# Characterization of the Process Mechanical Cell at the West Valley Demonstration Project

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## ABSTRACT

The West Valley Demonstration Project has initiated decontamination and dismantlement (D&D) of the most highly radioactive and contaminated cells in a former spent nuclear fuel reprocessing plant. The goals of the D&D project are to remove loose debris in the cells and estimate the residual radioactivity level of legacy plant equipment. To support accomplishment of these goals, a unique characterization approach was developed to gather the information to meet anticipated Waste Isolation Pilot Plant (WIPP) acceptance criteria for remote-handled transuranic waste, and to facilitate segregation and packaging operations.

Implementation of the characterization approach included the development and use of innovative, remote technology for measuring gamma radiation within the hot cell. The technology was used to identify and quantify radiation from individual debris items in radiation fields up to 2,000 R/hr (20 sieverts/hr). Sampling and analysis of the debris were also performed via remote handling means.

Significant challenges associated with characterizing the highly radioactive and highly contaminated hot cells were encountered. The innovative solutions for meeting these challenges are applicable throughout the Department of Energy Complex and help support the goal of targeting D&D efforts toward reducing risks to public health and the environment.

## BACKGROUND

The West Valley Demonstration Project (WVDP) is located at the site of the former and only commercial spent nuclear fuel (SNF) reprocessing facility to have operated in the United States. The plant was constructed in 1966 on land owned by the State of New York and operated for six years, reprocessing approximately 700 tons (636 metric tons) of SNF, and producing about 600,000 gallons (2,270 cubic meters) of liquid high-level radioactive waste.

In 1980, Congress passed the WVDP Act, authorizing the Department of Energy (DOE) to conduct a nuclear waste management demonstration project at the Western New York Nuclear Service Center near West Valley, NY. Since 1982, the DOE and its partners - the New York State Energy Research and Development Authority and the prime contractor, West Valley Nuclear Services Company (WVNSCO) - have demonstrated that the high-level waste (HLW) can be safely and successfully placed into a vitrified waste form that provides long-term stability in the environment. In 2002, the WVDP completed its HLW vitrification mission and produced the final canister of vitrified HLW glass.

The WVDP has now turned its focus to decontaminating shielded hot cells used during spent fuel reprocessing operations, one of them being the Process Mechanical Cell (PMC). The PMC along with an adjoining cell, the General Purpose Cell (GPC), are estimated to contain the greatest amount of long-lived radionuclides at the site, and therefore, presents the greatest potential long-term risk to public health and the environment. Decontamination and dismantlement (D&D) of the PMC includes the repair and replacement of failed equipment; retrieving, characterizing, processing, packaging, and storing loose debris; and determining the residual radioactivity content of the remaining plant equipment.

## DESCRIPTION OF THE PROCESS MECHANICAL CELL

The PMC is 12 feet (3.7 m) wide, 52 feet (15.8 m) long, and 25 feet (7.6 m) high. The concrete walls and floor of the PMC are 5.5 feet (1.7 m) thick and the ceiling is 6 feet (1.8 m) thick. The floor is covered with 304L stainless steel, which also extends up the walls to a height of 20.67 feet (6.3 m). The walls above the stainless steel are coated with a carboline-based paint.

During former reprocessing operations, the PMC was used to mechanically size-reduce SNF, separating the fuel assemblies from the spent fuel and shearing the fuel into 0.50 to 2.0 inch (1.27 to 5.08 cm) lengths. After chemical dissolution of the sheared fuel, some of the leached fuel hulls were transferred from the GPC through the PMC to the Analytical Labs where the effectiveness of the dissolution process was assessed.

Legacy plant equipment in the PMC included bridge cranes and a bridge-mounted power manipulator, wall-mounted manipulators, a cutoff saw, and a fuel shear. The equipment is in a failed condition or has not been operated for many years. The old bridge cranes and power manipulator and some of the wall-mounted manipulators have been replaced to enable decontamination of the cell.

The PMC is highly contaminated with spent fuel, activation products, and fission product radionuclides. Radiation levels in the cells vary from a general field of 30 to 100 R/hr (0.3 to 1 sieverts/hr) six feet from the floor, to 2,000 R/hr (20 sieverts/hr) in localized hot spots. Removable contamination levels are on the order of billions of disintegrations per minute. In addition, laboratory waste placed in the PMC from cleanup of the adjacent Analytical & Process Chemistry (A&PC) Laboratory in the 1980s may contain hazardous constituents.

## CHARACTERIZATION STRATEGY

On an upper-tier level, the approach taken toward characterizing the loose debris and legacy equipment in the cell did not differ from other routine characterization programs. That is, the purpose of the characterization program was defined, what was known about the composition of cell contents was collected, that process knowledge was evaluated and information gaps identified, and specific locations and/or debris forms were targeted for additional data gathering. Two important factors, however, required the standard characterization methodology to be tailored to this unique situation: the highly radioactive nature of the debris, and the need to

support an acceleration of the cleanup schedule. These factors necessitated a creative approach: conduct the least amount of data gathering as possible concurrent with cleanup operations, and meet both the debris characterization and legacy equipment residual inventory estimate goals; develop techniques that maximize the amount of data gathered while being limited to remote handling means; and keep personnel exposure as-low-as-reasonably-achievable (ALARA).

The eventual disposal site for the PMC debris waste was not known, therefore, the characterization approach developed targeted the requirements for the potential disposal facility, the Waste Isolation Pilot Plant (WIPP). The waste acceptance criteria (WAC) existing during the characterization planning period were evaluated, and the criteria dealing with the waste itself and not that of the packaged waste form were incorporated into the approach. Since waste acceptance criteria for highly radioactive (i.e., remote-handled) debris are not finalized, it was assumed that the criteria for the packaged waste form would be addressed when the WVDP is readying its waste packages for shipment.

## PRELIMINARY DEBRIS CHARACTERIZATION

During characterization planning, commitment was made to an ambitious acceleration of the cleanup schedule. This prompted a change in the characterization strategy; the debris would be preliminarily characterized using available information, which would be followed by additional field data gathering to be performed concurrently with actual packaging operations. This approach allowed packaging activities to start while ensuring that operational safety for packaging and storage would be maintained.

Considering the spent nuclear fuel radionuclides expected to be present (Pu-238, Pu-239, Pu-240, Am-241, Cs-137, and Co-60), it was projected that much of the debris would be transuranic (TRU) waste, and likely to be remote-handled (RH) TRU waste. The goal of the preliminary characterization phase, thus, was to support minimizing the amount of TRU waste or RH-TRU waste produced by designating which waste forms should be segregated during packaging. Preliminary characterization also involved development of a correlation between the package exposure rate and available information on radionuclide content to be able to identify filled waste packages as either TRU waste or low-level waste (LLW). The debris waste forms were evaluated for their potential radionuclide content which took into account many factors: the nature of contamination (surface contamination only vs. intrinsic to the debris form, radiological only vs. radiological and chemical); the previous processing use of the debris and the duration of its presence in the cell; the presence of significant activated metal and/or spent fuel; the planned method of retrieval (picking it up piece by piece vs. vacuuming); and whether the debris form could potentially be decontaminated to become LLW.

The evaluation resulted in the identification of six preliminary waste streams: contaminated equipment and scrap from fuel and waste handling, fuel assembly hardware, leached fuel hulls, fine particles generated primarily from in-cell cutting activities, laboratory waste, and potentially some miscellaneous fuel rod pieces. Table 1 below presents a summary of that evaluation.

From a preliminary characterization basis, the segregation scheme proved successful in minimizing the number of TRU waste packages produced. Some waste packages containing surface-contaminated materials generated during preparations for in-cell cleanup were categorized as LLW, while others containing legacy debris were categorized as contact-handled TRU waste. Most waste packages however were categorized as RH-TRU waste.

**Table 1**
**Debris Segregation Evaluation**

| **Type of Debris** | **Surface-Contaminated (S) or Intrinsic (I)** | **Radiological (R) or Chemical (C)** | **Activated (A) or Spent Fuel (S)** | **Picking Up (P) or Vacuuming (V)** | **Able to be Decontaminated to LLW?** |
|---|---|---|---|---|---|
| Contaminated Equipment and Scrap | S | R | n/a[1] | P | Y |
| Fuel Assembly Hardware | S, I | R | A | P | Y |
| Leached Fuel Hulls | S, I | R | A, S | V | N |
| Particles | I | R | A, S | V | N |
| Laboratory Waste | S | R, C | n/a[1] | P | Y |
| Fuel Pieces | S, I | R | A, S | V, P | N |

Note 1 - N/A due to the majority of the physical volume being uncontaminated material.

## IN-CELL GAMMA SPECTROSCOPY

Potential nondestructive measurement equipment was researched to find or develop a single unit to satisfy both near-term and long-term goals, to perform in-cell measurements of the highly radioactive debris to minimize the amount of sampling and analysis required, and also to quantify residual radioactivity after debris collection is complete. It was intended that the in-cell measurements would identify and quantify radioisotopic content, not simply provide general or gross radioactivity levels, and moreover, would do so in a wide view, not debris piece-by-debris piece. In addition, a combined representation of the visual evidence and the gamma radiation measurements would allow for easier interpretation of the results. The BNFL Instruments RadScan®:700 unit was selected as the equipment most likely to meet all these needs.

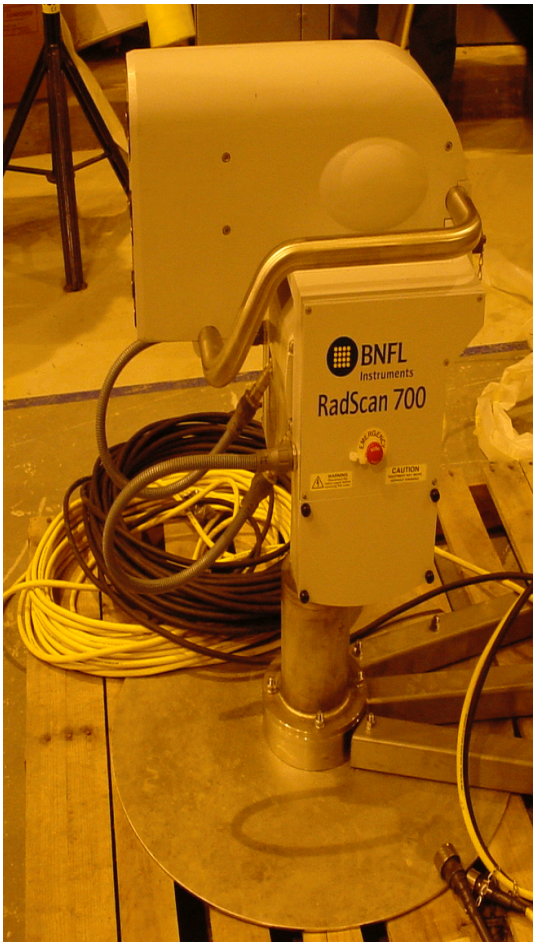


**Fig. 1 - RADSCAN® UNIT**

The BNFL Instruments RadScan®:700 is a gamma spectroscopy unit. To address site-specific conditions, the standard detector crystal for the unit was replaced with one approximately 1/20 the size to account for the higher radiation levels in the PMC. The unit was also energy calibrated to Cs-137 and Co-60, and efficiency calibrated to Cs-137, Co-60, and Eu-152 prior to shipment to the WVDP. It was tested and deployed over a three-week time period. Initially, calibration and operational mock-up were conducted in a non-radioactive facility to confirm the functionality of the equipment, followed by installation and setup in the PMC. Figure 1 shows the RadScan®:700 unit during testing. The unit was then deployed from the in-cell crane at 11 different locations to provide video images and gamma radiation measurements of the entire cell. The deployment positions were selected to provide coverage for the entire floor area and for the lower portions of the walls, and also to determine to some extent the effect of the radiation field around the detector on the identification of “hot spots.” A specially designed deployment stand provided contact points with the cell wall to stabilize the unit during measurements.

Gamma radiation count rates and videotape recordings were obtained of the floor debris and select pieces of legacy equipment, including the fuel bundle shear, shear vent pipe, and fuel assembly cutoff saw. Both the automatic and manual modes were utilized for these scans. Areas of higher radiation were identified and evaluated, and gamma spectra collected. The background

radiation fields in the 11 positions ranged from 1.5 to 70 R/hr (0.015 to 0.7 sieverts/hr). Figure 2 shows a gamma radiation overlay on the video image of the shear vent pipe, and Figure 3 is a graphical depiction of the Cs-137 activity for one of the automatic scans.

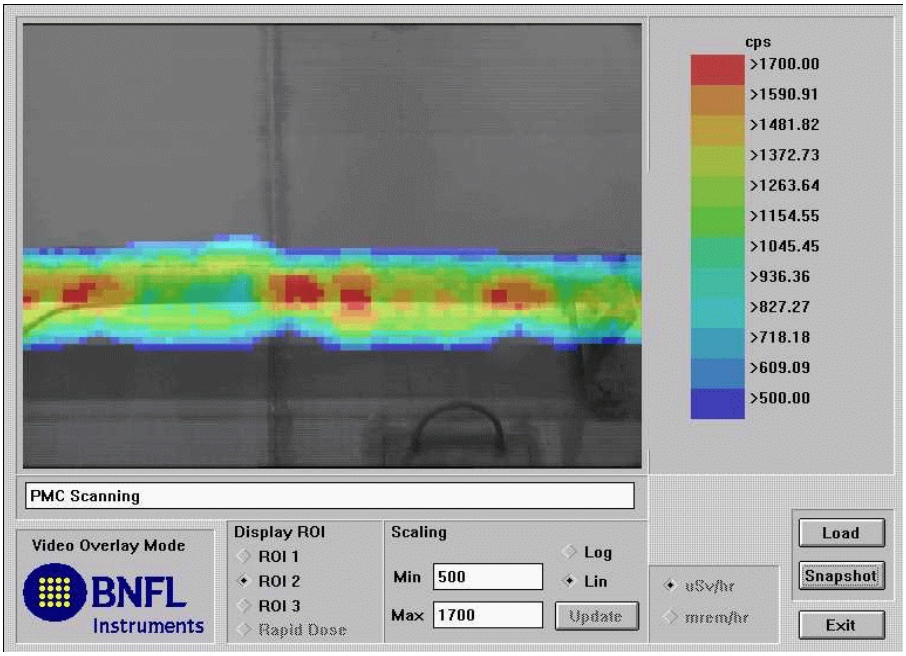


**Fig. 2 - SHEAR VENT PIPE OVERLAY**

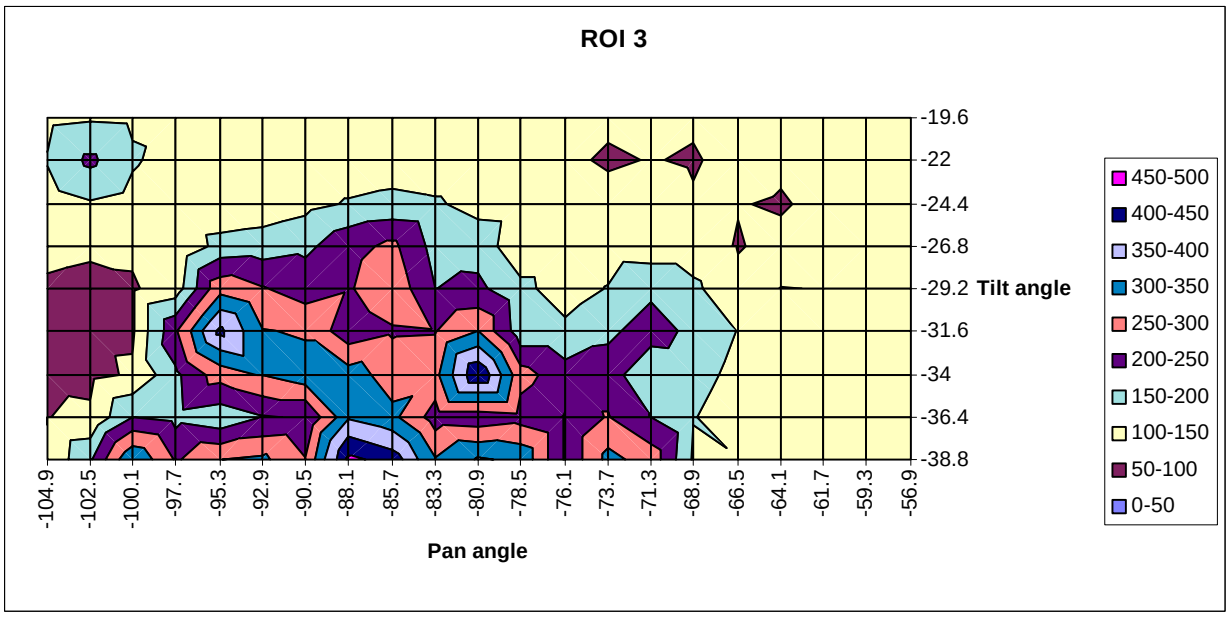


**Fig. 3 - GRAPHICAL PLOT OF CESIUM 137 ACTIVITY**

## SAMPLING AND ANALYSIS

To supplement available process knowledge, sampling and analysis were conducted to generate the radioisotopic and chemical content information necessary for eventual waste package disposal. Specific sample points within the debris were selected to encompass the variety of matrices included in each category’s general description (e.g., metal, glass, wood), to cover the entire geographic area of the cell floor, and to cover the range of gamma radioactivity levels associated with specific debris or areas. The results of the RadScan®:700 scanning of the cell floor were also used to support selection of sampling locations.

Sampling was performed by retrieving pieces of debris directly with the in-cell remote handling equipment or via vacuum sampler. Smear samples were also obtained from the retrieved pieces, and the smear samples and remaining “unsmeared” portions were then placed into a multi-compartmented sample holder. The smear samples were collected to determine if correlations could be developed between removable and overall radionuclide content and, thus, to determine if they could be used as a quicker means to characterize debris during field activities. The filled sample holder was then transferred through a hatchway into the A&PC Laboratory for analysis. The fuel pieces category was not sampled as such pieces were not able to be distinguished from the much more prevalent leached fuel hulls.

The analytical parameters shown below in Tables 2 and 3 were developed based on process knowledge of the debris and historical information on other WVDP wastes. Some samples were analyzed only for gross alpha, gross beta/gamma, and a gamma scan to reduce the amount of work effort associated with conducting the analyses while still providing an indication of the overall consistency of the results. Chemical parameters were chosen to determine if any of the debris would be categorized as mixed waste based on federal and New York State regulations on waste characterization contained in Title 40 Code of Federal Regulations Part 261 and Section 6 of the New York State Code of Rules and Regulations Part 371, respectively. Note, the New York State waste characterization regulations mirror the federal ones with the exception that polychlorinated biphenyls (PCBs) contaminated wastes are regulated as hazardous wastes under New York regulations.

In the initial analysis plan, limited testing was to be performed on-site while the remainder would be performed at an off-site subcontract laboratory. As initial radiological analysis results were produced, it became evident that the radiological content of the samples exceeded the acceptance levels of the off-site laboratory. Therefore, the on-site laboratory conducted all the radiological analyses, but much of the chemical parameters remain untested due to the current lack of analytical capability. That capability is being established for inorganic parameters, while testing for organic parameters on the laboratory waste stream is being deferred to some time in the future.

**Table 2**
**Radiological Parameters**

| |
|---|
| **General Gamma Scan:** K-40, Cr-51, Mn-54, Co-60, Nb-94, Zr-95, Ru-106, Ag-110m, Sb-125, Sb-126, I-131, Cs-134, Cs-137, Ce-144, Eu-152, Eu-154, Eu-155, Eu-156, Bi-212, Pb-212, Ra-226, Ra-228, Ac-227, Th-228, Pa-231, Pa-234, U-235, Np-237, Np-239, Am-241 |
| **Other Isotopes:** H-3, C-14, Fe-55, Ni-59, Ni-63, Sr-90, Tc-99, I-129, Th-228, Th-229, Th-230, Th-232, U-232, U-233/234, U-235/236, U-238, Np-237, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Am-243, Cm-242, Cm-243/244, gross alpha, gross beta/gamma |

Note that isotopes reported together (e.g., Pu-239/240) were not designated to undergo mass spectrometry to determine individual isotope concentrations. Rather, percent abundances from process knowledge information would be used to develop individual values from the totals reported.

**Table 3**
**Chemical Parameters**

| Analysis | Laboratory Waste Stream[1] | Laboratory Waste Internal Liquid | Laboratory Waste Internal Solid | All Other Waste Streams[1] |
|---|---|---|---|---|
| Moisture Percent | - | **X** | - | - |
| Paint Filter Liquids Test | - | **X** | - | - |
| Ignitability | - | **X** | - | - |
| Corrosivity (pH) | - | **X** | - | - |
| Reactivity (Cyanide & Sulfide) | - | **X** | - | - |
| Toxicity Characteristic Leaching Procedure (TCLP) Metals | **X** | **X** | **X** | **X** |
| TCLP Volatile Organic | - | **X** | - | - |
| TCLP Semi-Volatile Organic | **X** | **X** | **X** | - |
| PCBs | **X** | **X** | - | - |
| Inorganic Hazardous Constituents (Ni, Zn only) | **X** | **X** | **X** | **X** |

Note 1:
If items in this category are found to contain liquid or solid, then the analyses under "Laboratory Waste Internal Liquid" or "Laboratory Waste Internal Solid" will be performed.
Nomenclature:

- **X** Analysis required.
- \- Analysis NOT required, process knowledge or historical information to be used.

## RESULTS

The results of the in-cell gamma spectroscopy and analysis efforts confirmed to a great extent the process knowledge-based preliminary characterization of the debris. Spent fuel, fission product, and activation product radionuclides were all expected to be found in varying amounts and this was indeed the case. The gamma spectroscopy work also provided assurance that once debris retrieval is complete, the equipment will be useful for quantifying the residual radioactivity levels in the PMC.

The RadScan®:700 unit proved successful in identifying areas of higher radiation and distinguishing different radionuclides present. Much of the debris on the cell floor and some of the legacy equipment showed readings significantly above that of the lower wall portions, indicating that once the debris is removed, the cell surfaces may not require substantial decontamination to achieve long-term performance goals. That success of obtaining reliable measurements of the walls also confirms the unit’s ability to quantify residual radioactivity after debris clean up is completed. Placement of the detector in high radiation areas did not affect its ability to detect hot spots, as long as the object or area being measured had a higher radioactivity level than the field around the detector. No degradation of the visual images or detector was in evidence despite a final accumulated dose of approximately 1000 R (10 sieverts). The fuel bundle shear and the area around it were found to exhibit the highest levels of gamma radiation, with the shear vent pipe shown to contain the next highest level. Since radiation levels were so significant, engineering work for future removal of this equipment has begun. Identifying and quantifying the gamma energies from different radionuclides was also achieved. Both Cs-137 and Co-60 were identified in the cell, with Cs-137 by far the prominent radionuclide. The high levels of Cs-137 found throughout the cell, however, may have prevented detection of other radionuclides possibly present at lower activity levels. A greater degree of success is expected in identifying other radionuclides in facilities with lower radioactivity levels and/or different radionuclide mixes.

The predominant radionuclides from a waste classification perspective were expected to be the spent fuel-related transuranics, Pu-238, Pu-239/240 and Am-241, and the laboratory analysis results confirmed their significance in all the waste streams sampled. The laboratory analysis also confirmed the different radiological distributions of the preliminary waste streams, and thus, that continued segregation of them was warranted. The leached fuel hulls contained the highest levels of all radionuclides. Activation products, mainly Co-60 and Ni-63, still remain at levels equaling or exceeding that of fission products in the leached hulls and fuel assembly hardware waste streams despite the approximately 30 years of radioactive decay. The loose contamination consisted of a mixture of radionuclides, and where accumulated in quantity, showed significant radiation levels. The fission product radionuclides consisted primarily of Cs-137 and Sr-90. Other radionuclides present in any significance but not critical to handling or classification included Fe-55, Ni-59, Nb-94, Tc-99, Eu-154, and Pu-241. The limited inorganic chemical testing that was performed also confirmed the process knowledge information and did not dictate a change in the packaging scheme.

## CONCLUSION

A phased characterization approach was successful in using available information on the radiological and chemical content of the PMC debris to develop a packaging scheme and to accelerate the start of packaging operations, followed by further nondestructive measurements and sampling and analysis to confirm the process knowledge. This approach can be utilized whenever the available information of waste contents is reasonably well known, and is especially useful in areas of high radiation and high contamination levels. The development of a remote means to collect radioisotopic information on the debris was also critical in minimizing the potential radiation exposure to personnel and the impact to the packaging schedule. This in-cell gamma spectroscopy tool is now available for use in future D&D activities at the WVDP, and similar equipment may be applied successfully elsewhere in the DOE Complex.