead cap had been placed on top of the ACB, and the cutting tool, cutter drive unit, and central ball port had been removed (charcoal removal equipment). This left an 8-in. diameter hole in the maintenance shield which provided a relatively obstruction-free viewing path for NDA measurements between the top of the maintenance shield and the ACB. This configuration results in a strong beam of gamma rays from the ACB through the hole in the maintenance shield.

There were three heavily contaminated items present in the Charcoal Bed Cell (CBC) enclosure that posed potential gamma radiation interferences to the NDA measurements. These were the cyclone separator, radon filter, and HEPA filter. These were located in the enclosure that covered the CBC approximately five, ten, and eleven feet, respectively, from the main access port above the ACB.

Gamma spectroscopy measurements of the ACB were performed at varying heights above the maintenance shield. Due to the potential interferences described above, gamma dose rates were measured on a three-dimensional grid with 1-ft grid spacing on top of the maintenance shield directly above the ACB to properly determine the detector locations for these measurements.

Figure 3 shows a picture of the HPGe detector suspended vertically above the maintenance shield. The detector is the ~2ft-long cylinder near the technician's upper hand. The detector was supported and positioned with a crane. The shiny reflective surface seen in the photo is the lead shielding around the germanium detector. The dull gray cylinder above the detector is the liquid nitrogen Dewar. A 30-degree lead collimator was used on the front face of the detector to narrow the measured field of view. An umbilical cable runs from the detector to an amplifier/data acquisition module and laptop computer.

Gamma spectra were acquired for 60 – 300 seconds (live time) at multiple positions above some of the grid points. Due to the high gamma fluxes involved, the detector deadtimes were high, ranging from 6% up to 93% directly above the ACB gamma beam.

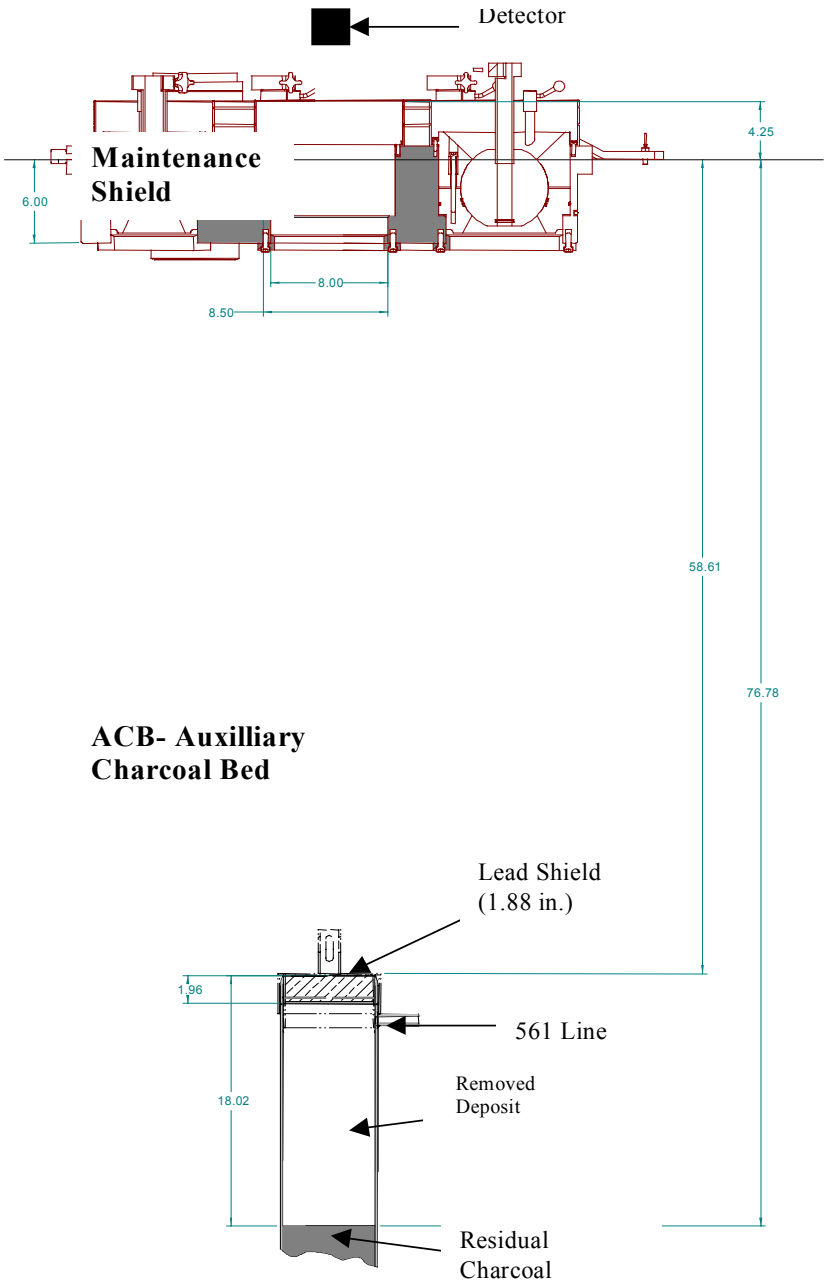

Fig. 2. Configuration of the ACB during May 16-17 NDA measurements.

Fig. 3. Photo of HPGe detector suspended above the maintenance shield portal.

Estimate of the Depth of Moderately Contaminated Charcoal

Based on vertical dose rate profiles along the side of the ACB (previously measured) it was inferred that the charcoal at the top 10-12 inches of the ACB is most highly contaminated and that the degree of contamination decreases the further down the column you go. Previous analyses of gamma spectroscopy measurements support this conclusion. Prior to the current series of NDA measurements the top ~14 in. of the most highly contaminated charcoal was removed and placed in a shielded canister. The distribution of uranium in the remaining charcoal is probably not uniform but it is likely that the top of the column is more contaminated than lower segments.

A series of the gamma spectra was acquired and the gamma flux in the ACB beam at six different heights above the maintenance shield was measured. The variation of the gamma flux with height will have some dependence on the depth of uranium loading within the ACB. Several different (uniform) uranium loading depths were modeled with NDA system (ISOCS) in an effort to derive some conclusions about the actual depth of significant uranium loading in the ACB. Comparisons of these modeling lead to the

conclusion that the uranium loading depth is greater than one inch. This data cannot distinguish between uranium loading depths of 2, 3, and 6 inches.

This measurement is not very sensitive to uranium loading depth because
1) the 2.6MeV gamma rays are highly penetrating and adding a few inches of charcoal does not change the gamma attenuation very much and
2) the change in the source dimension is small compared to the overall source-detector distance.

Geometrical Model and Assumptions

The geometry modeled by the ISOCS analysis software is shown in Figure 2. Residual activity in the 561 line was not modeled. The model makes the following assumptions:

A1) The source term is a segment of uniformly contaminated charcoal ("moderately contaminated charcoal" in the figure) plus a thin layer of debris on top of the lead cap
A2) The charcoal material has a radius of 3.18 in.
A3) The charcoal density is 1.1 g/cc.
A4) The contaminated charcoal is a mix of carbon with UF_4 1:1 by weight, corresponding to a hypothetical compound of $(UF_4)C_{24}$.
A5) The stainless steel canister wall thickness is 0.134 in.
A6) The ACB is capped with a lead cap consisting of 1.88 in. of lead and 0.25 in. of stainless steel.
A7) The distance from the top of the lead cap to the bottom of the maintenance shield is 52 in.
A8) The maintenance shield is 6 in. of stainless steel.
A9) The top surface of the rotating shield portion of the maintenance shield, 1 in. above the 4 ft. x 4 ft. steel plate, is the reference point for height measurements above the maintenance shield.
A10) The diameter of the portal to the ACB beam is 8 in.
A11) There are no other significant obstructions or gamma absorbers between the charcoal bed and the HPGe detector.

In addition, ISOCS accounts for
A12) The variable detector height above the maintenance shield
A13) The attenuation of the 30degree, 25mm or 50mm-thick lead collimator
A14) The intrinsic detection efficiency of the HPGe detector:

Also, the InSpector amplifier accounts for
A15) Deadtime losses up to 50%.
This series of gamma spectroscopy measurements directly measured gamma rays from Tl-208, Bi-212, and Cs-137. Estimates for U-232, total U, and U-233 were derived from the activity estimates of these isotopes. Tl-208 estimates were used for these derivations, rather than Bi-212 estimates, because the counting statistics were best for this isotope. The derived values are based on the assumptions of:

A16) The Tl-208 is in equilibrium with Bi-212 and the Bi→Tl branching ratio is 35.9%.
A17) The estimated Bi-212 is in equilibrium with U-232 and the ratio of Bi-212 to U-232 activity is 1.03:1.
A18) The U-232 specific activity is 21.41 Ci/g.
A19) The ratio of U-232 (grams) to total U (grams) is 155ppm
A20) The weight percent of U-233 to total U is 83.9%

Subtraction of Background and other Interferences

As discussed in the introduction, the measurement area within the Charcoal Bed Cell enclosure does not constitute a low gamma background environment. The gamma dose rates above the maintenance shield (and not in the ACB beam) range from 0.8 to 8 mrem/hour. These dose rates are presumably due to a combination of gamma fluxes from sources such as:

- the cyclone separator
- the radon filter
- the HEPA filter
- other misc. contaminated lines, hoses, and equipment
- leakage radiation from the ACB through the maintenance shield
- leakage radiation from other charcoal beds through the maintenance shield

Technically the debris and contaminated hoses above the ACB, but inside the maintenance shield box, are also a source of background interference but these have been treated as a separate source term.

For these measurements all these iterferences sources were properly considered in the calculations as best possible to determine the residual uranium left in the ACB.

The uncertainty in accurately accounting for background gamma counts that contributes to total uncertainty is in order of ±25% to the reported estimates of activity in the ACB.

Summary of NDA Results for ACB

Tables I summarize the results of the NDA measurements and analyses on the post-removal ACB bed for various material depth. The U-233 uncertainty in these tables represents the total uncertainty in these measurements.

Table I: Summary of U-233 Estimates and Total Uncertainties for the Post-Removal ACB

U loading depth assumed	U-233 (grams)	Total Estimated Uncertainty (grams, see Appendix 1)
1 inch	444	136
2 inches	470	145
3 inches	496	154
6 inches	605	187

It was concludes that, based on the best available information to date, the material remaining in the ACB is in the range of 350 – 550 grams of U-233. This series of in-situ NDA measurements cannot explicitly determine the uranium loading depth in the post-removal ACB. A 2-inch modeled depth distribution results in a slightly better fit to the gamma count rate vs. height data than does a 1-inch depth distribution. Deeper depth distributions (3 inches, 6 inches) do not result in any better fit to the experimental data than the 2-inch distribution.

Charcoal canister and Cask (shield container) Description

The sample consists of contaminated charcoal removed from the Auxiliary Charcoal Bed at MSRE. The charcoal is contained in a stainless steel canister (4.25 in.OD, 0.120 in. wall thickness, 48 in. tall) which is further contained within a 2 in.-thick carbon steel container. The cask consists of 21.75 in. of concrete with an outer layer of 0.25 in.-thick carbon steel.

A schematic of the sample, containers, and cask is shown in Figure 4. This is the geometry that has been modeled with ISOCS software to estimate the attenuation of the steel and concrete layers between the sample and the HPGe detector, as well as gamma self-absorption in the sample matrix and the attenuation of the lead detector collimator.

Fill Height Study

In order to estimate the fill height of charcoal material within the stainless steel canister, a series of HPGe measurements were conducted right at the surface of the cask. The reference point for detector height measurements was the steel floor. Starting at 9 inches from the floor, HPGe spectra were collected at 3 in. intervals up to a height of 69 in. The collection times ranged from two to five minutes.

ISOCS was used to model the variation of detector collection efficiency versus detector height for different assumed fill heights. Based on schematics provided to Canberra it was assumed that the bottom of the charcoal sample was 9.875 in. above the floor. A fill height was assumed and then the detector efficiency was calculated for a series of detector heights up to 69 in. above the floor. Relative detector efficiency vs. height was then plotted versus height and the data was compared with the measured detector response. The process was repeated for several different fill heights until an optimum agreement between the ISOCS calculation and the measured detector response was obtained. The fill height that best fits the measured data was between 20 in. and 25 in. A comparison of the measured detector response and the optimum fit with an assumed fill height of 22 in. and therefore best estimate of the uranium-bearing charcoal fill height in the stainless steel canister was determined to be 22 ± 1 inches.

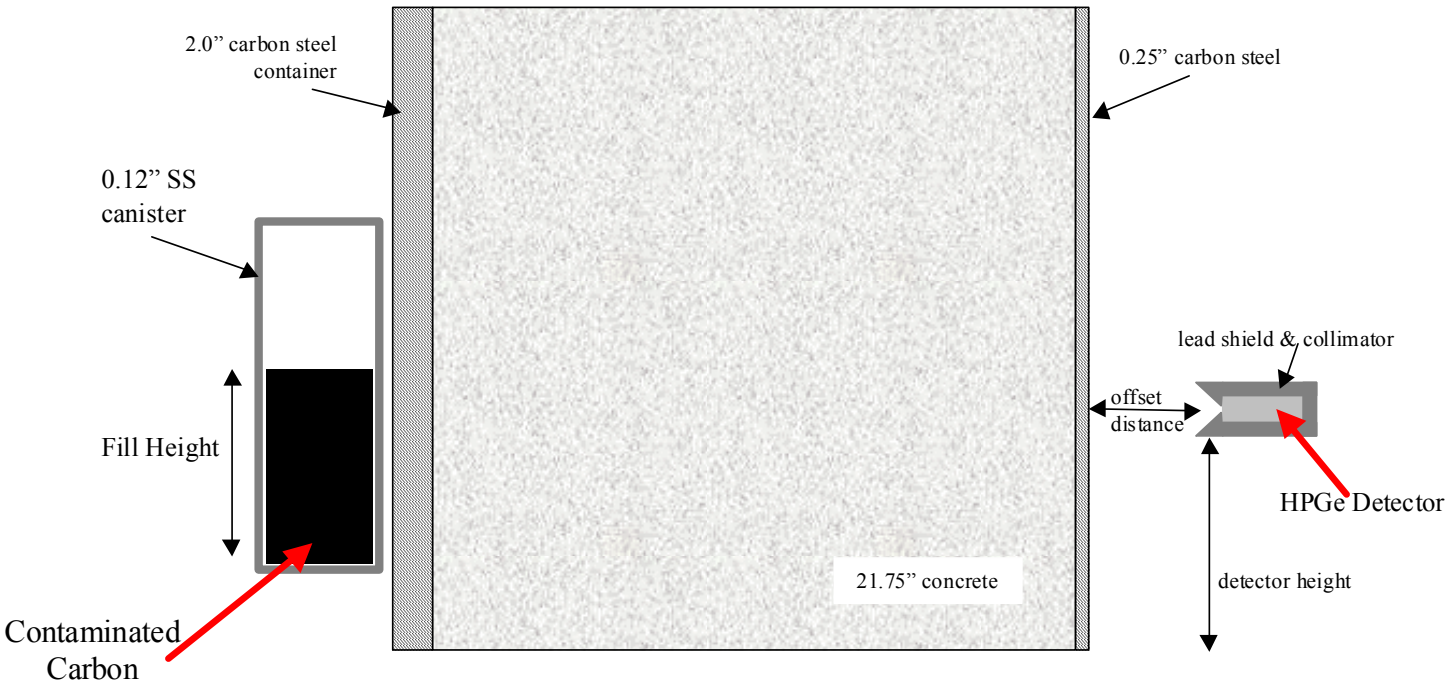

Fig. 4. Side cross-section view of the sample, canister, concrete cask, and HPGe detector.

Material Estimates of Cask Contents

Five different HPGe measurements were analyzed to estimate the activity of gamma-emitting radioisotopes inside the cask. These were measurements collected at the cask surface, 29 in. from the cask surface, and 69 in. from the cask surface. If the separate analyses of these five different measurements are consistent it is a good indication that the ISOCS model is accurate.

The ISOCS model uses the following assumptions:
1) The charcoal material has a radius of 2.13 in.
2) The charcoal density is 1.4 g/cc.
3) The contaminated charcoal is a mix of carbon with UF_4 1:1 by weight, corresponding to a hypothetical compound of $(UF_4)C_{24}$.
4) The stainless steel canister wall thickness is 0.12 in.
5) There is a 0.75 in. air gap between the stainless steel and carbon steel containers.
6) There is a 6 in. OD stainless steel liner, 16 gauge (0.062 in.) thick.
7) The carbon steel container wall thickness is 2 in.
8) The concrete cask wall thickness is 21.75 in. with a density of 2.31 g/cc.
9) The cask has an outer wall of 0.25 in. carbon steel.

For each of the three measurements the ISOCS model also accounts for:
10) The detector height above the floor
11) The detector offset distance from the external cask surface
12) The attenuation of the 30degree/50mm-thick lead collimator
13) The intrinsic detection efficiency of the HPGe detector
14) A fill height of 22 in.

A background spectrum was also acquired with the detector 90 in. from the cask and pointed away from the cask. This spectrum was subtracted from each of the cask measurements before analysis.

Although 911 peaks were observed in each of the five spectra only the 2.6MeV peak from Tl-208 was large enough to quantify.

To estimate the activities of U-232 and U-233 in the cask the following assumptions were made:

15) The Tl-208 is in equilibrium with Bi-212 and the Bi→Tl branching ratio is 35.9%.
16) The estimated Bi-212 is in equilibrium with U-232 and the ratio of Bi-212 to U-232 activity is 1.03:1.
17) The U-232 specific activity is 21.41 Ci/g.
18) The ratio of U-232 (grams) to total U (grams) is 155ppm.
19) The weight percent of U-233 to total U is 83.9%.

The derived estimates for U-232 and U-233 activities, based on the average measured Tl-208 value, are summarized in Table II. These assumptions lead to an estimated loading of U-233 in the cask of 2687 ± 707 grams. This uncertainty of ±26.3% includes the combined estimated uncertainties of all significant contributing factors.

Table II: Summary of Derived Activities for U-232, Total U, and U-233.

DERIVED VALUES		Assumptions
Tl-208 (Ci)	3.93	Average measured value
Bi-212 (Ci)	10.95	35.9% branching ratio
U-232 Ci	10.63	103%
U-232 grams	0.496	21.41 Ci/g
total U grams	3203	155 ppm
U-233 grams	2687	83.90% enriched
Total Estimated Uncertainty, $^{-233}$U (grams, 1 sigma)	707	

Charcoal Bed Cell Past and Present Measurements

Past and present measurements performed to estimate the amount of U-233 and total uranium within the MSRE charcoal beds, including the ACB, and miscellaneous UDR equipment are summarized in Table III. This table briefly summarizes and describes the types of measurements performed, the dates they were performed, the results of measurements (i.e., amount of U-233 and total uranium).

Table III. Summary of the Past and Present CBC NDA Measurement

Measurement location	**Type of Measurements**	**Measurement Date**	**Results of U-233 / Total U in in grams**	**Uncertainty**
MSRE ACB	♦ Portable Gamma-Spectroscopy system using Tl-208 peak, ♦ Along with TLD measurements for Rad profile. ♦ Source length 12+/- 2 inches. ♦ MSRE Uranium assumed to 83.8 % U-233 ♦ And U-232 is 161 ppm of U-232	September 1994	2200 / 2600	+/- 4%
MSRE ACB	Thermal Measurements of the ♦ ACB centerline, ♦ ACB side walls, ♦ and ACB cell ambient air ♦ Source length 12+/- 2 inches. ♦ MSRE Uranium assumed to 83.8 % U-233 ♦ And U-232 is 161 ppm of U-232	March 1995	2130 / 2500	Not Reported
MSRE ACB	♦ Portable Gamma-Spectroscopy (NOMAD) system using Tl-208 peak, ♦ Along with TLD measurements for Rad profile.	November 1998	1270 Average value for U-233/ Not Reported but if 83.8% used 1500 would be	Not reported, but a Range of 1230 to 1310 grams U-233 based

	♦ Source length ~10.5 inches. ♦ MSRE Uranium assumed to 83.8 % U-233 ♦ And U-232 is 161 ppm of U-232		the total U	on various gamma energy lines were reported
MSRE CB 1A	♦ Portable Gamma-Spectroscopy (NOMAD) system using Tl-208 peak, ♦ Along with TLD measurements for Rad profile. ♦ Source length ~35 inches. ♦ MSRE Uranium assumed to 83.8 % U-233 ♦ And U-232 is 161 ppm of U-232	November 1998	112 Average value for U-233/ Not Reported but if 83.8% used 134 would be the total U	Not reported, but a Range of 105 to 127 grams U-233 based on various gamma energy lines were reported
MSRE CB 1B	♦ Portable Gamma-Spectroscopy (NOMAD) system using Tl-208 peak, ♦ Along with TLD measurements for Rad profile. ♦ Source length ~25 inches. ♦ MSRE Uranium assumed to 83.8 % U-233 ♦ And U-232 is 161 ppm of U-232	November 1998	48 Average value for U-233/ Not Reported but if 83.8% used 57 would be the total U	Not reported, but a Range of 44 to 51 grams U-233 based on various gamma energy lines were reported
MSRE CB 2A	♦ Portable Gamma-Spectroscopy (NOMAD) system using Tl-208 peak, ♦ Along with TLD measurements for Rad profile. ♦ Source length ~42 inches. ♦ MSRE Uranium assumed to 83.8 % U-233 ♦ And U-232 is 161 ppm of U-232	November 1998	3 Average value for U-233/ Not Reported but if 83.8% used 4 would be the total U	Not reported, but a Range of <1 to 14 grams U-233 based on various gamma energy lines were reported
MSRE CB 2B	♦ Portable Gamma-Spectroscopy (NOMAD) system using Tl-208 peak, ♦ Along with TLD measurements for Rad profile. ♦ Source length ~38 inches. ♦ MSRE Uranium assumed to 83.8 % U-233 ♦ And U-232 is 161 ppm of U-232	November 1998	168 Average value for U-233/ Not Reported but if 83.8% used 200 would be the total U	Not reported, but a Range of 162 to 178 grams U-233 based on various gamma energy lines were reported
MSRE ACB	♦ Portable Gamma-Spectroscopy (ISOCS) system with Canberra Genie2K analyzing system, using Tl-208 peaks, ♦ Source length ~2 to 3 inches, best estimate. ♦ MSRE Uranium assumed to 83.9 % U-233 ♦ And U-232 is 155 ppm of U-232	June 2001	450 U-233, best estimate average value with a range of 350-550 best estimate / 537 total U for best estimate average value with a range of 317-656 total U	+/- 31%

MSRE UDR Collection canister /concrete shielded container	♦ Portable Gamma-Spectroscopy (ISOCS) system with Canberra Genie2K analyzing system, using Tl-208 peaks, ♦ Source length ~22 +/- 1 inches, best estimate. ♦ MSRE Uranium assumed to 83.9 % U-233 ♦ And U-232 is 155 ppm of U-232	June 2001	2687 U-233 best estimate / 3203 total U best estimate	+/- 26 %
MSRE CB 1A	♦ Portable Gamma-Spectroscopy (ISOCS) system with Canberra Genie2K analyzing system, using Tl-208 peaks, ♦ Source length ~60 inches, best estimate. ♦ MSRE Uranium assumed to 83.9 % U-233 ♦ And U-232 is 155 ppm of U-232	June 2001	293 U-233 best estimate / 349 total U best estimate	+/- 30 %
MSRE CB 1B	♦ Portable Gamma-Spectroscopy (ISOCS) system with Canberra Genie2K analyzing system, using Tl-208 peaks, ♦ Source length ~36 inches, best estimate. ♦ MSRE Uranium assumed to 83.9 % U-233 ♦ And U-232 is 155 ppm of U-232	June 2001	110 U-233 best estimate / 131 total U best estimate	+/- 30 %
MSRE CB 2A	♦ Best estimate is based on the comparison of the 1A and 1B 2001 measurements to those measured on November of 1998. Them the ratio was applied to this bed to estimate the amount of fuel present in this bed. ♦ MSRE Uranium assumed to 83.9 % U-233 ♦ And U-232 is 155 ppm of U-232	June 2001	7 U-233 best estimate / 8 total U best estimate	+/- 30 %
MSRE CB 2B	♦ Best estimate is based on the comparison of the 1A and 1B 2001 measurements to those measured on November of 1998. Them the ratio was applied to this bed to estimate the amount of fuel present in this bed. ♦ MSRE Uranium assumed to 83.9 % U-233 ♦ And U-232 is 155 ppm of U-232	June 2001	437 U-233 best estimate / 521 total U best estimate	+/- 30 %
MSRE UDR Cyclone Separator	♦ Portable Gamma-Spectroscopy (ISOCS) system with Canberra Genie2K analyzing system, using Tl-208 peaks, ♦ MSRE Uranium assumed to 83.9 % U-233 ♦ And U-232 is 155 ppm of U-232	June 2001	30 U-233 best estimate / 36 total U best estimate	+/- 30 %

MSRE UDR HEPA	♦ Portable Gamma-Spectroscopy (ISOCS) system with Canberra Genie2K analyzing system, using Tl-208 peaks, ♦ MSRE Uranium assumed to 83.9 % U-233 ♦ And U-232 is 155 ppm of U-232	June 2001	3 U-233 best estimate / 4 total U best estimate	+/- 30 %
MSRE UDR Rn Filter (adsorber)	♦ Portable Gamma-Spectroscopy (ISOCS) system with Canberra Genie2K analyzing system, using Tl-208 peaks, ♦ MSRE Uranium assumed to 83.9 % U-233 ♦ And U-232 is 155 ppm of U-232	June 2001	0 U-233 best estimate / 0 total U best estimate, The signal was all due Rn-220 gas	N/A
MSRE UDR Turbo Pig	♦ Portable Gamma-Spectroscopy (ISOCS) system with Canberra Genie2K analyzing system, using Tl-208 peaks, ♦ MSRE Uranium assumed to 83.9 % U-233 ♦ And U-232 is 155 ppm of U-232	June 2001	81 U-233 best estimate / 97 total U best estimate	+/- 30 %
MSRE UDR Misc. Tools in B-25 Box	♦ Portable Gamma-Spectroscopy (ISOCS) system with Canberra Genie2K analyzing system, using Tl-208 peaks, ♦ MSRE Uranium assumed to 83.9 % U-233 ♦ And U-232 is 155 ppm of U-232	June 2001	~ < 1 U-233 best estimate / ~ < 1 total U best estimate	+/- 30 %