ALARA CONTROLS AND THE RADIOLOGICAL LESSONS LEARNED DURING THE URANIUM FUEL REMOVAL PROJECT AT THE MOLTEN SALT REACTOR EXPERIMENT PROJECT

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ABSTRACT

The removal of uranium-233 (^{233}U) from the auxiliary charcoal bed (ACB) of the Molten Salt Reactor Experiment (MSRE), performed from January through May 2001, created both unique radiological challenges and widely-applicable lessons learned. In addition to the criticality concerns and alpha contamination, ^{233}U has an associated intense gamma photon from the co-contaminant uranium-232 (^{232}U) decaying to thallium-208 (^{208}Tl). Therefore, rigorous contamination controls and significant shielding were implemented. Extensive, timed mock-up training was also imperative to minimize individual and collective personnel exposures. Back-up shielding and containment techniques (that had been previously developed for defense in depth) were used successfully to control significant, changed conditions. Additional controls were placed on tests and on recovery designs to assure a higher level of safety throughout the removal operations.

This paper delineates the manner in which each difficulty was solved, while relating the relevance of the results and the methodology to other projects with high dose-rate, highly-contaminated ionizing radiation hazards. Because of the distinctive features of and current interest in molten salt technology, a brief overview is provided. Also presented is the detailed, practical application of radiological controls integrated into, rather than added after, each evolution of the project—thus demonstrating the broad-based benefits of radiological engineering and ALARA reviews. The resolution of the serious contamination-control problems caused by unexpected uranium hexafluoride (UF_6) gaseous diffusion is also explicated. Several tables and figures document the preparations, equipment and operations. A comparison of the pre-job dose calculations for the various functions of the uranium deposit removal (UDR) and the post-job dose-rate data are included in the conclusion.

INTRODUCTION – AN OVERVIEW OF THE MOLTEN SALT REACTOR EXPERIMENT

Origin

The Molten Salt Reactor Experiment (MSRE) is unique in that it is both fueled and cooled with uranium dissolved in a molten inorganic salt, primarily lithium fluoride (LiF). Small quantities of berylium fluoride (BeF_2) and zirconium fluoride (ZrF_4) salt were added to capture moisture and to form a stable 3-component eutectic mixture. The concept was originally tested on a small scale as part of the Aircraft Nuclear Propulsion (ANP) program in the late 1940's and early 1950's at the Oak Ridge National Laboratories (ORNL). The original Aircraft Reactor Experiment (ARE) was redesigned as a commercial test reactor with highly-enriched uranium-235 (^{235}U) (i.e., uranium enriched in the ^{235}U isotope) dissolved in the salt. Construction of the reactor building, located in Melton Valley, began in 1954, and operation commenced in 1964. Some of most eminent physicists/nuclear scientists of the twentieth century, including Nobel Laureate Drs. Glen Seaborg and Ed Bettis, Dr. Alvin Weinberg, and Dr. Dick Engel, collaborated on the MSRE project.

Advantages

There were many advantages to the Molten Salt Reactor, such as the following:

- Construction was inexpensive.
- Because freeze plugs were used, no valves were required.
- Any leak was self-healing.
- The coolant was light-weight and flowed like water at operating temperature (>840º F).
- The salt was not water reactive.
- The salt had excellent heat removal properties.
- Fuel could be removed from the matrix as UF_6, and the ^{208}Tl dose-rate remained in the salt.
- There were no fuel elements to be fabricated, handled or stored.
- The design allowed "on-line" refueling.
- The ease of removing a side stream allowed reprocessing.
- No pressure was required on the system, other than the pump-head pressure of a few pounds per square inch (psi) (30 to 40 psi); therefore, neutron integrity was maintained by quick removal of fission product poisoning gases (primarily xenon-137).
- The associated dose rate from ^{208}Tl is so high that security is not a concern, because it is self-protecting.

Shut-down and Subsequent Maintenance

The reactor operated successfully with ^{235}U fuel for 6000 full-power hours and was shut down in 1967. Subsequently, the ^{235}U was removed by diffusing UF_6 gas through the fuel matrix, which was then extracted with sodium fluoride (NaF) absorbers. On October 8, 1968, the reactor was powered for the first time using ^{233}U as the fuel. (See Figure 1.)

Fig. 1. Presentation of First Capsule Used for ^{233}U Addition, October 8, 1968. Scientists in this photo included Dr. Alvin Weinberg (far left), Dr. Glen Seaborg (center) and Paul Haubenriech (right).

This restart was authorized to test the feasibility of Molten Salt Reactors as breeder reactors using the thoron decay chain. After a successful operation, the reactor was shut down again in 1969, without the fuel being removed. In order to maintain a homogeneous matrix, the fuel salt was annealed annually thereafter.

In 1994, an inspection revealed that a large quantity of ^{233}U had accidentally diffused into the off-gas system and had collected primarily in the auxiliary charcoal bed (ACB). Unfortunately, the ACB had an unsafe geometry—a 6.36-inch diameter rather than the 4-inch geometry limit required for ^{233}U. Therefore, the purpose of the 2001 uranium deposit removal (UDR) project was to remove the uranium from the ACB and store it safely.

MSRE 2001 – THE URANIUM DEPOSIT REMOVAL FROM THE AUXILIARY CHARCOAL BED

Challenges

There were many unique radiological and other challenges associated with this project. Uranium-233 has not been used in many other applications; therefore, its singular characteristics are not commonly known. Unlike ^{235}U, ^{233}U contains an energetic gamma emitter hazard that exists because of the associated ^{232}U, which decays to ^{208}Tl. (See Figure 2. below.) The ^{208}Tl emits a 2.6 MeV gamma, 100% of the time, and thus produced a contact dose-rate on the ACB of up to 600 rad/hr.

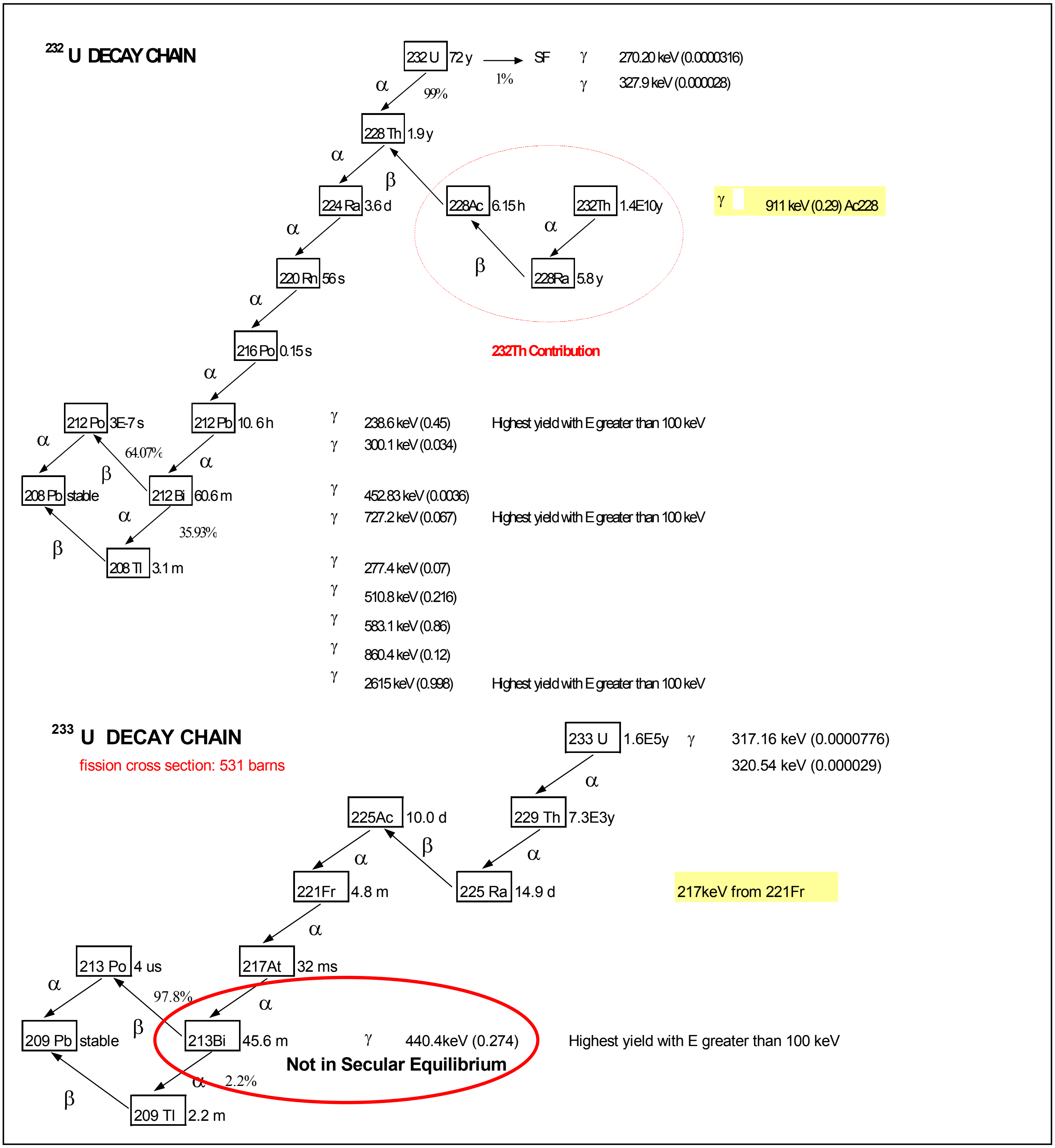

Fig. 2. Fuel Decay Characteristics.[a]

Charcoal, such as that located in the ACB, is flammable and potentially friable. The UF_6 in the charcoal produces several valence states of carbon and fluorine compounds (C_XF), which could cause a mild explosion (deflagration) when heated or shocked. Although the ACB is enclosed in a temporary sheet metal shed, it is not sturdy enough to allow significant negative pressure to qualify as a containment structure. The current work at the MSRE facility continues to be partly experimental, even during decontamination and decommissioning (D&D), because many of these activities were not part of the original design and have never been performed at a similar facility.

Interior space in the ACB shed was limited. The small hoist and the structural components present did not allow for heavy shielding. The pre-existing shielding had been designed and fabricated by a previous contractor whose operational assumptions proved to be infeasible when the components were assembled. Therefore, other methods and materials had to be designed.

Fortunately, extensive mock-up training and testing enabled the engineers and operators to work around the flaws. Nevertheless, the final operational solutions required more intensive hands-on labor and less use of the original robotic equipment. Numerous mock-up trials were performed in an attempt to minimize the time in the high dose-rate areas. The mock-up practice timing, along with the total individual and the collective dose calculations, proved that these procedures could not be performed safely without additional dose-rate reduction and contamination control techniques. Even though several additional technicians were trained for each high dose-rate job, there was still the potential for excessive, unnecessary, individual and collective exposure, administrative overexposure and a dangerous release of contamination.

Value-added design improvements

To reduce the individual and collective dose to the workers, supplemental source shielding was considered mandatory by the radiological engineer and management. Radiological engineering analyses also indicated that the design feature, utilizing large, heavy metal shield components, would be unlikely to match exactly the existing metal and concrete curved structures without leaving unshielded seams, which would in turn allow high dose-rate streaming. These decisions were based on extensive experience with gamma and neutron shielding installation during commercial reactor start-up and naval nuclear submarine overhaul and refueling.

The design team objected to additional shielding, because they thought that the structures and the crane would not accommodate the significant addition of weight required to have the desired reduction in the dose rate. In fact, the inverse square law (employed to reduce the dose rate from spherical emanation) was misapplied in part by the original design team. By removing the shield to a distance of 6 feet above the source, they were able to reduce the thickness of the shield at that location. However, this actually increased the total weight of the shielding, compared to placing shielding closer to the source. Furthermore, the position of the shield made the use of long-handled tools very difficult, because of the exaggerated movement across the fulcrum. Additional contamination control measures were applied, because of the high potential for contamination release, while using plastic sleeving to pass-out heavy and highly- contaminated equipment.

New temporary shield design features

The inverse square law for dose states that the dose is inversely proportional to the square of the distance from the source. However, the shielding weight is increased because the area is directly proportional to the square of the distance. Reversing the inverse square law requires shielding to be placed as close as possible to the source in order to decrease the size and weight of the shielding. In this case, the source was located inside of a right circular cylinder (6-inch diameter pipe), which when viewed from the end, was actually a small source (almost a point source).

The radiological engineer recommended that approximately 40 pounds of lead be poured into the existing hollow, 6-inch diameter, stainless steel, sheet metal pipe cap. The original purpose of the hollow pipe cap had been to cover the source, as a criticality control measure, in order to prevent liquid from inadvertently entering the ACB. Adding the lead to the pipe cap produced a shield over two inches thick. This shield was then used with a lockable, remote handle (Peter's tool) to hold it as a shadow shield prior to removing the ACB dome-end. Afterward, it was placed directly on the source and served to reduce the dose-rate by greater than a factor of ten. (See Figure 3.)

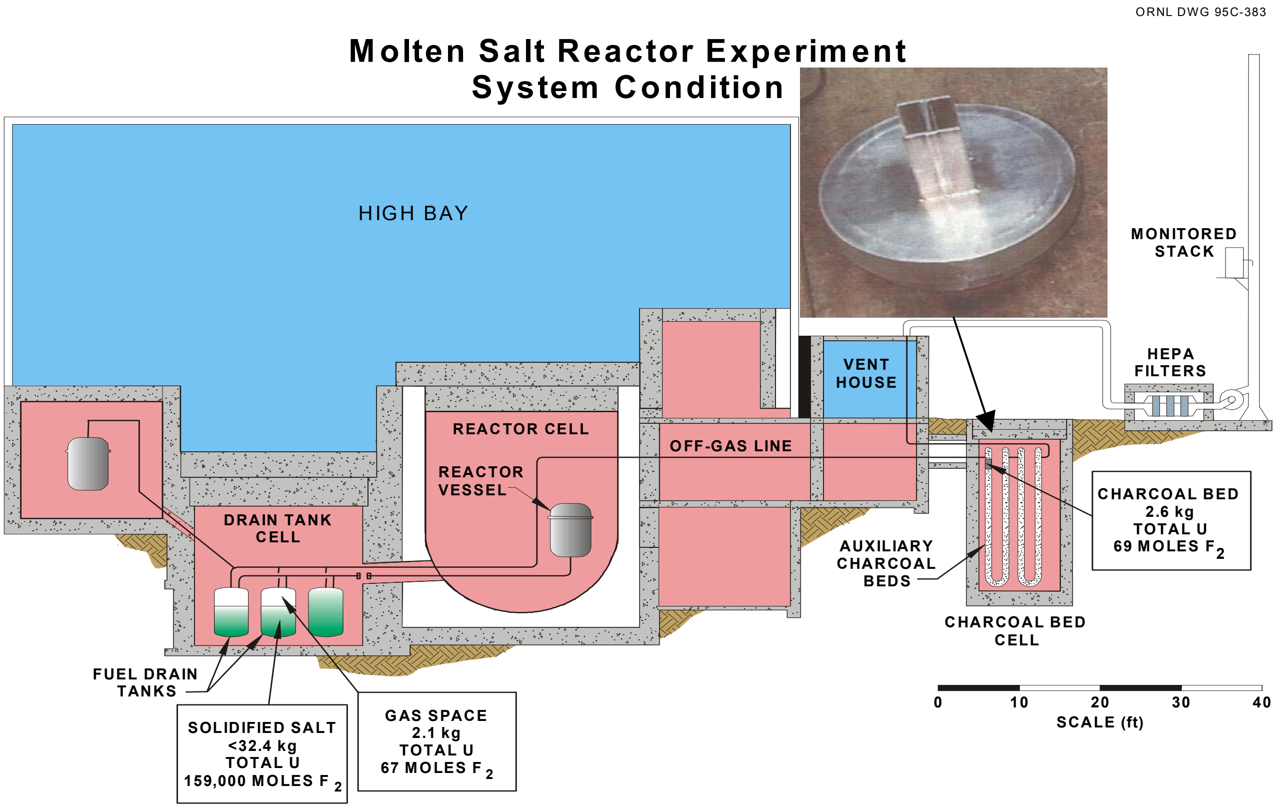

Fig. 3. Diagram of MSRE design[b], with an inset photographic enlargement of lead-cap shield.

A four-inch thick steel shield (doughnut or collar-sleeve type) was also added to the steel centering and milling shafts. This device was made to slide down the shaft to provide a shadow shield for workers, while the center-shielded manipulator plug was removed. Workers were thus able to hold on to the shaft directly over the unshielded port with a shadow cone of protection equivalent to the normally-installed shielding. The pre-job worker exposure estimates were re-calculated using the shielded dose-rates and found to be acceptable.

Fortunately, the multiple layers of additional shielding, including the lead-filled cap, the shaft doughnut and numerous lead blankets, decreased dose-rates to a lower level than those in the original estimates. (See Figure 4.) Technicians were able to use the polymorphous shielding concurrently, rather than consecutively; therefore, dose-rates were reduced by a factor of 20 or more for much of the work. In the final analysis, the eventual collective exposure was only 30% of the original estimate.

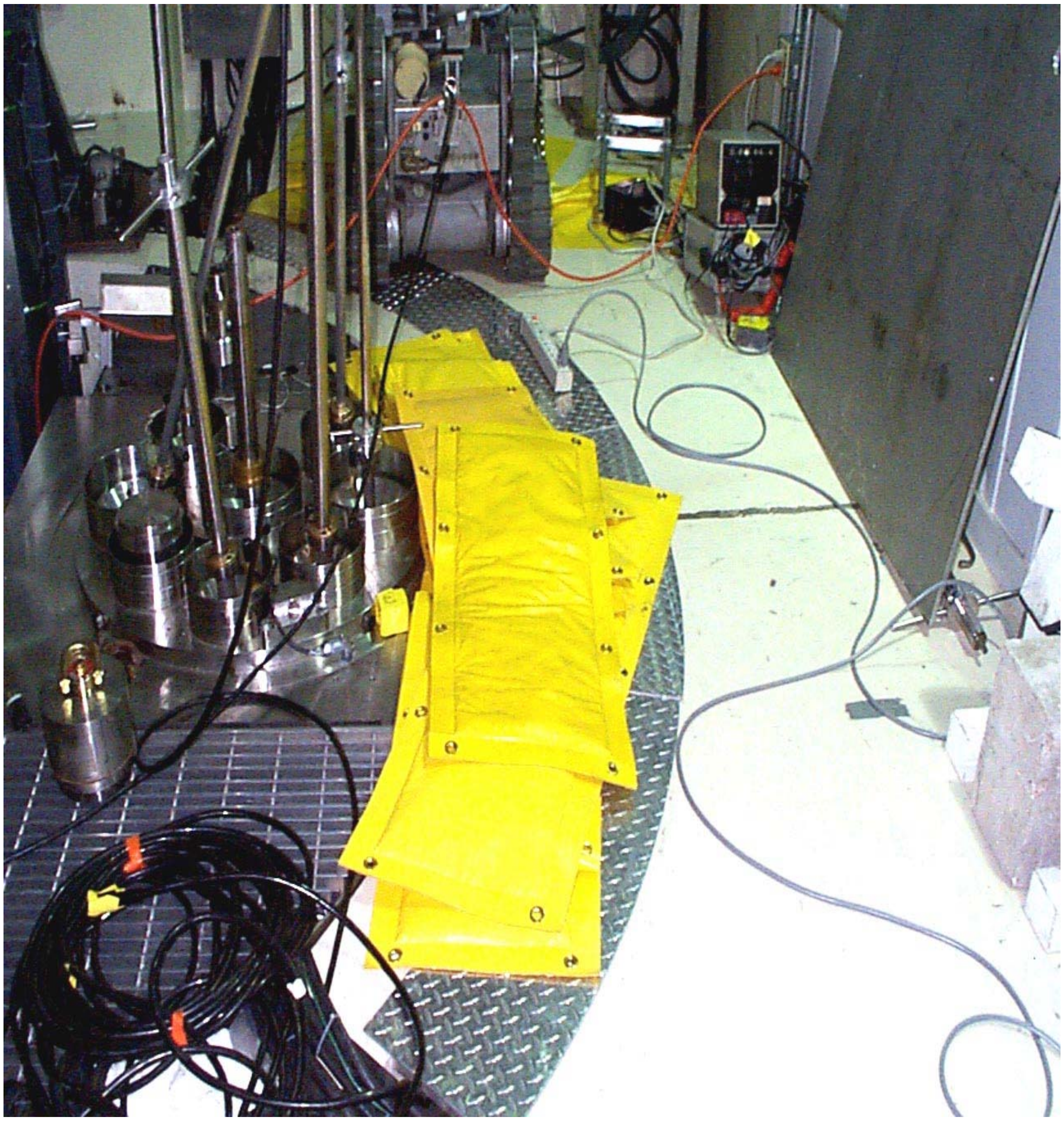

Fig. 4. Lead Blankets over the Source.

Additional contamination controls

PVC sleeving material had been chosen because heavy equipment would be passed-out through large diameter sleeving and PVC is much tougher and stronger than polyethylene. However, a problem was encountered during mock-up training while attempting to heat-seal the pass-out sleeving during removal of equipment from the tool port. The PVC sleeving would melt but would not seal. This difficulty was investigated at the ORNL tent/containment shop, where the sleeving was manufactured. PVC material, experimenters discovered, can only be seal-welded with a radio-frequency (RF) sealing machine. Only polyethylene sleeving could be heat-sealed. Thus a decision was made to use the twist and cut method. During the initial mock up practice this was not satisfactory. At first, technicians considered the method to be adequate, but, upon closer inspection, found the sleeving-cut to be too loose and thereby contamination could be lost. Radiological engineering decided to check for any lost contamination by using a fluorescent tracer and a black light. A liquid fluorescent tracer was tried first, but a fluorescent powder was found to be more sensitive. Therefore, the powder was placed in the sleeve, and the cut was performed over white blotter paper. Some spilled powder was noticed with the naked eye, and more was seen with the black light, which proved that corrective action had to be taken.

The following improvements were made to the plastic sleeving twist-and-cut method:

- A dilute (2:1) solution of water and Elmer's glue was added to the interior of the sleeving prior to the cut, to prevent loosing dry contamination.
- Electrician's elastic PVC tape was applied, instead of duct tape, to provide a tight compression to the heavy, stiff PVC sleeving.
- A ratcheting PVC pipe cutter was used to provide a very smooth cut through the pre-taped sleeving.
- A steel pipe clamp was added to hold the sleeving to the metal pipe (dock-out port), instead of the designed groove and "O" ring, which could not hold against the tugging during pass-out.

Unforeseen conditions encountered

Prior to the UDR, the charcoal in the ACB was thought to be granular. A previous test drill indicated that it was a more like a sticky mass of granules. However, upon opening the pipe, the material was found to be an extremely hard solid, which was very difficult to break even with a steel chisel and a sledgehammer. The original steel chisel fractured before cleaving the mass; but a well-designed, hardened steel spike was successful. After breaking the hard mass, the resulting material became a fine powder, which both penetrated and clogged the High Efficiency Particulate Air (HEPA) filters. Thus the work time was increased by a factor of 16, and numerous contamination releases were encountered.

Radiological Conditions

The general area dose-rate above the portable maintenance shield (PMS) was calculated to be approximately 37 mrem/hr gamma and 3.0 mrem/hr neutron, for a total of 40 mrem/hr general area whole body at one inch above the shield (assuming no streaming or hot spots). Since personnel were to be intermittently kneeling, sitting and lying on the shield, the extremity and whole body exposures were similar. (See Table I.)

Table I. Dose-rates used to calculate the pre-job estimates.

Locations	Dose-rates
3 feet from open source inside box with no shielding:	3.5 rem/hr
6 feet from source at open hole with no sheilding:	1.3 rem/hr
6.2 feet from source with 2 inches lead shielding at open hole.	150 mrem/hr
7 feet from source with open hole and 2 inches lead shield in place,:	126 mrem/hr
6.1 feet from source with 2 inches lead shielding and closed port:	22 mrem/hr
Top of the shield (beside the ports) with 2 inches of lead over the source and 4 inches of stainless steel shield in place:	3.3 mrem/hr
Center of the unshielded dock-out port 5 feet from centerline at 90 degrees from the source.:	16 mrem/hr

ALARA Work Controls

The following routine ALARA work controls were exercised to regulate radiological work activities:

- All work was prefabricated in non-radiological staging areas to the maximum extent possible, with mock-up training of significant work activities.
- An emergency response matrix was required for operator training as well as for reasonable repair and recovery activities.
- Radiological Work Permits (RWP's) were issued for all radiological work, and pre-job briefings were held.
- Personnel were trained on hot spot locations and instructed to stay away from elevated dose rate areas.
- Real-time radiation measurements collected by the radiological control technicians (RCT's) were evaluated periodically to confirm the presence of expected radiation levels.
- Personnel working within the radiation area were required to wear direct-reading gamma dosimeters and whole body thermoluminescent dosimeters (TLD's).
- Direct-reading, alarming gamma dosimeters were read and evaluated frequently, as directed by the RCT's.
- Extremity dosimetry was worn whenever hands were placed near sources of radioactivity and when other "close proximity" work was performed, especially during the docking out operations.
- Observers and nonessential personnel were prohibited from the work area but were allowed to view the operation from the control room monitors.

Work performed outside the scope of the ALARA plan, which was not in the dose calculations, include the following:

- routine radiological work activities performed in this area,
- staging equipment,
- pre-job set up,
- some utility installation,
- demobilization,
- RCT's periodically checking for surface contamination.

The UDR process for removing the deposit consisted of the following basic steps, which were included in the pre-job dose estimates:

- cutting the top off of the ACB,
- removing the steel wool pre-filter,
- scabbling and vacuuming the charcoal,
- removing tools.

Applicability

The activities that were performed by personnel located in the charcoal bed cell (CBC) ventilated enclosure required hands-on work where unshielded dose rates would have been up to a 1,300 mrem/hr. Therefore, temporary shielding and timed mock-up training were mandatory to minimize personnel exposure.

Work performed outside the CBC Ventilated Enclosure was not included in work duration calculations, nor were activities performed remotely by using the Ventilated Enclosure crane and ANDROS robotic manipulator. (See Table II.)

Table II. Pre-job estimated dose-rates with 2" Pb on 3 sides/ 1-3/4" stainless steel on the back of collector and cyclone.

Distance from Collector/Cyclone (ft)	Estimated Dose Rate (mR/h)
1	44
2	34
3	27
4	23
7.5	8

Technicians moved about and worked approximately one to two feet from the Vacuum Collector/Cyclone assembly. RCT's performed radiological monitoring in the same area. Averaging work locations based on Table I, it was anticipated that Technician one, Technician two and the RCT's would perform their tasks where the general area dose-rates were estimated to be about 40 mrem/hr, 30 mrem/h and 10 mrem/hr—which was in addition to the dose rates from Table I sources. (See Table III.)

Table III. Worker exposure estimates**.**

Worker	Exposure Estimate (mrem)	Activity	Time (h)
Technician 1	656	Cut ACB/ Scabble/Vacuum	12.1
Technician 2	656	Same	12.1
RCT	656	Survey and support.	12.1

Removal Operations pre-job estimates

A minimum total of 1.64 person-rem was anticipated, therefore, a UDR ALARA goal was chosen at 2.0 person-rem.

The detailed activity is presented in Table IV. below. Expecting the actual duration times to increase with Personal Protective Equipment (PPE) and space restrictions, some additional time was added as a conservative estimate. No airborne contamination was expected and respiratory protection was only used when the system breached as a precautionary measure, therefore no PPE protection factor was taken.

The following times were the averages observed during mock-up training with no equipment failure. The total UDR job exposures were expected to be higher than 2.0 person-rem with unforeseen operations. There was a Radiological Engineering assessment required at 1.5 person-rem, and at 2.0 person-rem, there was a hold-point for an engineering and management review.

ALARA Dose Control Levels

The following ALARA Dose Control Levels were established for the uranium deposit removal and transfer phase, based on the mock-up, timed trials and exposure estimates:

Table IV. Uranium deposit and equipment removal.

Activity	Time	mrem/hr	Totals: mrem/each position
Shear the thermocouple off of the ACB/Technician	0.4 hr	40	16
Move the shear tool and severed thermocouple to the dock-out shelf/Technician	0.2 hr	40	8
Prepare for installation of the cutter/drive unit/Millwright	0.3 hr	40	12
Install the cutter/drive unit/Millwright	0.3 hr	60	18
Verify operation of cutter/drive unit/Technician	0.3 hr	40	12
Disconnect the cutter/drive unit from the drive shaft/Technician	0.3 hr	60	18
Remove the cutter/drive unit/Millwright	1.0 hr	60	60
Bag-out the cutter drive shaft and bag-in the extractor tool/Technician	1 hr	100	100
Shear thermocouple from wire mesh assembly/Technician	0.2 hr	40	8
Place wire mesh assembly in poly box/Technician	0.5 hr	40	20
Dock-out using turbo-pig/Technician	2 hr	50	100
Scabble and vacuum/Technician	4 hr	50	200
Shut down ventilation system (close hand valves)/Technician	0.1 hr	40	4
Dock-out/dock-in tools/Poly pig/Technician	2 hr	40	80

The total measured mock-up time for "Removal of the Deposit" was 12.6 hours. The total calculated exposure for each of the "operations worker position" times two worker positions each of 656-mrem was 1312 mrem. The attendant RCT was expected to receive an additional 328-mrem exposure for a total collective exposure of 1.64 person-rem which was considered to be ALARA with no conservative factor (no margin for error).

For ALARA determinations, a repair to a tool, a manipulator or sleeving (although not planned) was considered. During damages and repairs, a tape patch was allowed only as a temporary fix until new material could be installed. For each of these situations, a duration of approximately 0.75 hr (duration of bag-out of cutter drive shaft/bag-in of extractor shaft) was expected for an additional 30 mrem exposure each. However, since these were unlikely event no exposure was added. These were accounted for by rounding the ALARA Goal upward. All calculations were based on temporary or permanent shielding being in place over the ACB after the ACB steel wool pre-filter had been removed and until the charcoal vacuuming started. Any additional work without temporary shielding would increase the dose rate by at least a factor of 10 over the center port and a factor of seven over the remainder of the maintenance shield. The use of lead blankets would probably decrease the exposure level further in the work areas.

The maximum pre-job exposure calculation for each worker was estimated to be less than 500 mrem whole-body, which was a temporary hold-point for any worker during the UDR. An initial extension was established at 600-mrem whole body. (See Table V.)

Table V. Worker individual whole-body dose hold-points.

Whole-Body Hold-Point Per Worker	Requirements
500 mrem	Workers were to be replaced at this level and project manager approval required to exceed this level
600 mrem whole-body final Hold Point per worker	With permission of the project manager

The collective dose was calculated to be 1.64 person-rem for the UDR prior to transport with no contingencies. The ALARA Goal and the collective dose hold-point were 2.0 person-rem and work would be stopped prior to reaching this collective dose for reassessment/re-engineering. (See Table VI.)

Table VI. UDR Project "Collective Dose" hold-points.

Person-rem collective dose	Requirements
1.5	An interim review point for the Radiological Engineer review
2.0 (collective dose hold-point)	To stop the job for an engineering and management review

During installation of the equipment, the shield did not fit up to the concrete rim as designed. Thus a large seam was left open in the work area, which allowed the workers to be exposed to approximately 700 mR/hr. The RCT's used the lead blankets to shield the dose rate to approximately 2 mR/hr. The lead blankets were more effective when placed side wise and stuffed directly into the seam. The installation crew also used the lead filled pipe cap and the 4-inch doughnut shadow shield concurrently. The combined use of each of the temporary shields eventually allowed workers to stay in the immediate work area for as long as necessary to complete the job while maintaining doses ALARA.

During the actual performance of the job, there were several unforeseen problems which caused a change of scope of the work. The charcoal material in the ACB was found to be a solid material, which was extremely hard. A chisel was welded to a solid 6-inch diameter center shaft and was used along with a sledgehammer to break the material. However, the chisel was destroyed before the diamond-like material could be broken and had to be replaced. A well-designed, hardened, sharply pointed spike was then manufactured and was eventually successful at fracturing the material. Nevertheless, continuous pounding with a heavy sledgehammer was required. Thus the total work time in the immediate area was eventually 16 times greater than the timed estimates, even with the previous conservative additions.

CONCLUSIONS

Because of the extensive additional temporary shielding and other ALARA controls, the total dose accumulated by 18 workers was only 478.4 mrem. This amounted to only 30% of the original dose estimate and only 24% of the ALARA goal. The following tables document the dose-rates during the UDR project (Table VII) and in the ACB after the uranium deposit had been removed Table (VIII).

Table VII. Resulting post-job dosimetry statistics

Worker	Dose Rate
Maximum dose	100.7 mrem
Average dose	26.6 mrem
Only 4 workers	Greater than 50 mrem for the job

Table VIII. Dose rates in the ACB, over the Portable Maintenance Shield after the UDR[c]. (The "M" location is over the center port unshielded and open.)

Measurements in mR/h					
*Location	** Contact	At 1 foot	At 2 feet	At 3 feet	At 4 feet
A	1.2	1.3	1.5	1.9	4.8
B	1.7	1.8	2.2	3.0	4.4
C	2.1	2.3	2.4	3.0	3.6
D	2.3	2.9	3.1	3.4	4.2
E	7.0	10.0	4.0	4.0	4.0
F	0.8	1.5	1.7	2.9	5.0
G	2.4	1.5	2.0	2.6	4.0
H	1.9	2.2	3.2	3.9	4.5
I	3.0	3.7	10.0	7.0	3.8
J	8.0	4.0	2.9	2.5	2.5
K	0.7	1.4	2.0	3.0	4.4
L	2.0	2.2	2.8	4.2	4.7
M	80.0	50.0	30.0	21.0	16.0
N	3.3	4.5	5.0	6.0	6.0
O	6.0	3.0	2.7	2.7	2.9
P	1.2	2.2	2.6	3.5	4.1
Q	2.1	2.3	2.9	3.6	4.2
R	2.6	3.2	5.0	6.0	8.0
S	2.3	2.3	3.0	3.0	2.7
T	5.0	2.7	2.5	2.3	2.5
U	1.8	2.8	3.2	3.6	3.8
V	2.0	2.8	3.0	3.3	3.2
W	2.2	2.8	2.7	3.2	3.4
X	2.0	2.6	2.6	2.5	2.4
Y	3.0	2.2	2.2	2.3	2.2

ALARA LESSONS LEARNED

1. For ALARA planning, expect the unexpected when attempting to perform unique work activities involving high dose rates and high levels of contamination and which have no precedent. These type of (research) projects may result in unforeseen situations and require conservative control measures.
2. Extensive experience with similar activities is very useful in predicting potential pitfalls.
3. Experienced radiological engineers should be included in the mechanical engineering design phase of new projects that involve ionizing radiation hazards.
4. Radiological engineers who do not have previous experience with a particular planned activity should have access to a network of professionals who have had similar experience.
5. Radiological engineers are often asked to review a design after all decisions have been made. Substantial previous investments preclude a substantial redesign of the equipment or the project.

6. Mock-up testing and training on new equipment is absolutely essential prior to starting to work in high dose-rate or high-contamination level work areas.
7. The actual work activity may vary from the original concept.
8. The actual work time may be much greater than the mock up training time.
9. Engineered and temporary shielding should be applied directly to the source whenever possible to "nip it in the bud." Discrete sources should be shielded down to very safe (near background) levels.
10. The use of the traditional Time, Distance and Shielding may be misapplied and should be carefully reviewed in the design phase.
11. The employment of shielding is more consistent with the ALARA principle and is normally less expensive than the training of additional personnel for the purpose of receiving additional exposure.
12. Work with high levels of contamination outside of hot cells requires training, positive, precise controls and redundant levels of protection.
13. A fixative (glue or wetting) is often useful in controlling high levels of contamination during direct handling and during containment pass out procedures.
14. Polyethylene plastic can be heat-sealed, but PVC cannot.
15. The elastic PVC electrician's tape was found to be superior to duct tape when performing the tape and cut procedure on pass-out sleeving.
16. The use of a fluorescent tracer with a black light was found to be a helpful instructional tool during contamination control training.

FOOTNOTES

[a] Jeff Chapman, Kevin Meyer, Ron Brandenburg, Daniel Huddleston, Richard Bailey, Neil Crass, Ian Gross, "The Measurement of ^{208}Tl to Estimate ^{233}U Fissile Mass at MSRE," Canberra, August 21, 2001.
[b] I.G. Gross, Jeff Chapmanet al, "In-Situ Nondestructive AssayMeasurements at the ORNL Molten Salt Reactor" Presented at the 43rd Annual Meeting of the Health Physics Society July 12-16, 1998.
[C] Surveys Courtesy Janet Cox, MSRE Health Physics Supervisor.