Safeguardability of A Commercial-Scaled ACP Facility

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Korea Atomic Energy Research Institute
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I. INTRODUCTION

The question of “how to manage the spent fuel discharged from reactors” is a key factor for the sustainability of nuclear energy. Approximately 6,000 metric tons of spent nuclear fuel from reactor operation has been accumulated in South Korea[i] with expectation of more than 30,000 metric tons, three times the present storage capacity (Table 1) by the end of 2040. The advanced spent fuel conditioning technology is designed to address these challenges, through the development of fuel cycle technologies coupled with advanced safeguards that will meet domestic and international nuclear materials management needs.

Since 1977, the Korea Atomic Energy Research Institute (KAERI) has been developing a pyro-chemical reduction process, called ACP (Advanced Spent Fuel Conditioning Process), to convert the oxide fuels into metallic form. The Electrolytic Reduction (ER) technology developed recently by KAERI is more economic and efficient concept for lithium reduction. The electrolytic lithium reduction process uses the metallic lithium dissolved in molten LiCl to reduce the oxide components of the spent nuclear fuel, yielding the corresponding metals and Li₂O. The metallic product will be collected separately and held in interim storage until its ultimate disposition is decided. Some fission products with high heat load such as cesium and strontium are dissolved in lithium chloride molten salt, and separated from the spent fuel product under ACP ER. The goals of the ACP are to recover more than 99.8% of the actinide elements and to minimize the volume and heat load of spent fuel to be

<table>
<thead>
<tr>
<th>Reactor</th>
<th>Capacity (Tons)</th>
<th>Storage (Tons)</th>
<th>Expected Saturation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Kori (4 units)</td>
<td>1,737</td>
<td>1,328</td>
<td>2008</td>
</tr>
<tr>
<td>Young-Gwang (6 units)</td>
<td>1,696</td>
<td>990</td>
<td>2008</td>
</tr>
<tr>
<td>Ulchin (6 units)</td>
<td>1,563</td>
<td>736</td>
<td>2007</td>
</tr>
<tr>
<td>Wolsung (4 units)</td>
<td>4,807</td>
<td>3,276</td>
<td>2006</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td><strong>9,803</strong></td>
<td><strong>6,330</strong></td>
<td><strong>-</strong></td>
</tr>
</tbody>
</table>

Table 1. Status of Spent Nuclear Fuel in Korea (As of 2003. 6.)
placed in interim storage and geological repository. Significant reduction of the volume and heat load of spent nuclear fuel by use of this conditioning technology could lighten the burden of final disposal in terms of disposal size, safety, and economics. The ACP concept will also enhance the long-term safety and economics of spent fuel disposal.

Although the main purpose of developing the ACP technology is for volume reduction in HLW, it also could be used for the next generation’s fuel cycle system, coupled with advanced safeguards that will meet international requirements for nuclear material management. Figure 1 shows main strategies for the development of a proliferation-resistant next-generation fuel cycle based on the ACP that is now under development by KAERI.

The success of the ACP will depend on a number of factors. One of key factor is “proliferation resistance,” and it will be judged by the manner in which it addresses the issue of proliferation. A joint study on the safeguardability of the ACP technology has been under way by the Los Alamos National Laboratory (LANL) and

Fig. 1. Proliferation Resistant Advanced Fuel Cycle Based on the ACP Technology in Korea.
system for the ACP. This report summarizes the preliminary results of joint research. The sub-processes and material flow of the pilot scale ACP facility were designed for this study. Then, their MBA (Material Balance Area) and KMPs (Key Measurement Point) were defined based on diversion scenario analysis. The limit of error in MUF value was also estimated with international target values for the uncertainty of measurement methods. Finally, the research and development ideas that should be pursued along with the development of process technologies were identified.
II. ANALYSIS OF ACP TECHNOLOGY

A. Main Process Concept

The ACP technology is based on the pyro-chemical process that was designed in the 1960s and 1970s. The pyro-recycle chemistry was originally developed specifically to yield a plutonium product that is inherently commingled with minor actinides (americium, curium, and neptunium), uranium, and fission products. The reference concept consists of six major sub-processes. These sub-processes are illustrated by the reference flow diagram shown in Fig. 2. They include;

- dismantling the fuel assemblies, cutting fuel rods, and removal of the Zircaloy cladding,
- thermal oxidation of UO$_2$ to U$_3$O$_8$,
- reduction of the oxide fuel into metals, using a suitable reductant in a molten salt,
- regeneration of the reductant metal by electrolysis of its oxide to allow recycling it and to minimize the waste generation,
- smelting of metalized fuel, and
- casting of metalized fuel in a form that is suitable for interim storage and deposition.

In the reference lithium reduction process, the oxide fuel elements are chopped into segments and are voloxidized, and the resultant oxide powder is loaded into a porous magnesia basket. The baskets are charged into a reduction vessel, where the fuel is reduced with lithium dissolved in molten LiCl at 650 °C. A layer of molten lithium floating on the surface of the salt maintains the lithium activity at unity. The different elements of the fuel (actinides and fission products) react with the lithium in various ways, depending on their chemical characteristics.

The actinide oxides, designated generically as MO$_2$, are reduced to metal according to the reaction:

\[ MO_2 + 4Li \rightarrow M + 2Li_2O \]  (1)
The reduced metal remains in the basket. The fission product oxides can be divided into four groups based on their reaction with Lithium. Fission products that form chlorides are classified as the “FPA” group. The alkali and alkaline earth elements are in this category. They are dissolved in the salt phase after being converted to their chlorides according to these reactions:

\[ \text{OLiCsClLiClOCs} \]
\[ \text{OLiSrClLiClSrO} \]

Also in the FPA group is europium, which behaves like the alkaline-earth metals and forms EuCl\(_2\). The rare earth (RE) elements’ behavior is complicated. Thermodynamically, they are not reduced and remain as oxides. However, some of the rare-earth oxides will interact with Li\(_2\)O according to the following typical equilibrium:

\[ \text{Nd}_2\text{O}_3 + \text{Li}_2\text{O} \rightarrow 2\text{LiNdO}_2 \]
The LiNdO$_2$ has limited solubility in the salt, but Nd$_2$O$_3$ is virtually insoluble. Thus, elements in the RE group are distributed between the basket and the salt. Fission products that react with Lithium to form salt-soluble compounds, such as LiI and Li$_2$Te, are designated as the “FPB” group. This group includes Te, Sb, Se, Br, and I. The balance of the fission products, the noble metal (NM) group, is reduced to metal along with the actinides and remains in the fuel basket.

Effective reduction is possible at temperatures as low as 500°C when lower melting point eutectics such as LiCl-KCl or LiCl-LiF are used. However, the LiCl-KCl combination proved incompatible with the mineral waste form and the LiCl-KCl combination was undesirable because of potassium vaporization at 525°C. In addition, the solubility of Li$_2$O in LiCl-KCl is very low at 500°C. These considerations lead to the selection of LiCl as the process salt and 650 °C as the process temperature. While the solubility of the Li$_2$O in LiCl at 650 °C is 8.7 wt%, it was reported that the Li$_2$O concentration must be kept below 3 wt% to obtain efficient actinide reductions [ii]. The reason is thought to be the equilibrium reaction:

$$2Pu + 3Li_2O \rightarrow Pu_3O_8 + 6Li$$  

(5)
To keep the Li₂O concentration at an acceptable value and to recover the Lithium for reuse after the reduction step, the Li₂O is electrochemically decomposed to liberate oxygen and Lithium:

\[ \text{Li}_2O \rightarrow 2\text{Li} + \frac{1}{2}O_2 \]  \hspace{1cm} (6)

This process is known as electro-winning. Equation (6) is the net result of the following two reactions; one is taking place at the anode and the other at the cathode:

\[ O^{-2} \rightarrow \frac{1}{2}O_2 + 2e^- \] \hspace{1cm} (Anode) \hspace{1cm} (7)

\[ 2\text{Li}^+ + 2e^- \rightarrow 2\text{Li} \] \hspace{1cm} (Cathode) \hspace{1cm} (8)

The degree of reduction of the TRU elements is a primary concern because of the goal to recover greater than 99% of the actinides in the PWR spent fuel. Table 2 is a summary of the results achieved in many reduction experiments for the Li reduction concept [iii].

The electro-winning step also serves to reduce the concentration of FPB and RE fission products dissolved in the salt. The rare-earth oxides that dissolved in the salt during the reduction step precipitate as the Li₂O concentration is reduced. These precipitated oxides are filtered from the salt before the recovered salt is returned for

<table>
<thead>
<tr>
<th>Oxide Reduced</th>
<th>Reduction Yield</th>
</tr>
</thead>
<tbody>
<tr>
<td>UO₂</td>
<td>100.000</td>
</tr>
<tr>
<td>PuO₂</td>
<td>99.996</td>
</tr>
<tr>
<td>NpO₂</td>
<td>99.998</td>
</tr>
<tr>
<td>AmO₂</td>
<td>99.982</td>
</tr>
<tr>
<td>CmO₂</td>
<td>99.986</td>
</tr>
</tbody>
</table>
use in another reduction step. The FPA group fission products are unaffected and remain with the salt. The concentration of these fission products is controlled when it reaches the design limit by discarding a small portion of the salt as part of the mineral waste form.

Based on the reference technology, a modified concept was proposed by KAERI to simplify the process and to increase the reduction performance. In the electrolytic reduction (ER) concept, the lithium electro-winning is conducted in the uranium oxide cathode simultaneously and there is no process for salt recovering. As shown in Fig. 3, the lithium is produced electrolytically at the uranium oxide cathode and this lithium reduces oxides in spent fuel to metal in the new concept. Consequently, the possibility of separate reduction of actinides is technically eliminated in this concept, and lithium recovery process is no longer needed.

B. Facility Design

1. Facility Concept

A pilot scale ACP facility with a capacity of 30 MTHM/year was designed in this study to analyze the safeguardability of the ACP facility. The facility stands alone physically (operationally), and is administratively isolated from reactors and interim spent-fuel storage facilities. The main process of the facility is assumed to be the electrolytic reduction concept, shown in Fig. 3, which has no need of a Lithium recovery system. The facility availability is assumed 60%, which is equivalent to 219 full operating calendar days per year. The process consists mainly of three parts: spent fuel handling area (spent fuel disassembling and rod extraction), main hot cell (decladding, reduction, smelting, casting, etc.), and U-metal handling area (loading metal rods into storage cask and temporary storage).

The reference fuel used in the ACP facility is Korean Yong-Gwang Unit 1 & 2 PWR's 17×17 standard spent fuel assemblies with a minimum 10 years of cooling time after discharge from the reactor. The design parameters of the reference fuel are shown in Table 4.
2. Process Flow and Description

The ACP consists of several process steps as shown in Fig. 4. Various solid wastes will be generated in the process, and all wastes containing nuclear material will be managed for safeguarding.

1. Disassembling: The spent PWR fuel disassembling is performed in dry cells or in a pool using commercially available technology. Some non-fuel bearing structural components are generated, and a volume reduction process of structural components is added.

2. Rod cutting: A conventional mechanical process such as shear cutter is used and approximately 0.1 wt% of cutting wastes is generated.

3. Decladding: A conventional mechanical slitting technology is used for decladding. The cladding hull is generated with 0.1 wt% of nuclear material is embedded in the hull.

4. Voloxidation: The purpose of this process is to increase the reduction rate in consequent process. UO$_2$ powder is converted to U$_3$O$_8$ with an oxidation temperature of 500°C.

5. Reduction: This reduction process is the key process in this technology. The U$_3$O$_8$ powder is reduced to a metallic form with 99.8% reduction yield for typical actinides. Further study to decide optimum operating conditions that maximize the reduction rate and minimize the process time shall be required. The generated salt waste can be recycled several times.

<table>
<thead>
<tr>
<th>Table 3. Basic Specifications of Pilot Scale ACP Facility.</th>
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<tbody>
<tr>
<td><strong>Target fuel</strong></td>
</tr>
<tr>
<td>Korean Yong-Gwang Unit 1 &amp; 2 PWR's 17×17 standard spent fuel assemblies with a minimum of 10 years of cooling time after discharge from the reactor.</td>
</tr>
<tr>
<td><strong>Facility throughput</strong></td>
</tr>
<tr>
<td>30 MTHM/y (approximately 0.36 MT-Pu/y)</td>
</tr>
<tr>
<td><strong>Facility availability</strong></td>
</tr>
<tr>
<td>60% plant production availability (i.e., equivalent 220 full operating calendar days/year)</td>
</tr>
<tr>
<td><strong>Impurities in product</strong></td>
</tr>
<tr>
<td>2 wt% of products (low decontamination product)</td>
</tr>
<tr>
<td><strong>Main process of the facility</strong></td>
</tr>
<tr>
<td>Electrowinning reduction process (reduction rate: 99.8% for all actinides)</td>
</tr>
</tbody>
</table>

2. Process Flow and Description

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6. Smelting/Casting: Smelting and casting processes are required to get a form of the 20-kg metal rod that is appropriate for storage. Approximately 0.2 wt% of waste and dirty scrap (rejected ingots) is generated in this process. The clean scrap is treated for recycling.

7. Loading metal rod into storage cask: A storage cask for the metal rod is provided and stored in the temporary dry storage area. After that, metal casks can be transported to interim storage area.

<table>
<thead>
<tr>
<th>Table 4. Typical Specifications of Reference Spent Fuel.</th>
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<tbody>
<tr>
<td><strong>Final burnup</strong></td>
</tr>
<tr>
<td><strong>Specific burnup</strong></td>
</tr>
<tr>
<td><strong>Nominal enrichment</strong></td>
</tr>
<tr>
<td><strong>No. of spacer grid</strong></td>
</tr>
<tr>
<td><strong>Pellet density</strong></td>
</tr>
<tr>
<td><strong>Total weight of fuel assembly</strong></td>
</tr>
<tr>
<td><strong>U-weight/assembly</strong></td>
</tr>
<tr>
<td><strong>No. of fuel rods/assembly</strong></td>
</tr>
<tr>
<td><strong>Pellet OD</strong></td>
</tr>
<tr>
<td><strong>Active length</strong></td>
</tr>
<tr>
<td><strong>Cladding OD</strong></td>
</tr>
<tr>
<td><strong>Cladding ID</strong></td>
</tr>
<tr>
<td><strong>Cladding Thickness:</strong></td>
</tr>
</tbody>
</table>
**Advanced Spent Fuel Conditioning Process (ACP)**

**Fig. 4. Process Flow of Pilot Scale ACP Facility**
III. DIVERSION SCENARIO ANALYSIS

The existing “open” or “once through” LWR fuel cycle is relatively proliferation resistant compared to closed cycles. As long as the fuel assemblies remain intact, the safeguards approach is straightforward, but may be resource intensive for large numbers of assemblies. If the assembly is broken down, proliferation risk increases from loss of identity of the assembly and the huge numbers of fuel pins that must be tracked. Breaking open pins to process the fuel introduces significant proliferation risk from bulk-handling operations. Any closed fuel cycle is likely to present an increase in proliferation concerns, and bulk-handling operations are a perfect diversion location because of the availability of the material and reliance upon materials accounting for detection. Material accounting must be the most sophisticated for these processes, and even then it may not be able to detect diversion for large processing plants. However, not all nuclear facilities are equally susceptible to proliferation purposes nor are they all equally easy to safeguard. Intrinsic factors influence both the attractiveness of materials/facilities to proliferators and their safeguardability [iv].

A basic premise of SNM safeguards is that all nuclear materials, regardless of how relatively attractive, are candidates for diversion. Any separation process can be modified, in fact, to separate an impure stream and avoid detection by measurement systems of any diversions, or to produce a pure or nearly pure fissile material stream. The IAEA will consider the means by which the State or the facility operator may divert material. The diversion hypotheses must be taken into account and defined for each reprocessing facility. Irradiated LEU fuel assemblies are a source of Pu and must be accounted for as such. This is typically referred to as “diversion path analysis.” Means must be identified to show first that a process operates as declared and second that a process cannot be modified without detection, even when an inspector is absent.

A. Diversions from the ACP

The details of an ACP flow sheet are highly speculative. Consequently, additional safeguards must be evaluated when options are narrowed and process flow sheets are fully developed. As a pragmatic scenario, we assume that the spent PWR fuel being processed at the ACP facility is out of the reactor core for approximately 10 years.
The Lithium Reduction process uses lithium metal dissolved in molten LiCl to reduce the oxide components of the fuel, yielding the corresponding metals and Li$_2$O. Most of the actinide oxides will be reduced to metal in the process. Alkali and alkaline-earth fission products are converted to chloride in the reduction step, whereas rare-earth fission products remain in oxide form. Some part of rare-earth fission products as well as alkali and alkaline-earth fission products remain in the salt, but other fission products including noble metals are reduced along with the actinides. There are several inherent attributes of the ACP process that make this fuel cycle unattractive for diversion when compared with conventional fuel reprocessing and plutonium recycling.

- The processes used for the ACP do not produce a pure or partially pure plutonium product. Because of the chemistry of the lithium reduction process, no fissile material can be separated in pure form. Plutonium is co-deposited together with minor actinides and some fission products. Therefore, the material requires further chemical reprocessing to separate pure fissile elements. This results in longer warning times and requires development of a technology standard in order to obtain material suitable for weapons purposes.

- The reconstitution options require a highly remote operation in canyons of highly shielded cells. It is difficult to gain undetected access to these cells to modify hardware or install new processes. The complexity of these operations with highly radioactive materials precludes manual operation. The process must be highly automated with inherent abilities to track and log in-cell operations included in the design. Incorporation of this information into a safeguards system will, therefore, not interfere with plant operations. The tracking and logging data could provide information on installation of diversion-related equipment in the processing cell and material movements within the cell that would indicate that diversion is in progress.

- The decay heat and radioactivity of the ACP product (metallic form of the spent fuel) are about 25% of those of the initial spent-fuel feed to the ACP. As shown in Fig. 5, the presence of some fission products leads to a high dose rate of radiation arising from the metal product. Furthermore, some of the processes used in the ACP require high-temperature furnace operations under controlled atmospheres even in the heavily shielded hot
Unauthorized penetrations into the processing cell to divert small quantities of materials using a clandestine material stream on a protracted basis would be extremely difficult and readily detectable. This feature may be concordant with the PIPEX concept as was proposed during the INFCE [v]. Similar observations have already been made for the IFR fuel cycle, where the material is considered as "self-protecting" for the reasons mentioned above [vi]. Nevertheless, it is needed to mention that the self-protecting is not valid when the State is the adversary.

- The ER process operates in a batch mode that simplifies material accounting compared to that for continuous flow systems like the PUREX. Batch processing combined with appropriate material sampling and physical security will provide greater assurance against theft.

- The most probable approach to final plutonium purification would be through conventional technologies, such as aqueous ion exchange or solvent extraction. However, these processes would have to be conducted outside of the main processing cell because they involve aqueous reagents that are incompatible with the cell design. Because of the high
radiation field associated with all plutonium-containing streams of the ACP, the purification operation would require a heavily shielded cell. Such installations within the ACP facility, outside the shielded cells, could be readily detected during routine inspections by the IAEA. The removal of diverted SNM from locations scheduled for the ACP materials would be relatively easy to detect because of the high radiation emissions.

- The possibility that normal materials in the bulk processing areas could be diverted in small quantities (during each material balance period) and processed off-site to separate plutonium cannot be ignored. If this off-site location were an undeclared facility, detection of such operations during routine inspections would be unlikely. Such diversions are detectable using special analysis methods that are available to detect protracted diversions. However, these detection probabilities do depend on the quality of measurement methods used for materials accounting and verifications.

- Another possibility for diversion is through the misuse of the accountancy system. A plausible scenario is to divert materials from desirable material inventories and conceal the diversion in MUF by overstating the SNM contents of undesirable materials, such as discard. For example, a common diversion path for reprocessing facilities is to divert material into the hulls stream and misstate the amount of material in the hulls. Such a diversion scenario could be detected using a well-designed verification regime that will adequately verify both the desirable and undesirable materials at the facility.

A single-act (abrupt scenario) diversion can occur in which enough material is diverted to meet the goal of the diverter, or a series of acts (protracted scenario) may occur. Each act in protracted diversion diverts material in quantities below the requirements for the ultimate goal of the nation intent on proliferation. Abrupt scenario is a diversion that can, in principle, occur at any point in the system: at the reactor facility, during off-site transportation, or at the fuel recycling facility. Protracted diversion is most likely to occur at the fuel conditioning facility. These two diversion strategies are discussed below.
1. Abrupt Diversion Scenarios

This discussion is limited to plausible diversion scenarios at the ACP facility or during transportation to and from the facility. The operator could conceivably abruptly divert a significant quantity of plutonium-bearing material in the following forms:

A. Two or more whole assemblies of spent PWR fuels.
B. Approximately 500 spent PWR fuel rods with a substitution of dummy rods for the plutonium bearing rods.
C. Approximately 1 Mt or more of materials in the process line and replacement of them with uranium.
D. Approximately 30 or more fabricated U-metal rods.
E. Some combination of items described above.

It is possible to design safeguards systems to detect such diversions with a very high probability and on a timely basis. The spent PWR fuel assemblies received at the recycling facility and U-metal rods produced at the facility are accounted for as “items” under item accountability. Each assembly or bundle is given a unique identity and a tamper-resistant seal in this approach. The assembly or bundle is tracked as an “item” through the system. Because the U-metal rods are too numerous, it is possible to use groups of storage baskets with seals. Diversion of such items will be readily detected with high probability during IAEA inspections that would include a well-designed sampling plan, item counting, seals verification, and an attribute measurement scheme to detect gross defects.

Other plausible abrupt diversion scenarios in the ACP concept occur during transport of spent PWR fuels to the conditioning facility or during the transport of U-metals to the storage site. However, this is not a very likely scenario because of the existing protocols of transportation across nuclear facilities and the consequent ease of detection.

2. Protracted Diversion Scenarios

We examine the conceptual flow sheet of the ACP facility to see how the process could be modified to make diverting material less easy to detect. We begin with the facility design and analyze several diversion paths to see what would be required to detect surreptitious removal of materials. The diversion paths for all plant materials
associated with significant amounts of highly radioactive fission products should include the ability to transfer materials using remote operations and to circumvent surveillance by radiation monitors. A well-designed material accountancy and verification system is expected to be able to detect protracted diversion as well as data falsifications. Typically, this means use of best available measurements for all materials entering and leaving the facility, as well as any materials expected to be present at the time of physical inventory verifications. At least two, and often three, tiers of measurements methods could be applied to detect “gross,” “partial,” and “bias” defects. The best way to achieve the detection goals, and to achieve the best safeguards overall, is to design in safeguards during the process and facility design. Those features that best promote safeguards are: ability to make good measurements, continuous knowledge of the amounts and locations of materials, and building and process design that promotes containing the material to easily controlled and monitored areas and pathways.

It should be noted that international safeguards only applies to the detection of proliferation by a State. They are not intended to deal with theft of materials by facility employees or by sub-national groups (terrorists), although, as practiced to date, they have some limited benefits in this area as well. Any material determined to be missing is considered an indication of proliferation under international safeguards. The State is expected to deal with sub-national threats through the application of its own domestic safeguards and security.

B. Material Diversion Path

Once material that containing plutonium has been identified and prepared for diversion, the next step is to remove it from the facility. Assuming that this facility is not continuously inspected, independent-accountancy-data recording systems, intelligent containment and surveillance systems capable of recording unauthorized material movements, etc., can effectively limit diversion paths.

Facility design verifications performed by the IAEA before plant startup are useful to evaluate material diversion paths. Because the facility design and plant layouts of the ACP facility are still to be determined, it is difficult to describe all clandestine material diversion paths accurately. Nonetheless, some general comments could be made that apply to any fuel reprocessing facility of this type. Possible diversion paths include the following:

• Unauthorized cell penetrations
• Unauthorized use of existing cell penetrations
• Diversion through the equipment maintenance cells
• Diversion through waste streams
• Diversion of normally produced material with a substitution of new materials or waste
• Diversion through normal channels of material movement

These and other diversion paths may be detected by any number of safeguards methods and by the failure of the facility to meet the expected production rate.

C. Use of a Clandestine Facility

Nuclear materials produced in an ACP facility as a normal product contain both major amounts of trans-uranium actinides (predominantly neptunium, americium, and curium) and considerable fission products. These contribute significantly to the radiation level of the product. Purifying the plutonium requires either a heavily shielded supplemental facility on-site or an off-site facility. Additionally, heavily shielded carriers (casks) would be needed to remove the material from the conditioning facility.

Materials diverted from ACP facility could be processed through a simplified PUREX cycle or its equivalent to remove the remaining fission products and uranium to recover plutonium. Because IAEA inspectors have access to many areas on-site, it is unlikely that such a process could be operated within the ACP facility could be operated without detection. By keeping track of all nuclear materials within the ACP facility using material accountancy, containment and surveillance, and routine inspections by the IAEA, one could detect significant diversion of feed materials to operate a clandestine facility operating to recover the plutonium.
IV. SAFEGUARDS SYSTEM ANALYSIS

The ACP facility does contain direct-use material but with an elemental composition essentially similar to spent fuel. The product, U-metal, will continue to bear the essential characteristic of being self-protecting because of its radioactivity. A reprocessing facility would be required to recover pure plutonium from any process material that is diverted from the ACP. Consequently, the timeliness criterion that the IAEA assigns to spent fuel, detection of diversion of one significant quantity (8 kg of plutonium) within three months, is appropriate for the ACP. A diverter would have to remove approximately 1 Mt of spent PWR fuel materials - or materials contained in two or more PWR assemblies.

It is claimed that effective safeguarding of pyro-processing plants would be more difficult than for conventional plants because it is more difficult to measure and keep track of the fissile materials in the process. Therefore, the KAERI and LANL have been working on methods for the quantitative analysis of ACP materials, especially in salt and waste matrices.

Establishing a material inventory requires a bulk measurement (volume or mass) and the measured concentrations of the elements in samples taken from the bulk. The bulk measurement determines the mass of the initial spent fuel and of the electrodes. Sampling of the solidified electrolyte or metal cathode is more difficult than sampling in aqueous processes. Because we expect that the ACP material may suffer from inhomogeneity, a higher number of samples are recommended for analysis. Either of two well-established methods of analytical chemistry, isotope dilution mass spectrometry (IDMS) or inductively coupled plasma mass spectrometry (ICP-MS), can be applied for the assay of these elements. Chemical preparation of the samples will be required for either analytical method. This purification, which is necessary to remove the matrix (i.e., non-SNM components of the item) and to obtain pure fractions in order to avoid isobaric interferences is relatively labor intense and requires a careful study of the chemical recoveries of the elements of interest.

Americium and curium can be measured by IDMS using $^{243}$Am and $^{248}$Cm as spikes. IDMS, however, cannot be applied to neptunium because there is no appropriate spike isotope. ICP-MS can be applied to all the Minor Actinides (MA) elements. However, separation chemistry (for example, ion exchange, extraction chromatography, or High Performance Liquid Chromatography (HPLC)) is still
required to avoid isobaric interferences. Radiometric methods may be most
expedient for the assay of the MA in samples from pyro-processing because efforts
required for sample preparation can be reduced to a minimum. Development work in
this context is ongoing at KAERI, in order to systematically study and establish the
analytical methodology.

Quantitative assay of U, Pu and the MA using non-destructive methods (i.e.
where samples from the process are measured without further treatment) may suffer
from severe matrix effects. Nevertheless, non-destructive techniques might be
applicable for measuring the spent fuel and for monitoring individual process
parameters.

These inventory measurements need to be complemented by the traditional
techniques of containment and surveillance, by verification of design information,
and by monitoring of essential system parameters. If continuity of knowledge is lost,
two analytical approaches can verify that essential system parameters have not been
altered with the intention to obtain pure products (Pu or Np). The first is that swipe
samples taken from inside a hot cell could be checked by particle analysis methods
for the presence of high purity materials. The second is that the presence of pure Pu
or pure Np on the electrode would indicate an anomaly. Only elemental analysis
would be required for the verification in either case.

A. Non-Destructive Assay Systems

Materials present in ACP have or will most likely have characteristics that are
extreme compared to materials that are currently measured. Spontaneous fission of
$^{242}\text{Cm}$ and $^{244}\text{Cm}$ is the major source of neutrons emanating from spent fuel. These
isotopes are produced through multiple neutron capture events when a fuel assembly
is exposed to the high neutron flux in a nuclear reactor. Fission products in the
irradiated fuel produce an extremely high radiation background in which the
neutrons must be detected. The high radiation environment influences the type of
techniques that can be deployed for spent fuel verification. One approach is to
choose a detector that is insensitive to $\gamma$-rays. Another approach is to shield against
the $\gamma$-rays while allowing neutrons to pass through the shield into the neutron
detector.

Spent fuel verification methods include not only neutron detection but also $\gamma$- ray
and ultraviolet light (Cerenkov radiation) detection. Table 5 lists the spent fuel
measurement systems in use by the IAEA. The Fork Detector (FDET) incorporates
both neutron and $\gamma$-ray detectors for gross defect verification of fuel assembly characteristics such as irradiation history, initial fuel content and number of reactor cycles of exposure. Detector systems are available to measure the $\gamma$-ray energy spectra from irradiated fuel (SFAT and IRAT), and $\gamma$-ray intensity as a function of fuel bundle storage position (CBVB and CBVS). Cerenkov glow viewing devices (ICVD and eventually a digital device, DCVD) examine the ultraviolet light that appears in the water surrounding spent fuel.

Advances in measurement technology for the ACP will be needed to measure plutonium and other fissile materials in process matrices that include high fission-product activity, high neutron flux, high attenuation or moderation, and extremes in

<table>
<thead>
<tr>
<th>Code</th>
<th>Equipment Name</th>
<th>Description and Applications</th>
</tr>
</thead>
<tbody>
<tr>
<td>FDET</td>
<td>Fork Detector Irradiation Fuel Measuring System</td>
<td>Detector system that straddles LWR fuel assemblies with pairs of neutron and $\gamma$-ray detectors. Gross $\gamma$-ray and neutron intensities and ratios of intensities can give specific information on the fuel assembly.</td>
</tr>
<tr>
<td>SFAT</td>
<td>Spent Fuel Attribute Tester</td>
<td>Gross defect device used for verifying the presence of fission product or activation product at the top of the irradiated fuel assembly.</td>
</tr>
<tr>
<td>IRAT</td>
<td>Irradiated Fuel Attribute Tester</td>
<td>Gross defect device used for verifying fission product presence in an irradiated fuel assembly.</td>
</tr>
<tr>
<td>ICVD</td>
<td>Cerenkov Viewing Device</td>
<td>Hand-held light intensifying device optimized to view Cerenkov light (near ultraviolet) in a spent fuel storage pond. System can be used in a lighted area. Primarily used to identify irradiated LWR fuel assemblies.</td>
</tr>
<tr>
<td>DCVD</td>
<td>Digital Cerenkov Viewing Device</td>
<td>Highly sensitive digital device for viewing Cerenkov light from long cooled, low burnup fuel.</td>
</tr>
<tr>
<td>CBVB</td>
<td>CANDU Bundle Verifier for Baskets</td>
<td>Attended radiation-monitoring systems that scan storage baskets or stacks of CANDU fuel bundles and record gamma intensity as a function of detector position.</td>
</tr>
<tr>
<td>CBVS</td>
<td>CANDU Bundle Verifier for Stacks</td>
<td>Facility specific system used to make high-resolution $\gamma$-ray measurements of spent fuel assemblies. Collimator in front of the Ge detector is built into the facility.</td>
</tr>
<tr>
<td>GBUV</td>
<td>Gamma Burnup Verifier</td>
<td></td>
</tr>
</tbody>
</table>
isotopic composition. The materials may be quite heterogeneous and may be contained in processing and containment vessels that are themselves highly attenuating or force measurements to be made from some distance. Measurement uncertainties will need to be approximately 0.5% or less to meet today’s measurement standards.

Design requirements of the NDA system for the ACP facility are summarized as follows:

- The NDA system must measure the solid type of metal ingot, powder and salt waste.
- The NDA system should be a radiation-hard system, so it would not contain any weak material for radiation (especially gamma-radiation).
- The NDA system must accurately determine Pu mass despite variations on the reduction rates of corresponding actinides during the metallization and voloxidation processes.
- The assay chamber must accommodate rod cuts and metal ingots. The rod cuts have a diameter of 0.95 cm and length of 25 cm, and the metal ingot has a diameter of 15.0 cm and a length of 10 cm with 20.0 kg mass. The neutron efficiency must be uniform over at least 25.0 cm along the sample cavity.

Details of technical concerns for NDA analysis on ACP materials and the NDA system design process are described in Appendices A and B.

**B. Destructive Chemical Assay Systems**

Destructive measurements must assay elements and determine isotopic composition of all solid and liquid types encountered in bulk handling nuclear plants. DA is used in as follows [vii];

(a) To verify that protracted diversion of safeguarded nuclear materials has not occurred.

(b) To certify working standards used for the calibration of NDA and installed verification instruments.

(c) To provide assurance of the quality and independence of on-site measurements (e.g. validation of facility specific procedures.)

(d) To carry out periodic verification of the operator’s measurement system.

DA verification measurements incorporate the following steps:
(a) Taking independent samples.

(b) Conditioning the samples at the facility to ensure that their chemical forms are adequate for maintaining their integrity during transport.

(c) Packaging, sealing and shipping the conditioned samples to facility’s analytical laboratory or the Agency’s Safeguards Analytical Laboratory (SAL.)

(d) Analyzing the samples at analytical laboratory or the Network of Analytical Laboratories (NWAL), consisting of laboratories in different States that have been certified to analyze safeguards relevant samples.

(e) Performing the statistical evaluation of the results of their analysis of samples.

It is necessary to apply optimized and validated procedures for each of these steps to obtain meaningful and accurate results. The DA measurements are used domestically for material accountancy, and used by the IAEA independently to verify the operators’ declarations.

1. Isotope Dilution Mass Spectrometry

Isotope dilution mass spectrometry is applied for U or Pu determinations in all samples of spent fuel input materials, but also for samples of low content, such as milligram size U–Pu samples and wastes.

The determination of U and/or Pu determinations in high-burnup spent fuel input materials uses an aliquot of the sample solution spiked with a known amount of a certified tracer containing enriched $^{235}$U and $^{239}$Pu. A spike of $^{233}$U is used for pure U materials; a spike of $^{242}$Pu or $^{244}$Pu is used for pure Pu materials or low-burnup spent fuel. Spiked solutions of Pu-bearing materials are chemically treated to attain an isotopic equilibrium of Pu. Two spiked aliquots and an unspiked aliquot are separately purified by chromatography in order to provide pure fractions for thermal ionization mass spectrometry. The chemical treatment of spent fuel samples is performed with a fully automatic, robotized system. The resulting U and Pu fractions are then evaporated to dryness and re-dissolved in nitric acid to yield solutions containing about 1 mg U and 50 ng Pu per microliter. The isotopic ratios of both the spiked and unspiked aliquots are measured by thermal ionization mass spectrometry, and the U and Pu contents are calculated accordingly. When the original sample can be spiked directly and total evaporation mass spectrometry measurements are done, the elemental assays have a precision and accuracy of 0.1% rel. or better.
2. Isotopic Composition by Thermal Ionization Mass Spectrometry

Thermal ionization mass spectrometry, employing three multi-detector mass spectrometers, each equipped with nine Faraday cups, is used to measure the U or Pu isotopic composition of all samples of nuclear materials. Comprehensive operation software includes routines for basic calibration steps such as the cup linearity test, relative cup efficiency measurements and a system calibration of mass fractionation effects. Isotope ratios of 0.05~20 can be measured with a precision and accuracy of 0.05%rel. using a data-collection procedure involving total evaporation of the sample loaded on the filament. This procedure greatly reduces the mass fractionation effects.

C. Containment and Surveillance Systems

Measurements alone are not likely to satisfy verification requirements for the ACP facility. Containment and surveillance techniques are extensively used by the IAEA because they are flexible and cost effective. The two main C/S categories are optical surveillance and sealing systems.

Optical surveillance is most effective in storage areas (such as spent fuel storage ponds) with relatively few activities that could be interpreted as the removal of nuclear material. A typical application would consist of two or more cameras positioned to cover the storage area completely. The field of view of the cameras is such that any movement of items that could be the removal of nuclear material is easily identified. This means that items must be sufficiently large within the field of view to be identified and that one or more images must be recorded during the movement of material. The image recording may be set at a periodic frequency (significantly shorter than the fastest possible removal time) or the motion (i.e. scene change) may trigger the recording. Optical surveillance is intrinsically an unattended operation that may be enhanced by the remote transmission of image data or system operation data (i.e. the status of the surveillance system).

Seals are typically applied to individual items containing nuclear material. A seal can help to indicate that material was neither introduced into nor removed from a container. At the same time, sealing provides a unique identity for the sealed container. Unattended IAEA monitoring equipment is also sealed. Most IAEA seals are applied for extended periods of time, typically several months to years. Seals may be either single-use seals that are replaced when checked or seals that are verifiable in situ (i.e. they can be checked for integrity and identity in the field). If
the seals are verifiable *in-situ* then the verification activity must be efficient (to limit radiation exposure to the inspector) and extremely reliable. The *in-situ* verification activity must consist of checking the item containment, the seal integrity and the method of attachment of the seal to the item.

A variety of monitoring systems must complement measurements in the ACP facility. These must track the fissile material, follow the actions of people and equipment, and determine that the process is being used appropriately. Some processes can be used for undeclared purposes. Any separation process can be modified to separate an impure stream, either to avoid detection from measurement systems or to produce a pure or nearly pure fissile material stream. Many processes claimed to be proliferation resistant can be modified in the absence of inspectors to produce materials for diversion to a weapons program. Proliferation resistance requires, first, that a process operates as declared and, second, that it cannot be modified without detection, even when the inspector is absent.

Except for NDA measurements of key parameters, the equipment for safeguarding the facility based on C/S should not interfere with production operations. Some of the C/S systems useful at the ACP facility include the following:

- Cameras or other optical monitoring devices for monitoring removal of assemblies from the facility.
- Monitoring for the preparation and packaging of the material.
- Radiation monitors on penetrations through which the material is moved.
- Position-sensitive detectors to monitor movement of radioactive materials.
- Radiation monitors in corridors
- Cameras or other optical devices in corridors
- Integrated systems of radiation monitors, process monitors, and video cameras for the continuity of knowledge.

Judicious use of advanced C/S techniques integrated with a well-designed material accountancy system could provide adequate safeguards for the ACP facility.
V. MATERIAL CONTROL AND ACCOUNTABILITY

Lacking specific design information for the ACP facility, the features such as the material balance area (MBA) definition, material flow pattern, key measurement points (KMPs), and inventories on material balance closing were designed for the conceptual facility. The assumptions necessary to calculate the detection sensitivity of the operator’s materials accounting system are presented below. Details of material flows, materials inventories, and material measurements, which determine the detection sensitivity, are also given. Figure 6 identifies MBA boundaries, KMPs, and locations of inventories at material balance closings. Basic specifications of the conceptual ACP facility to be evaluated with respect to safeguardability are summarized in Table 3. It is assumed that this facility operates 220 days/calendar year and that the facility closes material balances once every 3 months (or once after 54 days of operation). It is also assumed for this analysis that the present IAEA detection goals for spent LWR fuels would apply to materials within the ACP facility. Nuclear material contents for material balance were calculated based on the reference fuel (Table 4) and the material contents as summarized in Table 6.

A. Material Flow and Safeguards System

Figure 6 is an abbreviated representation of the major processes of this conceptual ACP facility. As shown in this illustration, the fuel conditioning facility is composed of two MBAs. The operations of MBA-A are based on individual item counts because the composition is not varied and items are only broken into other discrete items. Material accountancy in the MBA-A is similar to that in any storage area and is not described further in this report.

In MBA-B, the facility operator does material accounting based on some declared values for feed materials; destructive chemical analyses for mixed oxides and metal ingots; and NDA measurements for U-metals, recyclable scraps, and disposable waste streams. The nuclear material changes size, shape and chemical form in the ACP process, so a more sophisticated material accountancy method is required. Two types of material balance concepts are employed: batch closeout, which is the inventory difference for a single process, and material accountancy, which is an inventory difference in a specified time interval over several critical
The batch closeouts have two different steps based on available information. First, a mass balance is performed based on the total weights of the materials that enter and leave a piece of equipment during a batch. This balance must meet a specified accuracy or operations are halted to investigate possible sources of error. The check
provides the assurance that operations proceeded as planned and the inventory difference from the measured weights lies within expected limits. After analytical chemistry results are received, a second batch closeout is performed, which checks expected and measured compositions. The expected masses and compositions of new items are based on operational models and prior experience. This two-step closeout provides the best data for every item in the MBA-B. On-line mass tracking computer system is implemented that reads and stores the item weights and compositions, because items will disappear or be created, change their masses and compositions at different times between inventories. This system provides a model of discrete accountable items distributed in space and time and constitutes a complete historical record.

The batch-type process readily supports near real-time accounting (NRTA). Movement of material is controlled remotely with movements and weights recorded in real time by the NRTA system. When material is moved from one process step to the next, it is moved as a discrete mass in a labeled container and weighed before shipment from one station and after receipt at the next station. Waste and scrap are also handled as discrete and weighted items. The basic assumptions necessary to

<table>
<thead>
<tr>
<th>Isotopes</th>
<th>g/initial MTU</th>
<th>wt% of HM</th>
</tr>
</thead>
<tbody>
<tr>
<td>U-234</td>
<td>3.51E+00</td>
<td>0.018</td>
</tr>
<tr>
<td>U-235</td>
<td>1.28E+02</td>
<td>0.671</td>
</tr>
<tr>
<td>U-236</td>
<td>9.60E+01</td>
<td>0.502</td>
</tr>
<tr>
<td>U-238</td>
<td>1.86E+04</td>
<td>97.541</td>
</tr>
<tr>
<td>Np-237</td>
<td>1.25E+01</td>
<td>0.065</td>
</tr>
<tr>
<td>Np-239</td>
<td>3.66E-06</td>
<td>0.000</td>
</tr>
<tr>
<td>Pu-238</td>
<td>5.10E+00</td>
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<tr>
<td>Pu-239</td>
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<td>0.614</td>
</tr>
<tr>
<td>Pu-240</td>
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</tr>
<tr>
<td>Pu-241</td>
<td>2.05E+01</td>
<td>0.107</td>
</tr>
<tr>
<td>Pu-242</td>
<td>1.64E+01</td>
<td>0.086</td>
</tr>
<tr>
<td>Am-241</td>
<td>1.36E+01</td>
<td>0.071</td>
</tr>
<tr>
<td>Am-243</td>
<td>4.26E+00</td>
<td>0.022</td>
</tr>
<tr>
<td>Cm-242</td>
<td>5.36E-05</td>
<td>0.000</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td><strong>1.91E+4</strong></td>
<td><strong>100.00</strong></td>
</tr>
</tbody>
</table>
design the material accounting system are as follows:

- IAEA verification is preceded by a shutdown and cleanout of all major process areas and the accumulation of inventories at a few locations shown in Fig. 6 as KMPs.
- IAEA verification will employ attributes and variables measurements, preferably using NDA measurements.
- The facility closes material balances once every three months and plans to have the IAEA inspections coincide with this schedule for plant shutdown, cleanout, and material balance closing.
- The large inventories of feed materials and products (MBA-A and KMP-J in Fig. 6) are maintained as “items” for inventory purposes and are stored in separate storage locations.
- The cleanout operation before material balance closing recovers almost the entire residual process holdup. Therefore, inventory of plutonium as process holdup is negligible.

B. Key Measurement Points

The KMPs associated with both material flow and inventories at the conceptual ACP facility are illustrated in Fig. 7. The flow KMPs are represented by numbers and the inventory KMPs are designated by letters. Some details of flow and inventories at the bulk counting area (MBA-B) can be summarized as follows:

- **KMP-1: Feed Input Flow**
  The feed input to MBA-B is spent PWR fuel rods. Approximately 4,500 fuel rods are received in 3 months. Because of the difficulty of independently measuring the plutonium content of spent PWR fuel rods, the facility uses the measured weights and some declared values from previous MBA. Then this value would be verified by using the batch representative content values from the destructive chemical analysis on the samples of homogenized mixed powder at KMP-A. In reality, for materials accounting purpose KMP-1 and KMP-A are the same. Therefore, a variety of advanced monitoring systems should be adopted to verify the continuity of knowledge between fuel rods and mixed-
powder stage. These systems must have the functions that can track the fissile material, those that can follow the actions of people and equipment, and those that can determine if the process is being used appropriately.

- **KMP-2: Product Output Flow**

  The flow is the product output stream from the facility. Approximately seven U-metals are produced each day and a total of 371 newly fabricated U-metal rods are shipped out during 3-month period. The facility uses the destructive plutonium-content assay of the reduced U-metal ingot (KMP-H) to calculate the plutonium contents of the output metals. Each U-metal item contains approximately 0.36 kg of plutonium in total.

- **KMP-3: Waste Output Flow**

  This output stream includes measured waste forms, such as nuclear components of PWR fuel pin claddings contaminated with fuel residues, miscellaneous discardable materials, filters, salt, and other disposable trash. It is assumed that these wastes are measured with some NDA techniques to estimate the fissile content of discards and to declare them as measured discards. Approximately 60 kg of waste material containing 0.7 kg of plutonium are discarded as waste during a 3-month period.

- **KMP-A: Storage of mixed oxides after the rod-cutting and decladding stages.** Approximately 100 kg of materials, mostly containing uranium as UO₂, is located at inventory closing. These are in five containers, each containing approximately 20 kg. The plutonium fraction is assumed 0.01185. The facility performs destructive chemical analysis to determine the plutonium content. The analytical results are based on the reasonable sampling plan that provides the best attainable unbiased estimate of the true U or Pu concentration of the bulk material with minimum error due to sampling.

- **KMP-B: Discardable cutting waste of mixed oxide after the rod cutting.** Approximately 0.007 kg/rod of materials mostly containing uranium as UO₂ is produced by rod cutting. At the time of material balance closing, it is assumed that this location has approximately 7.5 kg of cutting waste in one container. To determine the plutonium content, the facility uses an
NDA technique based on the Cm ratio from the chemical analysis at KMP-A for each batch.

- **KMP-C:** Non-nuclear cladding materials of PWR fuel rods. The decladding ratio is assumed 99.9%. At inventory time, it is assumed that this location has approximately 250 kg of hull materials that contains 0.006 kg of plutonium. Just as KMP-B, these discards are measured with the NDA technique to declare them as measured discards.

- **KMP-D:** Disposable waste and associated powder residues mostly containing U₃O₈ from voloxidation process. About 7.5 kg of this material is accumulated during the MB period and remains at this location at inventory time. As there is no chemical change in actinides composition during the voloxidation process, the facility performs NDA measurements to determine the plutonium content.

- **KMP-E:** Extracted mixed oxides after the controlled Voloxidation stage. Approximately 100 kg of materials, mostly containing uranium as U₃O₈, are located at this location at MB closing. These are in five containers, each containing approximately 20 kg. The plutonium fraction is assumed to be 0.01183. There are no changes in actinides compositions after voloxidation, and the facility uses the NDA technique based on the chemical analysis at KMP-A to determine the plutonium content.

- **KMP-F:** Salt waste that is used for actinide reduction. It is assumed that the reduction ratio is 99.8% and the LiCl-Li₂O reductant is reused continuously for five processing batches. The salt ratio to reduce SNM is assumed to be three, and approximately 300 kg of salt is needed for each reduction process. At inventory time, it is assumed that this location has approximately 300 kg of salt waste in 15 containers, each containing 20 kg. The plutonium fraction is assumed to be 0.004. The facility uses the NDA technique to determine the plutonium content of the LiCl-Li₂O waste salt. The curium ratio to calculate the plutonium content is derived from the chemical analysis at KMP-A and KMP-H batch by batch. The (α,n) neutron yield will be high at this measurement point because of the high lithium content of the salt. Therefore, neutron NDA measurements (of curium) may demand counters with a very short neutron die-away time.

- **KMP-G:** Magnesia filter waste that is used for Li reduction and treated
as waste after the process of smelting the metalized powder. It is assumed that approximately 0.1% of the metalized powders are remaining in the filter waste. At inventory time, this location has approximately 150 kg of waste materials, containing 0.089 kg of plutonium. The facility uses the NDA techniques based on the chemical analysis at KMP-H to determine the plutonium content of magnesia filter wastes.
KMP-H: Uranium ingot after the controlled electrolytic reduction stage. Approximately 100 kg of uranium ingots are kept as five containers of 20 kg each at inventory time. The plutonium fraction is assumed to be 0.01187. The facility performs the destructive chemical analysis to estimate the plutonium content of the metal ingots. The analytical results are based on the reasonable sampling plan that provides the best...
attainable unbiased estimate of the true U or Pu concentration of the bulk material with minimum error from sampling and inhomogeneity.

- **KMP-I**: Dirty scrap of U-metal after metal casting process. The generation ratio of these scrap wastes is assumed to be 0.2%. At inventory time, it is assumed that this location has approximately 30 kg of waste materials, containing 0.178 kg of plutonium. The facility uses the NDA technique to estimate the plutonium content of the waste based on the chemical analysis at KMP-H.

- **KMP-J**: Fabricated U-metal rods. At inventory time, a total of 100 kg of U-metals are located as a form in interim storage. The plutonium fraction is assumed to be 0.01187. The facility uses the NDA technique to estimate the plutonium content of the metals. It is assumed that the IAEA may sample and verify the metals using a conventional neutron coincidence counter.

**C. Material Accounting and Verification**

Materials accounting is necessary to provide positive confirmation that all material has been properly handled. The need for verification stems from the basic principle that diversion of material could, in principle, be concealed by falsification of accounting data. To detect this possibility, the IAEA makes its own measurements of declared items and compares its values to those reported by the facility. If such comparisons fall within reasonable bounds (determined by error propagation), the inventory is deemed to be verified.

A “complete” verification, in which the IAEA would verify 100% of declared items, is impractical. The Agency does not have the resources to do this at all facilities; moreover, the cost to a facility in terms of production downtime for a 100% inspection would be substantial. Consequently, verification is done on a sampling basis, using NDA measurement techniques to get timely results.

The effectiveness of verification is limited by several generic factors, including

1. the number of inspection per year;
2. the amount of inspectorate effort expended per inspection;
3. the nature of the material to be inspected: amounts of SNM, numbers of items, and so on;
4. the sizes of measurement/sampling uncertainties on the parts of the facility and inspectorate;

5. the extent to which measurement uncertainties are well quantified, allowing accurate determination of control limits; and

6. the extent to which activities other than quantitative measurements can be relied upon, such as the use of seals and cameras.

Because inspection resources are finite, compromises must be made with respect to these generic factors.

We discuss below a hypothetical operating scenario. In a systems study fashion, related inventories are characterized. We make assumptions regarding measurement procedures on the part of the facility and inspectorate and address the implication of those assumptions for materials accounting and verification.

The results that follow from this characterization are limited by the assumptions. Many assumptions amount to educated speculation about a facility that does not exist. Other assumptions are convenient, but optimistic. For example, we assume that

1. there is no material holdup in the facility at the time of inspection (i.e., shutdown has been followed by a “complete” cleanout.)

2. at the time of inspection, all material has been run out of hard-to-measure forms; note that measurement uncertainties are related to the form of material and better measurement obviously leads to better accounting and verification.

3. NDA measurement technologies, which in some cases do not exist, will be used by the inspectorate to attain stated measurement uncertainties.

Many of the underlying assumptions make results look somewhat better than they should. However, we do not have enough detailed information to treat these issues more realistically.

1. Facility Materials Accounting

Because nuclear materials processed in ACP facility are contained in many types and forms of matrix, material accountability requires that the material contents of all flows entering and exiting the MBA and the quantities of nuclear material in the ending inventory be known. The MUF (Material Unaccounted For) is defined as the difference between the measured inventory and what is expected to be in the inventory based on the previous inventory and measured flows into and out of the process. The facility MUF for a given material balance period is a measure of the
performance of the facility with respect to its control of the nuclear materials involved. The MUF, as verified by inspection or, alternately, as adjusted on the basis of inspection results is the key index of performance used by the Agency in its quantitative assessment of facility performance. The MUF is calculated via the following equation:

\[ MUF = BI - EI + TI - TO \]

where, BI and EI are the beginning and ending inventories and TI and TO are the transfers of nuclear material into and out of the material balance area, respectively. Because measurement errors will occur, the actual amount of material measured will differ somewhat from the true value, creating a non-zero MUF.

The probability of detecting the loss of a given quantity of material depends upon the uncertainty associated with the determination of the MUF. Achieving the IAEA’s goal for Pu, to detect a loss of one SQ of Pu with 95% detection probability and a 5% false alarm probability, requires that \( \sigma_{MUF} \) must satisfy:

\[ \sigma_{MUF} \leq \frac{8}{3.3} = 2.424 \text{ kg} \quad \text{one-sided testing} \]
\[ \sigma_{MUF} \leq \frac{8}{3.65} = 2.192 \text{ kg} \quad \text{two-sided testing} \]

The one-sided testing means that the facility tests for loss and not for gain of Pu, so the statistical testing is one-sided and the alarm limit is at \( 1.65 \sigma_{MUF} \). Two-sided testing places the upper alarm limit for the inventory difference at \( 2 \sigma_{MUF} \). The control limit of \( 1.65 \sigma_{MUF} \) is such that an error means that the measured MUF has a 95% probability of being less than \( 1.65 \sigma_{MUF} \), assuming that the true MUF is zero, and assuming that all materials have been measured and accounted for and all sources of error are used in determining the limit of error.

The Limit of Error in MUF (LEMUF) value was determined based on a hypothetical operating scenario to investigate if the ACP facility would meet the detection goal of IAEA. Material accountancy requires quantitative knowledge of material 1) present in the material balance area at the beginning and ending of the accountancy period, and 2) transferred into and out of the area during the period.
Concrete results require the characterization of process operations and related material flows. It is also necessary to characterize the accounting system, facility measurement procedures, and related uncertainties. Because of insufficient detailed information on ACP facility to treat these issues at this time, assumptions regarding measurement procedures on the part of the facility and inspectorate were introduced.

Inventory for the bulk-handling area (excluding long-term storage) was assumed as shown in Fig. 7. The characteristics of 23 strata identified in ACP facility are summarized in Table 7. There are one bulk measurement method, three material type determinations, and four analytical methods in ACP. It is important from the standpoint of facility accounting that all items in inventory be associated with measured values. Such measured values should be obtained in a way compatible

<table>
<thead>
<tr>
<th>Stratum</th>
<th>KMP</th>
<th>Material Form</th>
<th>Total Element (kg)</th>
<th>Accounting Method</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>1</td>
<td>Spent Fuel Feed (most UO₂)</td>
<td>7500.00</td>
<td>DA + Weight</td>
</tr>
<tr>
<td>2</td>
<td>2</td>
<td>U-Metal Product (TRU+MA)</td>
<td>7440.00</td>
<td>NDA</td>
</tr>
<tr>
<td>3</td>
<td>3</td>
<td>Waste Output</td>
<td>60.00</td>
<td>NDA</td>
</tr>
<tr>
<td>4,14</td>
<td>A</td>
<td>BI and EI of Mixed Oxides Storage (for batch closeout, most UO₂)</td>
<td>100.00</td>
<td>DA + Weight</td>
</tr>
<tr>
<td>5,15</td>
<td>B</td>
<td>BI and EI of Cutting Waste (accumulated for 1 MB period)</td>
<td>7.50</td>
<td>NDA</td>
</tr>
<tr>
<td>6,16</td>
<td>C</td>
<td>BI and EI of Cladding Hull Materials (accumulated for 5 batches)</td>
<td>0.50</td>
<td>NDA</td>
</tr>
<tr>
<td>7,17</td>
<td>D</td>
<td>BI and EI of Disposable Waste &amp; Dirty Power Residues (accumulated for 1 MB period, most U₃O₈)</td>
<td>7.50</td>
<td>NDA</td>
</tr>
<tr>
<td>8,18</td>
<td>E</td>
<td>BI and EI of Mixed Oxides (for batch closeout, most U₃O₈)</td>
<td>100.00</td>
<td>NDA</td>
</tr>
<tr>
<td>9,19</td>
<td>F</td>
<td>BI and EI of Salt Waste (accumulated for 5 batches)</td>
<td>1.00</td>
<td>NDA</td>
</tr>
<tr>
<td>10,20</td>
<td>G</td>
<td>BI and EI of Magnesia Filter Waste (accumulated for 1 MB period)</td>
<td>7.50</td>
<td>NDA</td>
</tr>
<tr>
<td>11,21</td>
<td>H</td>
<td>BI and EI of Uranium Ingot (for batch closeout)</td>
<td>100.00</td>
<td>DA + Weight</td>
</tr>
<tr>
<td>12,22</td>
<td>I</td>
<td>BI and EI of Dirty Metal Scrap (accumulated for 1 MB period)</td>
<td>15.00</td>
<td>NDA</td>
</tr>
<tr>
<td>13,23</td>
<td>J</td>
<td>BI and EI of Uranium Metal Rods (for batch closeout)</td>
<td>100.00</td>
<td>NDA</td>
</tr>
</tbody>
</table>
with efficient operation. The destructive assay (DA) measurements for plutonium concentration are made on a batch basis. It is unnecessary, time consuming, and costly to obtain a sample from each individual container of powder. Instead, samples are drawn from containers deemed representative of other containers. It is assumed, for example, that the typical size of item container is 20 kg and the contents of the five containers are all originated from the same homogenized batch of oxide. It is also assumed that three concentration measurements per 10 containers are adequate for oxide powder and for metal ingots. Residues and waste strata after the voloxidation or reduction process have one concentration measurement for two containers.

The conceptual accounting for metal rods uses concentration values calculated from metal ingots. Thus, plutonium is not measured directly, again leaving the accounting system susceptible to an insider threat (i.e., an action on the part of one or more employees despite the best effort of the facility). Unfortunately, the alternative

<table>
<thead>
<tr>
<th>Sample Matrix &amp; Measurement Method</th>
<th>Uncertainty Component (% Rel. Std. Dev.)</th>
<th>Reference &amp; Notes</th>
</tr>
</thead>
<tbody>
<tr>
<td>DA : Spent Fuel Powder</td>
<td>0.2 0.2 10</td>
<td>- U &amp; Pu compounds measurement using IDMS at Hot Cell Conditions</td>
</tr>
<tr>
<td></td>
<td></td>
<td>- Sampling Uncertainty for dirty U scrap</td>
</tr>
<tr>
<td>NDA : Spent Fuel Powder</td>
<td>4 2</td>
<td>- Pu mass measurement using HLNC for LWR MOX</td>
</tr>
<tr>
<td></td>
<td></td>
<td>- Pu mass measurement using HLNC for MOX Scrap</td>
</tr>
<tr>
<td>NDA : Structural Materials &amp; Waste</td>
<td>10 5</td>
<td>- Error increased considering for difficulties in measuring salt waste</td>
</tr>
<tr>
<td></td>
<td></td>
<td>- U &amp; Pu compounds measurement using IDMS at Hot Cell Conditions</td>
</tr>
<tr>
<td></td>
<td></td>
<td>- Sampling Uncertainty for dirty U scrap</td>
</tr>
<tr>
<td>DA : U-Metal</td>
<td>0.2 0.2 10</td>
<td>- U &amp; Pu compounds measurement using IDMS at Hot Cell Conditions</td>
</tr>
<tr>
<td></td>
<td></td>
<td>- Sampling Uncertainty for dirty U scrap</td>
</tr>
<tr>
<td>NDA : Dirty Scrap</td>
<td>10 5</td>
<td>- HLNC for MOX Scrap</td>
</tr>
<tr>
<td>NDA : U-Metal</td>
<td>4 2</td>
<td>- Pu mass measurement using HLNC for LWR MOX</td>
</tr>
<tr>
<td>Weights</td>
<td>0.05 0.05</td>
<td>- Electronic Balance</td>
</tr>
</tbody>
</table>
is to use direct NDA-which is less accurate and increases the standard deviation. In this case, reliance on physical security is crucial.

The facility’s material control and accountability methods propagate all measurement and sampling uncertainties to give a standard error. As shown in Table 8, the measurement methods used for the material accounting are assumed to have various uncertainties based on the ITV 2000 [viii]. The measurement precisions and accuracies reflected in the table by the random and systematic uncertainties, respectively, are values achieved in the analysis of materials of nuclear grade or similar chemical impurity. They include the contributions of all uncertainties occurring after sampling. The effects of sampling, impurities and foreign components will vary with the type of material, to the extent that sampling uncertainties can become the dominant factor in the total measurement error [ix].

With the classifications in mind, the following assumptions were also made for the MUF calculation:

1. The number of samples drawn and the number of analyses per sample are both constants for the given batch.
2. The number of items per batch is constant within a given stratum.
3. No more than one scale or analytical method is used in a given stratum.
4. A given element concentration factor cannot apply to more than one stratum.

If the measurement methods in question are of the same design (e.g., same type scale), it is assumed that the use of several scales is equivalent to the use of one scale with several shifts in the systematic error (i.e., without a short-term systematic error).

Using these assumptions and uncertainty values, the result for the \( \sigma_{MUF} \) is 1.881 kg of elemental plutonium, assuming no data falsification. The corresponding limit of error value for MUF is 3.761 kg of plutonium. This result suggests that it would be possible to meet typical IAEA detection goals for campaigns having 3 months or fewer. This calculation is a preliminary estimate that is expected to be modified as more information becomes available about measurement performance.

2. Verification

One aspect of verification involves IAEA review of the MUF and LEMUF calculations, which are part of the facility accounting. The IAEA should be able to reproduce independently these calculations using declared data. As for inspection, declared items are stratified by the IAEA based on reporting requirements and type
of material, and the stratification need not be tied to accounting issues (e.g., how accurately material is measured and how homogeneity of strata in terms of material amounts).

It has been pointed out that the primary role of inspection from an accounting viewpoint is to install confidence in the reported MUF and its variance. In performing this function, the so-called D statistics, or the difference statistic, is of prime importance. The quantity D is an estimate of this bias in the facility MUF. In actuality, it estimates a relative bias between the facility and the inspection agency, which is interpreted as a bias in the facility MUF when the assumption is made that the agency inspection measurements are unbiased.

Once stratification has been completed (this is accomplished by the IAEA through a computer code), sample size are then calculated. These sample sizes determine the numbers of items to be inspected randomly by the IAEA for measurement. Just as the IAEA should be able to reproduce the facility’s MUF and LEMUF independently, the operator should be able to independently reproduce the IAEA’s stratification and sample size calculation. In other words, the facility should be in a position to know the inspection plan at the time it knows its inventory.

Timeliness of verification—e.g., the number of inspections per year—is also a relevant concern. An important consideration regarding the necessary timeliness of inspections is the time required to convert SNM to a weapon. If the current IAEA philosophy regarding timeliness were applied to the conceptual facility, the nature of ACP material would be similar to spent LWR fuels and lead to inspection at least four times annually. The related goal quantity for inspections is 8 kg of plutonium, which is equivalent to about 700 kg of ACP material.

Measurement options for the inspectorate are varied. In some cases, samples of material can be obtained and sent to a laboratory for analysis, although this is uncommon because of the expense and the desire to obtain prompt results. In other cases, so-called attributes measurements, which are obtained quickly and cheaply at some cost in measurement accuracy, are used to verify large numbers of items for so-called gross defects.

At the ACP facility, attributes measurements may be impractical if the majority of time related to NDA measurement involves remote handling of items and transport to/from the measurement stations. In the following, we assume that only one type of NDA measurement per item is used, with no destructive samples and no attributes measurements; this assumption will be revisited if conditions warrant.
The sample sizes of each stratum in Table 9 were determined by following formula.

\[ n = N \left( 1 - \beta^{\frac{X'}{SQ}} \right) \]

In this formula, the symbol \( N \) denotes the number of declared items in a stratum, \( n \) denotes the stratum sample size, \( X' \) denotes the average plutonium per item in the stratum, and \( \beta \) denotes the non-detection probability (for 1 SQ = 8 kg of plutonium per stratum at the level \( \beta = 0.05 \)). This relates the minimum sample size \( n \) required to achieve specified sensitivity against gross defects for the stratum in question [x].

Developing concrete results requires characterization of the inspectorate’s measurement procedures and related uncertainties. Generally speaking, the inspectorate must rely heavily on NDA procedures, because DA methods do not provide timely feedback (especially when samples must be shipped off site). In this study, therefore, it is assumed that inspectors use the most available NDA procedures

<table>
<thead>
<tr>
<th>Stratum</th>
<th>KMP</th>
<th>Total Pu (kg)</th>
<th>No. of Items per Stratum</th>
<th>No. of Samples per Stratum</th>
<th>No. of Analysis per Stratum</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>1</td>
<td>88.875</td>
<td>375</td>
<td>225</td>
<td>675</td>
</tr>
<tr>
<td>2</td>
<td>2</td>
<td>88.164</td>
<td>372</td>
<td>223</td>
<td>670</td>
</tr>
<tr>
<td>3</td>
<td>3</td>
<td>0.711</td>
<td>3</td>
<td>2</td>
<td>5</td>
</tr>
<tr>
<td>4,14</td>
<td>A</td>
<td>1.185</td>
<td>5</td>
<td>3</td>
<td>9</td>
</tr>
<tr>
<td>5,15</td>
<td>B</td>
<td>0.089</td>
<td>1</td>
<td>1</td>
<td>3</td>
</tr>
<tr>
<td>6,16</td>
<td>C</td>
<td>0.006</td>
<td>25</td>
<td>15</td>
<td>45</td>
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<tr>
<td>7,17</td>
<td>D</td>
<td>0.089</td>
<td>1</td>
<td>1</td>
<td>3</td>
</tr>
<tr>
<td>8,18</td>
<td>E</td>
<td>1.185</td>
<td>5</td>
<td>3</td>
<td>9</td>
</tr>
<tr>
<td>9,19</td>
<td>F</td>
<td>0.012</td>
<td>15</td>
<td>9</td>
<td>27</td>
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<tr>
<td>10,20</td>
<td>G</td>
<td>0.089</td>
<td>15</td>
<td>9</td>
<td>27</td>
</tr>
<tr>
<td>11,21</td>
<td>H</td>
<td>1.185</td>
<td>5</td>
<td>3</td>
<td>9</td>
</tr>
<tr>
<td>12,22</td>
<td>I</td>
<td>0.178</td>
<td>3</td>
<td>2</td>
<td>5</td>
</tr>
<tr>
<td>13,23</td>
<td>J</td>
<td>1.185</td>
<td>5</td>
<td>3</td>
<td>9</td>
</tr>
</tbody>
</table>
of facility for each stratum and the procedures have same uncertainties as facility accounting.

As was true for the calculation of the variance of MUF by general formulas, the assumptions will rarely if ever be completely valid in given application. However, experience has shown that this is not a great difficulty, since in many cases, even moderate departures from the assumptions have very little effect \[ \xi \]. The assumptions about the facility data were set forth in previous section. The additional assumptions relative to the inspection are as follows:

- The inspector and the facility use the same material sampling procedures, and hence, systematic errors in sampling will cancel.

- When there are batches within a stratum, the inspector may first sample batches at random and then measure the same number of items in each batch sampled.

In practice, the value of \( D \) will not equal zero because of measurement errors on the parts of the facility (for declared values) and the inspectorate (for verification values). In most cases, \( \sigma_D \) greatly exceed \( \sigma_{MUF} \) because the inspectorate’s accounting is based on poorer quality measurements (e.g., NDA vs. DA) of fewer items. It is necessary to compare \( D \) to a limit, based on propagation of the uncertainties involved, to evaluate the possibility of data falsification. For the inspection plan developed for the conceptual ACP facility, \( \sigma_D \) was estimated as 3.175 kg of plutonium. Thus \( \sigma_D \) is roughly 3.57% of the total plutonium handled during MB period. The largest single contributor to \( \sigma_D \) involves PWR powder measurement. From the D statistics results, it could be concluded that the sensitivity of the verification for conceptual ACP facility is very good because the inspection plan affords good protection against gross falsification and \( \sigma_D \) small relative to 1 SQ (8 kg of plutonium).

The fact that the probability of detection is larger for (MUF-D) than for the D and MUF tests applied separately has been shown to be true in general. This fact is consistent by analogy with the earlier reported finding discussed previous section in which the statistical advantages of making a single test of diversion without making sub-divisions by time or by space was pointed out. Thus, the (MUF-D) statistic, being a global statistic would be expected to have the same advantage as the MUF global statistic. Considering the ACP facility’s material accounting and verification discussed in previous, the \( \sigma^2_{MUF-D} \) and \( \sigma_{MUF-D} \) were estimated as 6.55 kg²Pu and 2.56 kgPu. The correlation coefficient between MUF and D was 0.74.
VI. SAFEGUARDS R&D REQUIREMENTS

Based on the key elements considered for the design of a safeguards system for the conceptual ACP facility, it was expected that the most of the technologies necessary to safeguard spent nuclear fuels at the reactor facility and during transportation are now available. However, because of the unique nature of electrometallurgical concept and the radiation characteristics of material associated with the ACP, not all technologies and system necessary to safeguard the fissile materials are presently available. Known principles of NDA techniques and destructive chemical analysis could be adapted to design equipment and procedures to meet the requirements of safeguarding nuclear materials at the ACP facility. During and following the selection of an ACP option for engineering demonstration, parallel efforts must be directed at developing systems for material accounting, measurements, containment and surveillance, and verification of the flow and inventories of materials at the ACP facility. A variety of material verification measurement approaches should be examined before deciding on an optimal plan for developing ACP-specific NDA instruments. Research and development requirements for such instruments include the following:

- A fully described flow sheet for the processes and material flows across the ACP facility should be developed.
- A computer model for the process operations and material flows to evaluate safeguards consequences might be needed before the actual operation of facility.
- A modern computerized nuclear materials accountancy system at the facility level capable of automated data recording at strategic points may be needed. Such data, using authenticated recording systems, can be shared with the IAEA to facilitate periodic inspections and detailed physical inventory verifications.
- A variety of advanced monitoring systems should be developed to complement material accounting systems in the ACP facility. These could include systems that can track the fissile material, those that can follow the actions of people and equipment, and those that can determine if the process is being used appropriately.
Some details of new systems and technologies that are desirable for establishing a modern safeguards regime for the ACP facility are the following:

- developing neutron multiplicity techniques to separate signals from $^{244}\text{Cm}$ and plutonium for spent PWR fuel;

- investigating the feasibility of using remote neutron monitoring to safeguard spent PWR fuel during the conditioning process to produce U-metal;

- investigating the use of LA-MS (laser ablation mass spectrometry) or LIBS (laser induced breakdown spectroscopy) for the monitoring of the Pu:U:Cm fraction in process material;

- developing authentication methods for unattended NDA of spent PWR fuel and fabricated U-metal;

- developing a variety of advanced C/S technologies that can track the fissile material and follow the actions of people and equipment based on the sophisticated integration of position sensitive radiation detectors and digital sensors.
VII. SUMMARY AND CONCLUSIONS

The ACP concept is an electrometallurgical treatment technique to convert PWR spent nuclear fuel into a single set of metal disposal forms, reducing the volume and simplifying the qualification process. In this study, a preliminary assessment of the safeguardability on a pilot-scale ACP facility was performed by the collaboration efforts of the LANL and KAERI.

An objective assessment of the safeguardability of the ACP requires simultaneous consideration of its inherent proliferation resistance and the ability to design an integrated system of technical measures that guard against diversions and achieve timely detection. We briefly examined the proliferation and diversion related issues in section III. To carry out a detailed analysis of diversion possibilities and to identify safeguards elements to detect such diversions, it is necessary to examine attributes of the ACP fuel cycle. At this research stage, however, most of these attributes are not clearly defined, and few of them are quantifiable to assess the safeguardability of the ACP. Therefore, this diversion scenario analysis should be considered only as a preliminary evaluation.

We examined the technical requirements for safeguarding and verifying the ACP fuel cycle in Section IV. We also analyzed the material accountability with known measurement methods and their uncertainties in Section V. The measurement uncertainties used in this report are obtained in the open literature related to safeguard issues and experience with a variety of fuel fabrication facilities now in operation worldwide. The assumptions made in this study may be too optimistic for the ACP facility because of the radiation characteristics of the materials. Nevertheless, our conceptualization of facility features and material flows across the ACP facility leads us to conclude that a safeguards system can be designed to meet the IAEA’s detection goals and to provide an independent verification scheme. As we get information on measurements and verification approaches that is more reliable, these data and calculations can be modified.

We identified the requirements for developing necessary instruments to verify the materials accountancy system at the ACP facility in section VI. Isotopic analysis for ACP materials with respect to mass distribution, total dose rate, neutron production rate, and heat recommends that a curium-monitoring method could be available if the amount of Pu relative to Cm is verified continuously at all stage of the process.
Based on the assumptions we have made during this preliminary study, the conceptual ACP fuel conditioning facility credibly meets reasonable diversion resistances and safeguards goals. Therefore, it is reasonable to conclude that (1) using the options proposed for Phase-I study, ACP fuel conditioning facility can be designed, built, and operated to meet the goals of IAEA safeguards; and (2) additional technologies necessary to make this possible can be developed in a timely fashion so that an inventory of the ACP facility materials can be verified by both the state and international safeguards inspectors.

The results described in this report should be considered only as a preliminary evaluation, subject to possible modifications as more reliable information on technical parameters becomes available. Those features that best promote safeguards on the ACP are: ability to make good measurement; continuous knowledge of the amounts and locations of materials; building designs that contain nuclear materials in easily controlled and monitored areas; and process operations that minimize pathways for diversion.
REFERENCES

i. RadWaste Management Center, http://www.4energy.co.kr/know/know1.html


<table>
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<tr>
<th>제목/부제</th>
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<td>있음(0), 없음( )</td>
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<td>본 연구에서는 상용규모의 금속전환공정에서 핵물질안전조치의 계량관리측면에서 국제원자력기구의 탐지기준을 만족시킬 수 있는지의 여부를 분석하였다. 금속전환공정은 고온융융열 공정을 이용하여 산화물 형태의 사용후핵연료를 금속으로 환원시키므로써 엄부하 및 부피를 크게 줄일 수 있는 공정이다. 핵물질 안전조치성을 부식하기 위하여 먼저 상용규모 시설의 설계 개념 및 물질흐름을 분석하여, 시설내 물질저항구역 및 주요측정지점을 설정하였다. 물질저항구역에서 MUF를 계산하기 위하여 측정지점에서의 핵물질 측정 방법 및 측정오차들이 가정되었으며, 이를 위하여 주로 2000년에 발간된 IAEA의 Target Value를 참조하였다. MUF 평가 결과 현재 가정된 측정 오차를 기준으로 할 경우 금속전환시설은 핵물질안전조치성이 있는 것으로 분석되었다.</td>
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</tbody>
</table>
The Advanced Spent Fuel Conditioning Process (ACP) is an electro-metallurgical treatment technique to convert oxide-type spent nuclear fuel into a metallic form. This report documents a preliminary study on the safeguardability of ACP. The sub-processes and material flow of the pilot scale ACP facility were designed for this study. Then, their material balance areas (MBA) and key measurement point (KMP) were defined based on diversion scenario analysis. Finally, the limit of error in the MUF value was estimated using international target values for the uncertainty of measurement methods. Based on the results of preliminary study, we concluded that the safeguards goals of International Atomic Energy Agency (IAEA) could be met if the assumptions regarding measurement instruments can be achieved in a safeguards system for the ACP facility.