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FEASIBILITY STUDY ON THE VERIFICATION
OF FRESH FUEL ASSEMBLIES IN
SHIPPING CONTAINERS

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SUMMARY

The purpose of this study was to examine the feasibility of using various nondestructive measurement techniques to determine the presence of fuel assemblies inside of shipping containers (attribute measurements) and to examine the feasibility of measuring (variables measurement) the fissile content of the containers. Passive and active techniques based on both gamma and neutron assay were examined. The experience of other investigators was evaluated through both literature and conversations with selected investigators. In addition, some experiments were performed and calculations were performed to evaluate neutron techniques.

Passive counting of the 186 keV gamma from ^{235}U is recommended for use as an attributes measurement technique. Experiments and studies indicated that a bismuth germanate (BGO) scintillator is the preferred detector. A properly designed system based on this detector will provide a compact detector that can selectively verify fuel assemblies within a shipping container while the container is in a stack of similarly loaded containers. Missing fuel assemblies will be readily detected, but gamma counting of assemblies cannot detect changes in the fissile content of the inner rods in an assembly.

A variables technique for the assay of assemblies in the containers (active neutron interrogation) does not appear feasible due to the poor coupling of sources and detectors to the assemblies in the shipping containers which reduces the sensitivity of the neutron techniques. Calculations performed also indicated poor sensitivity to changes in total fissile content of the containers. If a variables technique is required, it is recommended that more extensive calculations be performed and removal of the outer shipping container be considered.

Marking (sealing) of the assemblies with a uniquely identifiable transponder was also considered. This would require the development of procedures that would assure proper application and removal of the seal. Although this technique appears promising, when the anticipated change to a metal outer container occurs, the technique will no longer be useful unless a radiolucent window is included in the container.

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ACRONYMS

BGO	bismuth germanate
BWR	boiling water reactor
CIND	AmLi Interrogation System
ENDF-B/V	Evaluated Nuclear Data File-B Version V
JNF	Japan Nuclear Fuel
HLNCC	high-level neutron coincidence counter
IAEA	International Atomic Energy Agency
MCA	multi-channel analyzer
MCNP	Monte Carlo Neutron Photon Code
PWR	pressurized water reactor
UNCL	uranium neutron coincidence collar

1.0 INTRODUCTION

In the operation of a fuel manufacturing plant, it is necessary to store fuel prior to shipment. Often, this fuel is stored in standardized shipping containers. The containers for boiling water reactor (BWR) fuel are designed to hold two fuel assemblies and constitute a package of considerable bulk. At present, it is impossible for the International Atomic Energy Agency (IAEA) to verify the fuel while it is stored in these containers. However, opening the containers to verify and measure the assemblies intrudes on facility operations. At Japan Nuclear Fuel (JNF), for example, the assemblies in the container have been through quality control checks and certified prior to storage. Accessing a selected container requires at least the following six steps:

1. removing the container (~1300 lb) from a stack of containers
2. opening the container for access to the assembly - The wooden overpack and the metal inner container must be opened.
3. moving the assembly to measurement location
4. completing measurements
5. recertifying the fuel and repackaging
6. replacing the container in storage.

It is desirable to have a method of quantitatively verifying the ^{235}U content of the shipping pack without opening the pack and, ideally, without having to unstack the containers.

This study, performed by Pacific Northwest Laboratory^(a) for the IAEA, examines several measurement techniques for this application and to recommend a technique for gross or partial defect measurements on the packaged assemblies. Alternative, non-measurement-based schemes are also examined. The examination includes the estimated accuracy of each method, together with cost, complexity, inspection effort required, and the level of intrusiveness to the operator.

1.1 THE PROBLEM

The IAEA is faced with the problem of following the flow of fissile materials and accounting for such materials throughout their use in the nuclear industry. In the area of the fuel cycle for light water reactors, this is primarily a problem of accounting for large quantities of low-enriched uranium as it progresses from uranium hexafluoride to powder, to pellets, to fuel assemblies, and to spent fuel. At one point during the cycle, the

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uranium will be in manufactured fuel assemblies stored at the fuel manufacturing plant, awaiting shipment to a light water reactor for power production. In many cases, these fuel assemblies are stored in standard shipping containers, which are stacked in a storage area. The containers actually consist of an outer wooden container with an inner metal container (see Figure 1.1). The containers are very large so that opening the containers to observe or make measurements on the contents is a labor-intensive operation. In addition, in Japan the containers are sealed by government officials, further complicating their being opened.

Thus, at this stage in the fuel cycle, the inspector will be faced with large stacks of containers and their declared contents. These containers will be stacked up to five high and require a fork lift in order to be moved. He or she must independently decide to accept or reject the operator's declaration. This can be done by randomly opening and/or measuring a sample of the containers, using prescribed statistical sampling plans. Alternatively, if the container contents had been verified and the containers sealed and they were to be later verified at opening, it would only be necessary to verify the seals. The operations of the inspector are further complicated by the limited time available for the inspection and the limited amount of equipment an inspector is able to carry into the plant. Alternatively, dedicated equipment can be purchased and left at the plant under seal between inspections.

It is important to examine what happens to the fuel assemblies before being placed in the storage containers and after removal. Based on information received from the Agency, the fuel assemblies are measured in an installed neutron collar after being manufactured and before being placed in the containers for storage. After they arrive at the power reactor, the containers will be opened and the fuel assemblies will be stored until introduced into the reactor. This is generally a dry storage, in which the assemblies are accessible for measurement and verification of the imprinted identification numbers.

1.2 THE FUEL ASSEMBLY

The fuel assembly, which is 447 cm long with a fuel region 365 cm long, is made up of an 8 x 8 array of fuel rods containing uranium pellets of various enrichments, as shown in Figure 1.2. In reload assemblies, some of the rods will contain burnable poisons such as Gd_2O_3 . The typical assembly contains ~208 kg of UO_2 , consisting of approximately 5.02 kg of ^{235}U . The typical fuel assembly weighs ~320 kg (704 lb) and is 4.47 m long.

Boiling water reactor fuel assemblies are asymmetrical and are not always oriented identically (i.e., same side up) when placed in containers. A typical BWR fuel assembly consists of an array of 63 (8 x 8) or 79 (9 x 9) pins. The assembly will contain pins with enrichments from 1.50% to 4.34% by weight. Although the enrichments will vary throughout the assembly, a typical enrichment is 3.1%. This corresponds to a linear density of 14.2 g/cm of ^{235}U and 448 g/cm ^{238}U . A cross-section of a typical BWR assembly is shown in Figure 1.3. For the purposes of this study, all assemblies are assumed to be uniformly loaded with pins containing 3.1% by weight ^{235}U .

S900905.2

FIGURE 1.1. RA-2 Shipping Container for BWR Fuel Assemblies
(Source: JNF n.d. p.14)

FIGURE 1.2. Typical BWR Fuel Assembly (Source: T. Osabe. 1987. Draft Feasibility Study for Application of Non-destructive Assay on Fuel Bundles in Shipping Container. T-179, JNF, Kurihama, Japan.)

S9009051.6

FIGURE 1.3. Cross-section of BWR Fuel Assembly Showing Variation in Fuel Enrichment (Source: Menlove and Pieper 1987)

1.3 THE SHIPPING CONTAINER

The shipping containers consist of reusable inner and outer containers. The inner container, which is designed to hold two fuel assemblies, is a double-walled steel container with a bolt-on lid. Overall, the inner container is 465 cm long, 46 cm wide, and 29 cm high. The inner and outer walls of the steel container are separated by approximately 5 cm and have welded inner reinforcement pieces. The lid of the container is bolted on and sealed by a foam gasket. The fuel assemblies are buffered in the container by pieces of foamed polyethylene of 2-cm thickness. In the containers observed, the buffer material was not continuous, but consisted of several pads glued to the walls. The overpack or buffer materials had been used repeatedly and showed some wear.

The outer container is a 1/2-in. plywood box with an outer frame of 2-in.-by-4-in. lumber. The inner metal container is buffered by a combination of foamed polyethylene and cardboard honeycomb pieces. Again, the buffering materials were not continuous in the containers observed. Each container is 526 cm long by 84 cm high by 81 cm wide, including the bracing attached to the bottom to provide a gap of approximately 10 cm between stacked shipping containers. Figure 1.1 shows a sketch of a RA-2 type of shipping container and Figure 1.4 shows a cross-section. Although consideration has been given by JNF to building a tube into the shipping container to accommodate detectors (or sources) in an improved counting geometry, the difficulties in licensing a container with such a structural change have eliminated further consideration of such a technique (see Appendix). Japan Nuclear Fuels is also considering constructing the outer container of metal; the latest correspondence indicates this change should occur in 1991.

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FIGURE 1.4. Simplified Cross Section of Shipping Container Showing the Location of the Fuel Assemblies

2.0 MEASUREMENT TECHNIQUE CONSIDERATIONS

To rank measurement techniques, the parameters to be considered include accuracy, intrusiveness, cost of operation (including time), and cost of implementation.

For this analysis, accuracy was defined as high ($< \pm 6\%$), moderate ($\sim \pm 25\%$), and low ($> \pm 50\%$). If the measurement technique did not provide some measure of total ^{235}U mass, an accuracy statement was not useful and was considered to be not applicable (na).

For the present measurements, the system can be placed in three levels of intrusiveness. First, minimal intrusive techniques would permit a measurement on the fuel assemblies without moving the shipping containers. A moderately intrusive technique would require isolation of a shipping container. Finally, a highly intrusive technique would require direct access to the assembly (i.e., the shipping containers must be opened). The highly intrusive techniques were considered unacceptable for this study. This immediately eliminates the conventional techniques for assay of fuel assemblies such as the neutron collar.

Cost of operation would be dictated by the cost of consumables and the time required to make a measurement. For the purposes of this study, measurement time estimates were based on verification of 40 assemblies in 20 containers stacked four high. Cost of implementation is dictated by the cost to fabricate or purchase hardware, transport equipment, calibrate equipment, and train personnel. Thus, adaptation of existing hardware and techniques would prove to be the most cost-effective.

3.0 TECHNICAL APPROACHES

The technical approaches considered included passive gamma and neutron measurements, active neutron measurements, fuel length verification, remotely read seals, and weighting.

3.1 PASSIVE GAMMA MEASUREMENTS

Passive gamma analysis is commonly used for the assay of ^{235}U in powder and pellet form. The 185.7-keV gamma ray from ^{235}U is a good indicator of ^{235}U and often used for enrichment determinations. There are other weaker gamma rays associated with ^{235}U and Pa gamma rays associated with ^{238}U . The most intense gamma ray associated with ^{238}U is the 1001-keV $^{234\text{m}}\text{Pa}$ gamma ray. Since for safeguards applications it is the mass of ^{235}U that is of interest, the main interest is in the 185.7-keV gamma ray. Unfortunately, this gamma ray is of low penetrating power and heavily attenuated by the uranium matrix and any containment. In the case of fuel assemblies, few gamma rays will penetrate from beyond the surface of the nearest rods. Relating the intensity of these gamma rays to the ^{235}U in the fuel assemblies in storage containers is extremely difficult since these gamma rays will be further attenuated in the packing materials and the interpretation will be complicated by the fact that the two assemblies in the shipping container are located side by side in the center of a large container, thus making discrimination between the assemblies difficult.

3.1.1 ^{235}U Measurements

Studies on the feasibility of using the 185.7-keV gamma ray as a signature for the fuel assemblies have been performed by the JNF. Information from one such study is included in the Appendix. The study shows that counting over the shipping containers does provide a signature that will indicate the presence or absence of a fuel assembly independent of the location of the assembly within a stack of shipping containers. Thus, at a minimum, the technique is feasible as an attribute test. The system as designed by JNF shows poor resolution of fuel assemblies and low count rates. JNF has proposed using the system in a scanning mode to determine the number of fuel assemblies. The tests will not be sensitive to inner rods, the data is imprecise, and the interference from neighboring assemblies complicates the interpretation of data. However, the data is precise enough to detect large changes in enrichment of the outer rods in an assembly.

Analysis of the data from JNF has shown that there are several improvements that can be made to the system to improve its sensitivity and thus shorten the overall analysis time. First, improving the collimation will provide better resolution of fuel assemblies and reduce the number of counting positions required. Improving the collimation will also decrease the counting efficiency since the efficiency varies approximately as the fourth power of the radius of the collimator and inversely as the square of the length of the collimator. Second, increasing the detector area will increase the count rate and should more than compensate for any loss in count rate due to improved collimation. Third, the use of a better shielding material such as tungsten

will reduce the background in the detector and improve the analysis. For example, 3 cm of lead will still transmit 9% of 150-keV gamma rays, but only 1% of 150-keV gamma rays will be transmitted through the same thickness of tungsten. Finally, by counting at selected positions over the fuel assemblies, the number of counts needed will be greatly reduced. Analysis of the data provided by JNF shows that, directly over assemblies, the count rate for assemblies was 2190 ± 250 counts in 200 seconds with a range from 1750 to 2600 counts. Over an empty position, the count rate was 650 ± 230 counts per 200 seconds with a range from 400 to 1000 counts. Thus, taking a count at a position over each fuel assembly should be adequate to determine if the assembly is present or absent. The range of count rates is large and it is recommended that counts be taken over each of the assemblies and at a point half-way between the assemblies. With the present collimator, the penumbra includes both assemblies in a single shipping container; thus, the maximum count rates are observed at a position half-way between two assemblies. When a single assembly is present, the ratio of a count rate over the assembly to the mid-position count rate is always greater than 1; when both assemblies are present, this ratio will be less than 1. This adds extra credibility to the measurements.

Since the distance between shipping containers is only 10 cm, the detector configuration must be limited to a maximum height of approximately 8 cm to permit free movement in the gap between the two containers. Because of the limited space available, it is essential that the highest Z detector be used, along with the most effective shielding material. The higher Z of the detector material permits a small volume detector and, thus, more shielding material around the detector. Bismuth germanate (BGO) is the most effective detector material currently available. Four mm of this material will stop 99% of 150-keV gamma rays, whereas it takes 2 cm of NaI (Tl) to stop 99% of the same gamma rays. The BGO detector has a lower light output than NaI (Tl) (Anderson 1989, Evans 1980), but has a superior photofraction, thus leading to less interference from higher energy photons. In addition, an improvement in geometry can be realized by orienting a long narrow detector parallel to the axis of the fuel assembly.

Section 3.1.3 describes test results for a BGO and a NaI (Tl) detector.

Based on data from the JNF tests and a simplified procedure, it would take an estimated 320 min to make measurements on 40 assemblies in an array of shipping containers four containers high and five containers wide.

3.1.2 ²³⁸U Measurements

Although assay by the 765- and 1001-keV gamma rays from the ^{234m}Pa daughter of ²³⁸U is possible, it takes several months before the daughter is in equilibrium. Thus, any uranium which has recently been chemically processed will not be in equilibrium. In addition, substitution of depleted uranium for the enriched uranium would not be detected. Thus, this measurement is not considered reasonable as a safeguards approach.

3.1.3 Bismuth Germanate Tests

In order to better address the feasibility of using a BGO scintillator, a sample 1 cm x 0.5 cm x 5 cm was obtained and tested. This was advisable because research reports often represent optimum results, which are difficult to reproduce in a production mode. Figure 3.1 shows overlapping ^{67}Co (122-keV) spectra from a small-volume NaI (Tl) crystal and the BGO crystal. The system was operated to provide identical electronic amplification for the signal from the two crystals. The higher light output (gain) of NaI (Tl) is obvious. The 122-keV photons from the ^{67}Co source are partially obscured by the dark noise of the photomultiplier. Optimization of the photomultiplier (Kume et al. 1986) and the crystal should improve the separation of background and signal. Other techniques such as pulse-shape discrimination, coincidence (Swinth 1974), cooling, or magnetic limitation can be used to reduce the photomultiplier noise but are not recommended for the present application.

3.2 PASSIVE NEUTRON MEASUREMENTS

Approximately 95% of the uranium in each assembly is ^{238}U , which has a spontaneous fission yield of 5.64×10^{-8} fissions per second per gram. With a uranium sample mass of nearly 180 kg, the total emission rate from the two assemblies in a single container would be over 4000 neutrons per second. The emission rate from a section of the fuel assemblies that was approximately 0.5 m long would be about 550 neutrons per second. These fission neutrons will have good penetrability. The main absorption mechanism for the neutrons will be fast neutron fission, resulting in additional neutrons. Counting neutrons from the assemblies while they are in their stacked storage containers would make it difficult to separate or isolate neutrons from the various assemblies. That is, the detectors would be approximately as close to neighboring assemblies as to the assemblies in the container of interest. Therefore, for any successful passive neutron counting it is necessary to unstack the containers and partially isolate them from the main stack.

Once the containers are separated from the main stack, two different techniques are used for passive neutron measurements: totals counting and passive coincidence counting.

3.2.1 Totals Counting

For totals counting of the neutrons from the two assemblies in the container, detectors would be placed around the section of the box containing uranium and all neutrons would be counted. The neutron response may be ambiguous due to the presence of a variable neutron component from alpha reactions. However, this problem should be minimal unless reprocessed uranium is used in the manufacture of the fuel, in which case a variable amount of ^{234}U could cause large variations in the number of neutrons from (α, n) reactions. An additional problem with totals counting will be the remaining background of neutrons arising from the additional storage containers in the room. It would be very difficult to effectively shield or totally isolate this fast neutron background in a practical safeguards approach.

FIGURE 3.1. ^{57}Co Spectra, 122 keV, for a BGO and a Small NaI Scintillator
(Note that the system gain was identical for both measurements.)

3.2.2 Passive Coincidence Counting

Placing detectors around the assembly and performing passive neutron coincidence counting can reduce the background signal, the signal originating from the more distant assemblies, and the signal from (α ,n) reactions. The passive coincidence rate under ideal conditions is only 3 counts/sec for a single BWR assembly in a neutron collar (Menlove and Pieper 1987). Expanding the collar around the shipping container will seriously degrade the counting geometry so that coincidence counts of significantly less than 1 count/sec could be expected. In former tests with an expanded collar and a pressurized water reactor (PWR) test assembly, the count rate was halved in a geometry significantly more favorable than the present one (Menlove and Sapir 1988). A larger neutron counter will be required if the counting efficiency drops in the coincidence mode. Since the passive neutron signal comes from ^{238}U , it is not useful to differentiate between depleted and low-enriched uranium. The passive coincidence counting would provide a measure of the total ^{238}U mass, which would have to be combined with active counting or gamma ray enrichment measurements to determine the ^{235}U mass. If the ^{235}U mass is determined from gamma ray enrichment measurement, the further assumption must be made that the sample is homogeneous, i.e., none of the inner rods of the assembly have been removed.

3.3 ACTIVE NEUTRON MEASUREMENTS

By irradiating samples containing fissile material, ^{235}U or ^{239}Pu , fissions can be induced in the materials of safeguards interest. Then, by determining the number of fissions induced for a fixed source strength, the quantity of fissionable material can be determined. This is the basis for several safeguards techniques. Several parameters distinguish the techniques: the counting of delayed or prompt fission neutrons or gammas, the use of energy thresholds to discriminate against the interrogation source, or the use of coincidence counting techniques to provide a unique signature for neutrons from fission events. The techniques also require some tailoring of the interrogation spectrum to provide adequate penetration of the samples. Compared to gamma measurements, however, the neutron interrogation methods can provide a more reliable bulk measurement of fissile materials in the sample due to the improved penetrating power of the neutrons. The following subsections discuss several types of neutron counting and interrogation techniques: neutron energy threshold counting, coincidence counting, delayed neutron interrogation, and the differential die-away method.

3.3.1 Neutron Energy Threshold Counting

By using low-energy or thermal neutrons to interrogate the sample and then counting the high-energy fission neutrons in a neutron-energy-sensitive system, the interrogation source neutrons can be distinguished from the neutrons from the sample. In addition, in the case of uranium samples containing ^{238}U , it is important that the neutrons remain low in energy to avoid exceeding the fission neutron threshold for ^{238}U , which is approximately 1.4 MeV. This will prevent an interference signal from fissions in the ^{238}U .

Totals Neutron Collar

One of the first devices developed for the IAEA for the verification of light water reactor fuel assemblies used low-energy neutron source interrogation together with neutron energy threshold counting (Crane 1978). The device, known as a neutron collar, used an AmLi (α, n) neutron source to induce fissions in the ^{235}U in the sample. The detectors used in the system were two ^4He gas detectors in the recoil counting mode. The electronic energy threshold on the ^4He detectors was set to eliminate most of the lower energy AmLi source neutrons (average energy approximately 0.5 MeV) and count the induced fission neutrons (average energy approximately 2 MeV), which were above the threshold. In this first collar, the two detectors were located on opposite faces near the center of the fuel assembly, while the AmLi source was on the face perpendicular to the two detectors at approximately the midpoint. The polyethylene holding the detectors and sources was tailored to provide good thermal irradiation of the fuel elements and, thus, provide uniform sensitivity to all of the rods in the assembly. This arrangement provided maximum sensitivity to the removal of a single rod in the assembly. The system used a source with a yield of approximately 5×10^5 neutrons per second. For a PWR fuel assembly with an enrichment of 3.19% and a fuel loading of 38.76 g of ^{235}U per cm, measured sensitivity to rod removal was about 1% for a 1000-second count. A single fuel assembly provides a much more compact geometry for this technique than does the present problem. The larger geometry in the shipping containers will produce a much more inefficient coupling between the interrogation neutrons from the source to the sample and the signal neutrons from the sample to the energy threshold detectors.

Testing of the threshold counting technique was performed using three-dimensional Monte Carlo calculations with Monte Carlo Neutron Photon (MCNP) code, version 3B. Cross-section data were from ENDF-B/V, a current compilation of nuclear data generally accepted in the nuclear community. For the modeling, the geometry in Figure 3.3 was used with an AmLi source in a 5-cm-radius polyethylene moderator located on top of the container and ^4He detectors (5 cm thick, 4 atmospheres pressure) located below and on each side of the container. The side detectors were 29 cm high by 35 cm long, and the bottom detector was 46 cm wide by 35 cm long. The fuel assemblies were modeled by "smearing" the 3.1% uranium fuel throughout the volume of the assembly.

Figure 3.2 shows the neutron fluence at the bottom detector with both assemblies in the shipping container and with source neutrons only - no fissions occur in the assemblies. Figure 3.3 shows the neutron fluence under three conditions, with a normally loaded shipping container, with one assembly replaced by air and with one-half of the fuel removed. Table 3.1 summarizes the fluence for the three detector positions under the same conditions shown in the figures. With a threshold of 1.5 MeV the background is negligible, but the signal from induced fissions is not directly related to the total fuel content. The bank of detectors next to the position that remains occupied throughout the sequence of measurements (Det. A) is influenced as much by the neighboring assembly as it is by the loss of one-half of the rods in the assembly. These calculations make it apparent that the accuracy for such assays will not be adequate for variables measurements. Removing the outer container and placing the source and detectors against the container increases

FIGURE 3.2. Fluence Below Loaded Shipping Container When Using AmLi Interrogation Source

FIGURE 3.3. Fluence Below Shipping Container for Three Loading Conditions

TABLE 3.1. Neutron Fluence at Detector Positions for Various Test Conditions
($n/cm^2 \times 10^{-7}/\text{source neutron}$)

Conditions	$E_n: 0.1 - 1.5 \text{ MeV}$			$E_n: > 1.5 \text{ MeV}$		
	Det. A	Det. B	Det. C	Det. A	Det. B	Det. C
A. Assemblies intact	17.1	11.1	19.1	7.61	6.67	7.61
B. Assemblies intact only source neutrons	7.65	0.811	7.65	0	0	0
C. Assembly missing	17.4	6.75	12.25	6.67	3.87	3.21
D. One-half rods missing from each assembly	17.1	9.12	16.9	6.46	5.70	6.36

the fluence by a factor of 20, but it is doubtful that the sensitivity to geometry will be improved enough to make this configuration successful.

If this approach is considered further, several improvements should be made in the calculations. The fuel assemblies should be modeled in more detail. Tests of fission versus source neutrons should be performed as a function of source moderator. The calculations may not be sufficiently accurate since polyethylene spacers are inserted between rods in the assembly and the packing materials are not continuous. Including such factors in the calculations would be costly and time-consuming.

The IAEA is currently evaluating a system very similar to the neutron collar for measurement of enrichment in large UF_6 cylinders. This system is known as the CIND system (Menlove and Sapir 1988) and is being developed as part of the French support program. Because the source and detectors are placed on the same side in the CIND system, the system samples only the surface nearest to the detector system (Menlove, Forster, and Matthews 1973).

The CIND system was investigated by Los Alamos (Menlove and Sapir 1988) for application to the assay of UO_2 -containing vessels. The system was evaluated by the MCNP transport codes, which showed that the system had a poor signal-to-noise ratio and was not sensitive to internal voids in the sample nor to substitution of depleted uranium within the sample. The lack of sensitivity of the system was due primarily to a slow change in counting rate with ^{235}U enrichment, compounded by the poor signal-noise ratio. The CIND system could produce a signal indicating the presence of fissionable material within the shipping containers but would not provide the accuracy needed for variables measurements. Again, if measurements are attempted with stacked containers, the neighboring containers will be at approximately the same distance and the signals will be impossible to discriminate.

^{124}Sb -Be Neutron Interrogation

Photo-induced neutron sources can produce fluxes of low-energy neutrons that are below the ^{238}U fission threshold. By using a photo neutron source such as ^{124}Sb -Be, the signal-to-background ratio can be improved because the average source neutron energy is lower than that for $AmLi$. The Sb -Be-type system, which is known as the PHONID instrument, has been developed in Europe and is used for the assay of ^{235}U -containing samples (Filss 1975; Menlove, Forster, and Matthews 1973).

The photo neutron system has several disadvantages that make it impractical for routine fuel assay situations. Antimony-124 has a short half-life (60 days). Other source materials are available (Kull, Beyster, and Schillaci 1971) but have only slightly improved half-lives and still have the shielding problems associated with the antimony source. The exposure rate from the bare antimony source is greater than 1 R per hour but the source combination does have the advantage that the neutron output can be "turned off" by removing the antimony from the proximity of the beryllium. The gammas from the antimony source are high energy (0.603 MeV, 1.69 MeV) and will require heavy lead shielding for personnel protection. The photo-neutron systems are usually tailored for smaller samples, and because of the

disadvantages of having to shield the sources in the field, this technique will not be considered further in this report. In addition, the short half-lives would make it difficult to keep a source of adequate strength on station for the periodic physical inventories.

Discrimination by Attenuation

Another technique that has been used to discriminate between interrogation source neutrons and the fission neutrons is to take advantage of the smaller mean free path of the interrogation neutrons in an attenuating/absorbing medium (Filss 1975). One scheme would be to place an interrogation source on top of the shipping container and place a thermal neutron detector inside a moderator (~6 cm thick) beneath the container. Although the shipping container does contain low-density moderating material, it does not have any absorption properties. A thin (~1-cm) moderator with thermal absorber (borated plastic) would be placed on top of the counter to assist in attenuating the source neutrons. Intuitively, this scheme will have a poor signal-to-noise ratio. If performed in the stack of containers, there will be interference from spontaneous fission neutrons and induced fissions in other assemblies. If oriented toward using an individual container, the concept may warrant additional consideration since it may be possible to adapt counter banks from a uranium neutron coincidence collar (UNCL) for this application and partially reduce the heavy penalty on coincidence counting caused by the poor geometry. The (α, n) and spontaneous fission backgrounds could be corrected by making counts without the interrogation source.

3.3.2 Coincidence Techniques

Neutron coincidence counting techniques have become very important for safeguards measurements since they provide a mechanism to identify the signature from fissile materials. When a material fissions, it emits two or more neutrons in coincidence, plus several high-energy gamma rays. By counting the coincidence neutrons one can identify events occurring in fissile material with certainty and effectively separate this signature from that of (α, n) reactions occurring in the material. The technique works well with materials that have a high spontaneous fission rate, such as ^{240}Pu .

The high-level neutron coincidence counter (HLNCC) is an effective implementation of this technique for plutonium samples and is widely used by the IAEA for assay of plutonium samples. However, for materials with a low spontaneous fission rate, it is necessary to irradiate the samples with thermalized neutrons to induce fissions. The HLNCC can be adapted to uranium by using a low-energy neutron source and moderator to irradiate the material with thermalized neutrons. Configurations similar to the HLNCC have been used for such applications when there is a need to assay small containers of enriched uranium. The UNCL is an effective application of the neutron coincidence technique to fuel assemblies (Menlove and Keddar 1983). In the standard configuration, the UNCL is closely coupled to the fuel assembly under assay. The UNCL has been very carefully calibrated (Menlove and Pieper 1987) in order to provide highly accurate results (better than 6%). The standard UNCL can be configured for either PWR or BWR fuel. A UNCL system has been installed at the JNF facility for assay of fuel assemblies before storage and shipment.

The UNCL cannot be directly adapted to the present measurement problem and still maintain its calibration and accuracy. However, two schemes could be used with the UNCL to perform assays on fuel assemblies and shipping containers on a "relative" basis. First, the containers could be separated from the stack and detector banks from the UNCL positioned at the sides and bottom of the shipping container. Second, two or three of the banks could be positioned underneath the shipping container with the source positioned above the shipping container in the gaps between the stacked containers. The first technique should produce the highest sensitivity to assemblies in the container and, thus, the most accurate results. The techniques are referred to as relative since it would be impractical to perform the extensive calibration done on the UNCL for the present application; thus, comparing measurements on one container with a master container yields a relative result. Thus, the master container would act as a standard and could be verified by opening and making separate measurements on the assemblies.

In order to obtain the maximum accuracy from the UNCL technique, corrections must be made for counts from spontaneous fission neutrons (^{238}U) and for the influence of poison rods. Counts can be made without the source to obtain a background count rate. In the standard UNCL configuration, this can be used for independent verification of the ^{235}U content. If counts are attempted in the stacked containers, this will not be useful due to interfering counts from adjoining containers.

Corrections for poison rods are generally based on the operator's declaration of poison rods, but by lining the UNCL with cadmium, counts can be performed to verify the operator's declaration. This will not be possible with the shipping containers due to the large amount of moderating materials within the containers. An error in the declaration of the number of poison rods by one rod could change the corrected coincidence rate by approximately 3.5%, which could result in a change in the assessment of the ^{235}U linear density by approximately 4.5%.

As a reference in evaluating the feasibility of such a scheme, Menlove and Sapir (1988) evaluated an expanded UNCL technique for application to assay of UO_2 containers. When expanding the UNCL configuration from its standard 27-cm-x-27-cm configuration around a PWR assembly to a 40-cm-x-40-cm configuration around the same assembly, they found that the coincidence count rate would drop from 116 to 9.3 counts/sec. This is an order of magnitude decrease in count rate and it is estimated that one could expect at least an additional order of magnitude decrease in going from this configuration to an approximately 80-cm-x-80-cm configuration, as represented by the shipping containers. The linear ^{235}U density for two side-by-side BWR assemblies is slightly less than the linear density for a single PWR assembly. This will reduce the signal, as will the different geometry. If detectors are placed at the sides of the container, the distance from the assembly to the detector will approximately double while the distance from the bottom of the assembly to a detector will approximately triple. Thus, the count rate for isolated containers will be higher than count rate for containers that are still in the stacks. It is difficult to accurately evaluate the effects of the expanded configuration without making actual tests on assemblies in a storage container since the accidental rate is hard to predict.

A random driver (Rogers 1983) operates on the same principle as the UNCL, but uses scintillation detectors. Such a system has been used with 5-gal. drums and 4 ft³ filter packs providing an accuracy of 2.4% and 3.5%, respectively. Adaptation of such a system to the present problem would lead to a significant loss of accuracy due to the size of the sample.

Prompt Gamma

In addition to the fission neutrons, approximately 6-7 prompt gammas are emitted during fission. The gammas do not have good penetrability and are generally discriminated against in assay techniques (Hines et al. 1987). In addition, there will be a high gamma background from the neighboring assemblies and from the interrogation source. The technique will not be considered further in this report.

3.3.3 Delayed-Neutron Interrogation

The major problem with active neutron interrogation techniques is the separation of the interrogation neutrons from the induced fission neutrons, which constitute the desired signature of the assayed material. A very effective technique for performing the separation is to count delayed fission neutrons after the interrogation source has been removed from the vicinity of the sample. The source is generally placed next to the sample, then removed to a shielded position while counts are taken of the sample to detect the delayed neutrons. The source is then returned and removed to a shielded position, and the sample recounted. Because the neutron source is repeatedly placed in position and removed, these systems are known by the general term of "shufflers" (Menlove and Crane 1978; Crane 1978). The systems generally use ²⁵²Cf neutron interrogation sources together with ³He detectors to count the induced delayed neutrons. The technique has been adapted to many different problems (Eccleston et al. 1983; Stein et al. 1979) and can measure ²³⁵U content down to a few grams. The estimated accuracy for a ²⁵²Cf shuffler system on a large sample (spent light water reactor fuel) is 5% (Rogers 1983). Accelerators can be used in addition to sources and in one unique application both prompt and delayed neutrons were counted (Matsuda, Tamura, and Murata 1978). However, accelerators are not practical for the present application. The technique has not been applied to packages as large as the shipping containers and would require the careful design of a sample chamber and counting heads. Shufflers require a ²⁵²Cf source, a large shield, and a mechanical system to transfer the neutron source and are expensive (\$150-300K; Rogers 1983). Such a system is not portable and components of the system are not in the Agency's inventory of equipment. However, some of the neutron-counting systems could be adapted for shuffler-type measurements. Because of the complexity of the shuffler system, the present problem would not warrant the investment required for development of such a system. Thus, the shuffler system will not be considered further. If the access tube had been built into the shipping containers (Section 1.3), the shuffler approach would have warranted additional consideration.

3.3.4 Differential Die-Away System

In recent years, the differential die-away technique has been developed for assay of small amounts of fissionable materials in waste containers

(Caldwell et al. 1986). The method uses a pulsed neutron generator which produces 14-MeV neutrons that are thermalized to interrogate the samples. The system operates in a pulse mode and the induced fission neutrons are counted between interrogation pulses using ^3He detectors inside cadmium shields. This technique can measure samples containing very small quantities of ^{235}U , but thermal neutron interrogation leads to lack of penetration for high mass samples such as the dual BWR assemblies in a large storage container. The disadvantages of the differential die-away technique are similar to that of the shuffler system. The system requires specialized equipment, a design for the specific application, and large amounts of neutron shielding. The system would definitely not be portable and does not use equipment now available in the IAEA inventory. Because the system would be complex and costly, the present application does not warrant further consideration of this technique.

3.4 FUEL LENGTH VERIFICATION

The neutron assay techniques described above will give a relatively accurate measurement of the linear ^{235}U fuel density. However, determining the total mass of ^{235}U within the storage or shipping container will require some technique to measure the active length of the fuel. The fuel length could be measured by several techniques, although not all appear to be practical. For example, one technique would be to develop large detectors that could enclose the shipping container and do a total assay of the ^{235}U content. However, this is not practical due to the size of the shipping containers. As an alternative, the entire length of the shipping containers could be scanned through an active system to determine the total fuel loading. However, this also is not practical since it would require moving the entire shipping container past the active source and detectors or alternatively moving the source and detectors past the container at a uniform rate. The technique that appears to be most promising is to use gamma scanning to determine the total ^{235}U active length of the fuel assemblies.

When the shipping containers are removed from the storage stack, it is possible to use a right cylindrical collimator with a large aperture to provide relatively high count rates but also with a large length-to-diameter ratio to provide good resolution. By scanning such a detector along the shipping container and plotting activity in the 185.7-keV photopeak versus position, it is possible to determine the two ends of the fuel column and thus to determine the length of the fueled region of the assembly. The inspector could use the IAEA's HM-4 for this measurement (see Menlove and Pieper 1987). One could expect to obtain an overall accuracy in fuel measurement on the order of ± 2 cm. The accuracy obtained for PWR fuel assemblies is approximately ± 1 cm or $\pm 0.3\%$. The reduced accuracy in our case is due to the lower count rates anticipated and the resulting inaccuracies of determination of the midpoint of the slope in count rates at the two ends of the assembly. It would be possible to improve the accuracy of the fuel length measurement by careful attention to the collimator design and by counting longer. An estimated accuracy of ± 2 cm in the overall fuel length could introduce an error of $\pm 0.5\%$ in the fuel assay.

3.5 REMOTELY READ SEALS

Another scheme to aid in verifying fuel assemblies is to attach a remotely interrogated device to each assembly that can be identified during an inspection, while the assembly is still in its shipping container. Several models of passive transponders are available that can be read a few inches from the reader. A typical molded and sealed transponder, which could be preprogrammed with a unique identification number, is about 3.5 in. long, 2.125 in. wide, and 0.2 in. thick and weighs 0.7 oz. The unit cost of these transponders in volume is \$15.50 (U.S.). Equipment to read the transponders in the field would cost approximately \$1280. The sensor would have to be placed close to the location of the transponder but would communicate the unique identification number to a computer several feet away.

The scheme envisioned would be to attach a transponder to the bail of the fuel assembly with a length of sealing wire using a metal cap seal. Use of a VACOSS seal may be needed to meet safeguards criteria. The transponder would be read by a sensor at the installed neutron collar during assay of the assembly. Thus, a unique identification number would be associated with the assay results for each assembly. It is understood that presently the facility operator conducts the assay of each assembly using an installed neutron collar to the facility. When the assembly is loaded into the shipping container, the facility operator would route the transponder with attached wire past the gasket seal on the inner metal container and mount the transponder on the surface of the lid of the inner container in a holder. It may be necessary to design a special silicon gasket bridge to maintain integrity of the seal on the inner container. Since the transponders for the assemblies would be in known positions, it would be possible to quickly pass a sensor over each position and read the identification number while the assemblies are in the storage area. This would provide a verification of the presence of the assemblies and could be backed up by passive gamma analysis (see Section 3.1) on selected assemblies. When the assemblies arrive at the destination, selected assemblies could be remeasured with a neutron collar in a variables mode, thus providing assurance that the assembly had not been disturbed between the time it was assayed at the fabrication facility and its arrival at the destination. The transponders would have to be recovered by IAEA inspectors and read out to make sure that they corresponded to the proper assembly.

The transponder and reader can be provided with custom firmware at a cost of \$5000. This involves a custom readout algorithm that will make it extremely difficult for a facility to substitute transponders. The transponder can contain 16 ASCII characters for identification and has a seed number that can also be changed for security purposes. Reading distance is limited to approximately 30 cm and is based on U.S. Federal Communication Commission limits.

Based on the transponder, it would take between 20 and 30 minutes to read the transponders in an array of 40 assemblies in 20 boxes stacked four high and five wide.

3.6 WEIGHING

Another technique that can be considered for analysis of shipping container contents is weighing. Load-cell-based weighing systems are used for UF_6 cylinders (Fainberg et al. 1983) and provide good precision in this application. Load-cell systems can be used with a fork lift and thus could be used to weigh shipping containers. Since this provides no information on the shipping container contents and, unlike the UF_6 cylinder case, many low value materials could be substituted for the assemblies, this option will not be considered further.

4.0 CONCLUSIONS AND RECOMMENDATIONS

Table 4.1 summarizes the main features of the various techniques considered. The techniques of passive gamma counting (^{235}U), active neutron interrogation (AmLi), and remotely read seals emerge as the most promising techniques.

The remotely read seals are the least intrusive and most promising technique if they can be operated as part of a system. They appear to meet requirements as a sealing technique if (1) Agency inspectors recover (remove) the seals and (2) the Agency accepts operator attachment of the seal. The technique will be applicable for a short period unless a radiolucent window can be placed in the outer shipping container when the new metal containers are put into service.

The passive gamma technique is adequate as a low-accuracy technique for attribute testing. Measurements would be required on eight assemblies (randomly selected) in our test stack of 40 assemblies. This could be done rapidly by constructing a gamma-counting system based on a portable multi-channel analyzer (MCA). The proposed detector would be a BGO crystal, 1 cm wide by 0.5 cm thick and 5-10 cm long, in a tungsten shield with a 1.0-cm slit collimator containing a removable 1 mm thick septum to improve collimation. The shield would be approximately 8 cm in diameter. Preparation of the engineering drawings and specifications plus procurement of a prototype detector will cost an estimated \$13-15K. This will not include any analysis software, but will include an evaluation of the detector and preparation of documentation. The detector could be positioned either manually or with a positioning mechanism that could be controlled by the MCA. The remote positioning feature would add approximately \$5000 to the hardware costs plus about \$15-20K for software development costs.

This still leaves the Agency without an accurate measurement technique that could be used for variables measurements. The number of assemblies required for variables measurements was not estimated since the accuracy of the techniques is not well known and the composition of the overall inventory of the facility was not known.

The active neutron technique appears to be the most promising variables technique and should be investigated further. Additional calculations should be performed to improve the design of the source moderator and this should be followed by test measurements on BWR fuel assemblies in containers. Tests on coincidence counting could be performed concurrently. However, coincidence counting does not appear to be promising. The accuracy of the neutron techniques will always be limited by the inability to correct for poison rods.

The poor coupling (geometry) of sources and detectors to the assemblies in the shipping containers reduces the sensitivity of the neutron techniques. Calculations performed with a Monte Carlo code using interrogation neutrons from an AmLi neutron source indicated poor sensitivity to changes in total fissile content within the container. If a variables technique is to be successful, it will require separation of the selected shipping container from the stack and (in all probability) removal of the outer container. If the

TABLE 4.1. Comparison of Assay Techniques(a)

Technique	Accuracy	Intrusiveness	Costs	
			Implementation	Operation
Passive Gamma ^{235}U ^{238}U	Low na	Moderate Moderate	Low Low	Low Low
Passive Neutron Totals Coincidence	na na	High High	Moderate Moderate	High High
Active Neutron AmLi interrogation $^{124}\text{SbBe}$ interrogation	High-moderate High-moderate	Moderate High	Moderate High	Moderate High
Coincidence counts Delayed neutron	Moderate High	Moderate High	Moderate Very high	Moderate High
Differential die-away Prompt gamma	High Low	High Moderate	Very high Moderate	High Moderate
Remotely read seals	na	Low	Low	Low

(a) See Section 2.0 for explanation of terms.

IAEA wants to pursue this area, it is recommended that the feasibility study be extended beyond the scope of this effort. Calculations should explore the feasibility of detecting a few missing rods, and experiments should be performed with actual containers to verify the calculations. Such studies are necessary before attempting design of a detector system.

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APPENDIX

CORRESPONDENCE WITH JNF

JAPAN NUCLEAR FUEL CO., LTD.

3-1-2 CHOME, UCHIKAWA, YOKOSUKA-SHI, KANAGAWA-KEN, JAPAN

ZIP CODE: 239

FAX (0468) 35-6951-1141

TELEX: 345167 JNF J

RECEIVED K. L. SWINTH AUG 14 1989
FILE
HOLD
ACTION
CIRCULATE TO
TICKLE FILE
XEROX

August 11, 1989

Mr. Kenneth L. Swinth
Technical Group Leader
Dosimetry Technology Section
Health Physics Department
Pacific Northwest Laboratories
P.O. Box 999
Richland, Washington 99352
U. S. A.

Dear Mr. Swinth:

I thank you for your letter to me dated July 31, 1989.
Enclosed please find informal paper concerning measurement technique of fuel assembly in shipping container.
In May 1989, Mr. Abedin-Zadeh of IAEA was observed this measurement at our fuel fabrication facility and confirmed that a number of fuel assembly in the shipping container and gross defect can be verified without opening the container by this technique, but partial defect.

On the other hand, now we are being designed new metal shipping container and anticipated to replace with current wooden container by April 1992.

As you mentioned in your letter, I was proposed to place some guide tube through outer container to inner container in order to insert NaI or Cadmium detector along with the surface of the fuel assembly to achieve more precise measurement for partial defect verification but this idea was rejected due to difficult to meet design license criteria.

I have received your workplan prepared for A.164 from Mr. Tom Shea of IAEA when he was in Tokyo on July 23, 1989 and explained that we will provide new container information upon design is completed which is scheduled December 1989.

If you need any further information in this regards, please contact me at your convenience.

Yours sincerely,

Takeshi Osabe, Deputy Chief Engineer
Quality Assurance Department
Japan Nuclear Fuel Company

Enclosure

DEVELOPMENT OF VERIFICATION METHOD ON FUEL ASSEMBLY
IN SHIPPING CONTAINER BY NDA

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1. Introduction

In order to achieve inspection goal on the IAEA safeguards, physical verification by identification number and non-destructive analysis (NDA) or so on for randomly selected fuel assembly is stipulated.

At Japan Nuclear Fuel Company (JNF), two (2) fuel assemblies are stored in each shipping container which is composed with metal inner container and outer wooden container after produced and inspected, then maximum of 500 containers are stacked of 5 containers at shipping container storage area until ship to the reactor site. In this circumstance, opening of the shipping container to verify identification number and NDA for randomly selected fuel assembly will provide significant interfere to JNF's operation and therefore achievement of inspection criteria for fuel assembly in the shipping container is difficult at this facility.

In order to carry out domestic safeguards inspection activities with effectively and efficiently, and to achieve the inspection goal on IAEA safeguards, Science and Technology Agency and Nuclear Material Control Center were developed NDA method to apply verification of fuel assembly in the shipping container on the store condition, under cooperation with JNF.

There is a significant limitation on measurement condition (e.g. space of the measurement, thickness of lead shield) to implementing NDA measurement for shipping container under such store condition. For this purpose, 3 different type of relatively small γ -detectors were selected and finally selected one γ -detector to be used for this study by measurement of these characteristics.

As a result, we were concluded that the detector which is composed with NaI(Tl) Crystal and photo-multiplier to be used for this study. By means of this detector, measured 186 keV gamma intensity emitted from the uranium containing in the fuel assembly was measured from the outside of shipping container.

As a basis of intensity and distribution of gamma-ray obtained by these measurement, development and discussion has been performed in order to make it possible on attribute test, or variable test with attribute mode and to verify that either two (2) fuel assembly in the container.

In July 1987, a primary test was conducted at JNF and the feasibility study was performed in August 1988 by means of improved equipment.

As a result of this study, we were concluded that the number of fuel assembly in the shipping container can be discriminated either one (1) fuel assembly, or two (2) fuel assemblies or empty without move the shipping container from the storage where is stacking at the shipping container in the shipping container storage area, and it can be achieved inspection goal (current SIR criteria) for fuel assembly in the shipping container.

* Currently Assigned to Power and Nuclear Fuel
Development Corporation.

2. Selection of NDA equipment

In JNF, two(2) fuel assemblies are stored in the shipping container and packed with bolt and steel belt as shown in Fig-2, then normally will be stored by stacked 4 to 5 containers (Approx. 3.3m - 4.2m high) until the shipment. In this condition, the space allow to setting of detector is approx. 10 cm of gap between the stack of the shipping container or 80 cm at both side of the shipping container.

In case of apply NDA technique to these double structural shipping container, the active neutron measurement method shall be considered in general, however, development of measurement equipment and radiation shield adequate for these measurement space is rather difficult.

On the other hand, detector for passive method can be made it lighter and smaller, radiation shielding is relatively easy and cheaper in compare with active method.

In consequent of this, passive γ -measurement method composed with small γ -detector and pulse height analyzer was used. Then three(3) different types of small γ -detectors described below were studied with consideration of availability on the market and technically usable.

i) Cadmium Telluride Detector (CdTe)

- Radiation Monitoring Devices Inc. USA: S101.2/P4 model
- Crystal size: 8 mm ϕ x 2 mm
- Detector size: Approx. 14 mm ϕ x 25 mm

The size of this detector is very small and appropriate to move in the narrow space. But low detection efficiency against energy around 186 keV, because 2.2 mm of crystal thickness, and it was insufficient for short time measurement as this study.

ii) Detector composed with CsI(Tl) crystal and photo-diode (PD)

- Horiba Mfg. Co.: CsI (PD)-242 model
- Crystal size: 1/2" ϕ x 1/2"
- Detector size: Approx. 25 mm ϕ x 50 mm

This detector is relatively larger than Cadmium Telluride Detector, but smaller than one use photo-multiplier, and suitable to use at high moisture area because low electric voltage required for detector. But it was unsuitable for this study because a noise from photo-diode itself (affect up to 100 keV).

iii) Detector composed with NaI(Tl) Crystal and Photo-Multiplier

- Ooyu Koken Kogyo: SP-05 Model
- Crystal size: 1/2" ϕ x 1"
- Detector size: Approx. 26 mm ϕ x 220 mm

The disadvantage of this detector is slightly larger than above detector and required high voltage for the detector, but it has excellent mutual interaction on energy resolution and detection efficiency. Therefore, we were decided to use this detector for this verification test.

3. Test method

3.1 Selection of measurement point

In the condition such as 4 to 5, shipping containers are being stacked (Fig-3), approx. 10 mm of space between the upper and lower container, except upper container and approx. 80 mm of space at the both side of the shipping container are exist. Therefore following three(3) alternatives can be considered as to detector setting position for this situation.

i) Measure at underneath of the shipping container by means of 10 cm of the gap between the upper and lower container.

ii) Measure at top of the shipping container by means of 10 cm of gap between the upper and lower container.

iii) Measure at upper or lower end of the fuel assembly by means of 80 cm of space at the both side of the container.

A measurement point described iii) above is most suitable for the detector setting and operational point of view. However, the measurement was difficult because the distance is be off from the surface of the container to the fuel assembly and γ -ray (186 keV) was shielded by the natural uranium which is loaded in the both end of the fuel assembly produced by JNF. Also as a results of the preliminary study, a distribution of gamma intensity could be obtained more clearly at position ii) above because position of the fuel assembly at i) above is slightly be off than position ii) above. In consequent of this we were judged that measurement point ii) above is most suitable.

3.2 Measurement method

Under this store condition as described above, following diversion scenario can be postulated.

i) Place container with only one fuel assembly into the part of the stack of shipping containers.

ii) Place empty container into the part of the stack of the shipping containers.

In consideration of these diversion scenario, three(3) different stack pattern such as A, B and C as shown in Fig-3 was prepared. A is denote the normal store pattern that two (2) fuel assemblies are stored in each shipping container. B is place the shipping container with only one (1) fuel assembly which is considered of diversion scenario of i) above. And C is place the empty shipping container which is considered of diversion scenario of ii) above.

In order to perform measurement by sliding the NaI(Tl) detector, placed detector sliding rail at near center part in the 10cm of gap between the stacked containers at near center part of longitudinal side of the shipping container. Attache the detector to the lead collimator (Fig-1) and put the detector on the rail, then scanned 5 cm intervals from the end of the shipping container. At each measurement point (upper surface of the shipping container), γ -spectrum from fuel assembly was measured 200 sec. by means of multichannel pulse height analyzer, and total of 51 points were measured at the surface on the movement of the lead collimator (A1, A2, B and C on Fig.-3) which is equivalent to 3 shipping containers.

A position of the collimator and the distance of movement during measurement was accurately measured by means of attach the tip of Convex Rule to the collimator. Also attached

chemical illuminant to the collimator body in order to confirm the position and to make adjustable of "O" point through the narrow gap between the shipping container. The lead collimator used to this study was made by covering the cylindrical lead shield with 2mm thick stainless steelplate and cut out 10mm slit with 40° angle as shown on Fig -1. A γ -ray spectrum data which is emitted from the uranium containing in the fuel assembly was measured by above detector, then obtained gross and net count of photoelectron peak at 188keV by data analyzer. Also these data were stored in 8" floppy disket with floppy disk drive attached to data analyzer.

3.3 Date and equipment used.

Place of the study : JNF shipping container storage

Date : August 29 - 31, 1988

Equipment used :

- i) Gamma Ray Detector = NaI(Tl), Made by Ooyo Kouken
Kougyou Ltd. (SP-05, 1/2" ϕ x 1")
- ii) Movable lead collimator with rail = Refer to Fig-1
- iii) Multichannel Pulse Height Analyzer = NAIG E-Series
- iv) Data analyzer = Desk top electronic calculator
HP-9845B
8" Disk Drive HP-9895A

4. Measurement results

Intensity of 188keV γ -ray (Gross and Net count) emitted from the uranium loaded in the fuel assembly at each measurement point for each stack (A, B, C, in Fig-3) is shown in Fig.5-8. And measurement results of the scanning surface at A, and B, is shown in Fig. 5-6. In consequent with this, it is clarified that the distribution of gamma intensity (net count) shows no significant influence to the measurement even the shipping container stored with fuel assembly is stacked upper and lower side of the shipping container which is the subject to the measurement. Also, as a results of the measurements shown in (Fig.7-8), we were concluded that discrimination of the assembly in the shipping container either two(2), or one(1) or no assembly in the shipping container can be sufficiently achieved from the configuration and peak position of gamma ray intensity (net count) distribution even the shipping container is being stacked.

Approx. 3 hours were required for measurement at one moving direction (200 sec. for each measurement point, measured 51 points equivalent to three(3) shipping container for each movement direction).

In consequent of these measurement results, an attribute verification for fuel assembly in the shipping container became available, and current IAEA SIR criteria can be achieved by means of this measurement method at the inspection.

5. Conclusion

Following conclusion was obtained by this feasibility study

- 1) Influence to the measurement by other fuel assembly in the shipping container .

It is clarified that there is no significant influence to the measurement data even shipping container with fuel assemblies are stacked upper and lower side of the shipping container to be measured.

- 2) Discrimination of the number of fuel assembly in the shipping container

A number of fuel assembly in the shipping container can be sufficiently discriminated by the distribution of 186keV gamma ray intensity(net count), and peak position and height, even under such condition that shipping container is being stacked.

As a results of above, an attribute verification became available by analyzing the distribution of gamma ray intensity, peak position and height which was obtained by this study and concluded that achievement of the inspection goal for the fuel assembly in the shipping container at JNF is possible.

Hope these results will be reflected on the inspection activity at JNF and use as a verification technique for verification of fuel assembly in the shipping container.

Lastly, we are greatly appreciates the cooperation extended for this study by related personnel of Safeguards Office of Science and Technology Agency and Japan Nuclear Fuel Company.

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FIG. 1 SCHEMATIC DRAWING OF MOVABLE LEAD COLLIMATOR

(A)

(C)

(B)

FIG.-3 SCHEMATIC DRAWING OF SHIPPING CONTAINER STACK AGAINST EACH MEASUREMENT SURFACE

*10~1.

FIG.-4 EXAMPLE OF MEASURED GAMMA SPECTRUM

FIG.-5 186keV GAMMA DISTRIBUTION AT LEAD COLLIMATOR MOVEMENT A₁

FIG.-6 186keV GAMMA' DISTRIBUTION AT LEAD COLLIMATOR MOVEMENT A₂

FIG.-7 186keV GAMMA DISTRIBUTION AT LEAD COLLIMATOR MOVEMENT B

FIG.-8 186keV GAMMA DISTRIBUTION AT LEAD COLLIMATOR MOVEMENT C

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