*Design Study for MOX Fuel Rod Scanner for
ATR Fuel Fabrication
Phase I: Design of Active Neutron Scanner
Phase II: Design of Passive Neutron Scanner
Phase III: Design of Passive Gamma-Ray Scanner*

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Phase III: Design of Passive Gamma-Ray Scanner

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H. O. Menlove

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DESIGN STUDY FOR MOX FUEL ROD SCANNER FOR ATR FUEL FABRICATION

PHASE I: DESIGN OF ACTIVE NEUTRON SCANNER
PHASE II: DESIGN OF PASSIVE NEUTRON SCANNER
PHASE III: DESIGN OF PASSIVE GAMMA-RAY SCANNER

by

G. W. Griffith and H. O. Menlove

Abstract

Phase I: Design of Active Neutron Scanner*

An active neutron fuel-rod scanner has been designed for the assay of fissile materials in mixed oxide fuel rods. A ^{252}Cf source is located at the center of the scanner very near the through-hole for the fuel rods. Spontaneous fission neutrons from the californium are moderated and induce fissions within the passing fuel rod. The rod continues past a combined gamma-ray and neutron shield where delayed gamma rays above 1 MeV are detected. We used the Monte Carlo neutron-photon (MCNP) code to design the scanner and review optimum materials and geometries. An inhomogeneous beryllium, graphite, and polyethylene moderator has been designed that uses source neutrons much more efficiently than assay systems using polyethylene moderators. Layers of borated polyethylene and tungsten are used to shield the detectors. Large NaI(Tl) detectors were selected to measure the delayed gamma rays. The enrichment of the fuel zones of a thermal reactor fuel pin could be measured to within 1% for practical rod speeds. Applications of the rod scanner include accountability of fissile material for safeguards applications, quality control of the fissile content in a fuel rod, and the verification of reactivity potential for mixed oxide fuels.

Phase II: Design of Passive Neutron Scanner

A passive neutron fuel-rod scanner has been designed for the assay of the plutonium in mixed oxide fuel rods. The ^{240}Pu -effective is measured by counting the spontaneous fission neutrons using a high-efficiency thermal-neutron detector. We used the MCNP code to design the ^3He tube-based neutron detector. Moderator materials including polyethylene, graphite, and Teflon were evaluated to obtain the maximum counting efficiency. The optimized design was a composite moderator detector using 18 ^3He tubes that provide an efficiency of 52% for counting ^{240}Pu

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neutrons. This passive neutron detector would be combined with a high-resolution gamma-ray system (HRGS) measurement to obtain the total plutonium from the plutonium isotopic ratios. The system would be used by the International Atomic Energy Agency (IAEA) and the operator in the attended mode to verify the plutonium loading in each of the fuel zones to approximately 1% accuracy.

Phase III: Design of Passive Gamma-Ray Scanner

A passive gamma-ray scanner has been designed for the measurement of the ^{241}Am and plutonium uniformity in mixed oxide fuel rods. The passive gamma-ray emissions from ^{241}Am (60 keV) and plutonium (150-400 keV) are used to verify the uniformity of the fuel enrichment zones and to check for any pellets that are out of specification. The fuel rod is moved through the interior of an NaI(Tl) or a bismuth germanate detector to measure the passive gamma-ray emissions. A tungsten sleeve collimator is used in the through-hole to improve the pellet-to-pellet spatial resolution. We used the MCNP to design the detector geometry and shielding, and to predict the performance of the detector system. The counting statistical performance as a function of the rod speed and collimator width was calculated. A dual-range detector was designed where the ^{241}Am uniformity is measured using a rectangular NaI(Tl) detector (1.3 x 5.08 x 5.08 cm) with a 2-cm-diameter through-hole. A tungsten collimator in the through-hole has a width of 1-2 mm to provide excellent spatial resolution for the individual pellets and the fuel zone boundaries. The same detector is used to verify the plutonium uniformity in the pellets with a 13-mm tungsten collimator. The low-resolution passive gamma system would be used in the unattended mode.

Calculations and measurements were made for HRGS that would be placed in the side of the passive neutron detector. This detector would be used to measure the ^{241}Am and the plutonium isotopics for a multi-pellet fuel zone. The same fuel zone would be simultaneously measured in the passive neutron detector. A mockup experiment was made with an HRGS to compare the measured isotopic ratios with theoretical predictions. The HRGS measurements would be made by the IAEA in the attended mode with the fuel-rod stationary.

PHASE I: DESIGN OF ACTIVE NEUTRON SCANNER

I. INTRODUCTION

Reactor fuel rods can be assayed nondestructively in several ways. The enrichment of uranium oxide fuel can be measured passively using the gamma rays from ^{235}U . The enrichment and a known fuel density can be used to calculate the fissile fuel loading in each rod. Plutonium oxide fuel can be measured passively for plutonium isotopics using gamma rays and then measured for plutonium mass using a passive neutron detection system.¹ However, passive gamma-ray counting times are excessive for on-line service. Furthermore, if the fuel contains both uranium and plutonium oxide (mixed oxides, MOX), the ^{235}U is difficult to measure using passive gamma-ray techniques because the ^{235}U gamma-ray peak is lost within the coincident plutonium spectra. The long count times required to measure ^{235}U would make sampling every fuel rod in a production facility impossible.

An active neutron rod scanner avoids the long count times required for uranium measurements in passive MOX rod scanner systems. Californium-252 is used as a source of spontaneous fission neutrons. The source neutrons are thermalized by the core of the fuel-rod scanner. The thermalized neutrons create fissions in the fuel rod as it passes through the scanner near the neutron source. As the rod passes out of the scanner and past a shield, the delayed gamma rays from the fission products are measured by gamma-ray detectors. Gamma rays above 1 MeV are counted. The delayed gamma rays are of much higher energy than the decay gamma rays and x-rays of plutonium.² The effective fissile content is directly measured using neutron interrogation. Robust room-temperature detectors can be used, which makes the system more reliable and simple to maintain. The similar response of plutonium and ^{235}U to thermal neutron interrogation makes separating the contribution of each element to the final signal more difficult and dependent on other measurements. The plutonium signal can be separated with passive gamma-ray measurements and passive neutron counting.

A disadvantage of active rod scanners is the high cost of the ^{252}Cf source, moderating materials, and the shielding materials required to achieve reasonable efficiency. The ^{252}Cf decays with a 2.64-year half-life, making source replacement a recurring cost. In addition, the cost and regulation associated with transporting strong neutron sources makes regular source replacement difficult. This is of particular interest for safeguard applications where overseas shipments of ^{252}Cf are required. The money invested in moderating materials for the rod scanner is a nonrecurring cost that justifies reasonable expenditures on higher-performance materials.

These factors helped define the criteria for the rod scanner design. The design was optimized to create the most fissions per source neutron so the smallest ^{252}Cf source could be used. The source strength is also greatly reduced because we are counting rod zones instead of individual pellets in the active mode. For MOX fuel, passive gamma-ray counting can be used to identify any abnormal pellets. The design-basis fuel was MOX for light-water reactor (LWR) recycling with a uranium enrichment of 0.7%–2% and a plutonium loading of 1–4 wt%.

II. PREVIOUS WORK

Active neutron interrogation systems have been used for the nondestructive assay (NDA) of several different types of fuel rods during the fabrication process for both quality control and safeguards. The initial rod scanning systems for boiling-water reactor (BWR) and pressurized-water reactor (PWR) fuel rods used large ^{252}Cf sources of $\sim 1\text{--}10\text{ mg}$ to accommodate the high fuel rod throughput rates of fabrication facilities.³

The LWR rod scanners were followed by a more sophisticated rod scanner for MOX fuel rods. A combined passive-active (decay-delay) gamma-ray scanner was built to assay fast breeder reactor (FBR) fuel and fuel for the Fast Flux Test Facility (FFTF).⁴ Fast neutrons were used to better penetrate the FFTF fuel rods.

The FBR rod scanner measured passive gamma rays as the rod was inserted through the detectors and into the scanner. Once the rod had been activated within the scanner, the fuel rod was withdrawn back through the detectors to measure the delayed gamma rays. The average error for a rod scan was less than 1% in the mass of fissile material. Approximately half the error came from counting statistics and half from the variance in the standards used to calibrate the system.

III. MODERATOR MATERIALS AND DESIGN

The first step in the present design was to survey the practical moderator materials for the rod scanner. Industry practice centers around polyethylene, carbon, and beryllium as moderating materials. The radial neutron flux profiles in a 30-cm-radius sphere for these moderating materials are shown in Fig. 1. The fluxes were calculated and benchmarked previously using the computer code Monte Carlo neutron photon (MCNP).⁵

Fig. 1. Neutron flux in sphere as a function of radius for different moderating material.

The polyethylene shows a much greater flux peak at the center than beryllium or carbon because the high density of hydrogen in CH_2 efficiently slows neutrons to thermal energies. Polyethylene has a distinct disadvantage because the large capture cross section of the hydrogen rapidly absorbs the neutrons once they have thermalized, making polyethylene inefficient away from the source.

In contrast to polyethylene, carbon does a poor job of peaking the flux near the source but has a very flat flux profile. The flux profile is flattened by the low-absorption cross section. Carbon also slows neutrons less efficiently than polyethylene, allowing neutrons to diffuse farther as they thermalize, flattening the flux profile.

The beryllium flux profile is shaped like the carbon flux profile but has higher values. The lower weight of beryllium thermalizes the neutrons better than carbon and the higher atom density shortens pathlengths, helping to increase the flux value. The slightly higher absorption cross section of beryllium eliminates neutrons slightly more than in carbon.[6]

Light water produces a flux profile very similar to polyethylene. Heavy water produces a shape much like that of carbon. The practical difficulties of designing around a liquid eliminate these materials from our current design.

A measure of the relative material performance can be estimated by the areas under the flux profiles. The areas represent the population of neutrons available to interact with the fuel rod in the scanner. Polyethylene has the largest area of the curves shown. When the flux between the source and through-hole (0 to 4 cm on the figure) is removed, the beryllium has a larger area. The polyethylene thermalizes more neutrons that are not able to irradiate the fuel rod making beryllium the best moderator for this application.

A high price must be paid for selecting a beryllium system. Beryllium costs several hundred dollars per kilogram in finished form. Beryllium can be precisely machined but machining is an expensive process because of health and safety concerns. These economic considerations demanded that the optimum design be compromised to greatly reduce the cost of the required beryllium.

Homogeneous beryllium outperforms other homogeneous moderators but it is not as efficient as an inhomogeneous moderator. The area under the beryllium flux profile can be increased by adding a small (≈ 0.5 –5 cm) layer of polyethylene near the source and the fuel rod to promote flux peaking. The amount of polyethylene surrounding the source and fuel rod depends on the local flux distribution in energy and the transport of neutrons in the system. The addition of too much polyethylene eliminates the flux-peaking gain by absorbing too many neutrons and creating a flux depression.

IV. DESIGN OPTIMIZATION

A right circular cylinder, 60 cm in diameter and 60 cm long, with a ^{252}Cf source emitting 1×10^8 neutrons per second was selected as the design-basis volume and source strength. Commercial rod scanners use source strengths of approximately 2×10^{10} neutrons per second. An optimum moderator design will reduce the neutron source strength requirements; in the long run, this should provide the most economic scanner.

The very compact dimensions of a ^{252}Cf source are an advantage in this application. The source can be placed very close to the fuel rod to take advantage of the large flux values very near the source. The source is also small enough to be treated as a point source computationally.

The first stage in the design was to benchmark the calculations to a prior calculation. The flux profiles shown in Fig. 1 were recalculated using the Monte Carlo neutron-photon (MCNP) code.⁷ The results agreed with previously published values given in Reference 5.

Design calculations were performed to maximize the neutron flux in the through-hole per source neutron. By tallying fissions along the entire surface of the through-hole, we calculated the integral of the flux along the through-hole. Tallying subregions along the through-hole allows a flux profile to be calculated. Neutron production from the γ -n reactions in the beryllium was neglected. The fission rate was not calculated until a full response model was developed for optimized designs.

The first models were spherical systems with a through-hole for the fuel rod. Two simple design rules became obvious. Optimum performance occurs with the minimum practical radius for the through-hole. Larger through-holes decrease the flux value at the surface of the through-hole, reducing the total number of fissions induced in a fuel rod. Another consideration is moving the neutron source as close to the through-hole as is practical to maximize the total flux.

The spherical geometry was largely controlled by the cylindrical through-hole, making the spherical model nearly as complex as a full cylindrical model. Almost all of the development work was done on cylindrical models. The neutron flux per source neutron along the through-hole is shown in Table I for different design features in cylindrical geometry.

The first designs modeled were right circular cylinders of solid polyethylene and solid beryllium. A stainless steel californium source holder and a through-hole were the basic features of the model. These first simple designs provide a good baseline for measuring performance improvements.

Table I. Performance of Rod Scanner Designs			
Design Option, Additional Features	Thermal Flux Along Through Hole Per Source Neutron, ϕ	Flux Ratio Compared to Solid Poly $\frac{\phi}{\phi_1}$	Flux Ratio Compared to Solid Be $\frac{\phi}{\phi_4}$
1. Solid Poly	2.61×10^{-3}	1.00	0.61
2. Solid Be	3.125×10^{-3}	1.20	0.73
3. Poly Around Source	4.00×10^{-3}	1.53	0.94
4. Poly Around Outside of Be	4.26×10^{-3}	1.63	1.00
5. Poly Tube Along Through Hole	4.37×10^{-3}	1.67	1.03
6. Two Poly Tubes Along Through Hole	4.71×10^{-3}	1.80	1.11
7. ^{252}Cf Source Poly Cylinder Offset	5.07×10^{-3}	1.94	1.19
8. Final Design	5.23×10^{-3}	2.00	1.23

The polyethylene cylinder around the ^{252}Cf source efficiently thermalizes the source neutrons and creates a flux trap near the through-hole. The polyethylene reflector returns many escaping thermal neutrons to the system. These two features produce more than half the improvement in the total flux value.

The design was further improved by adding a polyethylene tube around the through-hole. By changing the length and diameter of the tube, we found an optimum design. The tube was stopped well short of the axial beryllium-polyethylene interface because the neutrons far from the source are well thermalized. Two concentric tubes of polyethylene were also modeled. A relatively short, large-diameter tube near the neutron source and a much thinner tube running out toward the beryllium-polyethylene boundary were found to give the best result. More complicated polyethylene tube designs did not significantly improve the performance.

Moving the source polyethylene cylinder off-center created the most surprising result. The shift came from trying different polyethylene densities to promote streaming from the source to the through-hole.⁸ A reduction in the polyethylene density behind the source led to consistently better performance. Replacing the back half of the source polyethylene cylinder with beryllium continued to improve performance. Further optimization produced the offset cylinder geometry.

The performance is increased by the combined effects of polyethylene moderation and neutron multiplication in the beryllium. Reducing the polyethylene behind the source allows more high-energy neutrons to interact with the beryllium, multiplying neutrons in (n,2n) interactions. Reducing the polyethylene behind the source also allows more neutrons to return to the through-hole because of fast-neutron reflection in the beryllium. The offset polyethylene-cylinder geometry creates the best balance between multiplication and neutron thermalization. Table II lists the multiplication and total flux values for different geometries. Half the improvement of the offset geometry results from increases in the multiplication and half from better neutron transport to the through tube.

A drawing of the optimized design is shown in Fig. 2. The addition of a layer of lead around the entire moderator to act as a gamma-ray shield was also modeled. The lead provides some marginal neutron reflection by returning some of the neutrons that would otherwise escape.

Table II. Effect of Changing Neutron Multiplication*		
Geometry and Materials	Neutron Multiplication*	Total Through-Hole Flux per Source Neutron
Solid Poly Rod Scanner	1.00	3.00×10^{-3}
Optimum Rod Scanner with Poly Cylinder Centered on Source	1.10	5.00×10^{-3}
Optimum Rod Scanner with Poly Cylinder Offset from Source	1.13	5.23×10^{-3}
Solid Be Rod Scanner	1.17	4.53×10^{-3}
*The multiplication corresponds to the (n,2n) reaction in Be.

Fig. 2. Rod scanner moderation materials and geometry.

Changing the shape of the polyethylene between the source and the through-hole from a right cylinder to an ellipsoid provides a small measure of improvement. However, the required elliptical shapes provided too small a performance gain to justify the cost of machining the shapes.

The flux profile along the through-hole is sharply peaked at the center of the rod scanner near the ^{252}Cf as shown in Fig. 3. If the element being irradiated was larger in volume, a flatter-profile design would be more attractive.

Fig. 3. Axial neutron flux profile in fuel rod through-hole.

The sharply peaked neutron flux shows that the beryllium away from the source is much less important than beryllium near the source. Little performance is lost by replacing the outer boundaries of the beryllium with graphite. The effect of replacing the beryllium perimeter with carbon is shown in Fig. 4. The figure shows that approximately 11 cm of beryllium can be replaced by carbon at the axial ends of the scanner with only a 5% loss in performance. Approximately 9 cm of beryllium can be replaced by carbon at the outer radius of the rod scanner with only a 5% loss in performance. Making both these changes will reduce the total flux in the through hole by 12% while reducing the beryllium mass by 76% (190 kg to 45 kg). The very large savings in beryllium make the rod scanner an economic possibility.

Fig. 4. Total neutron flux ratio for moderators with different beryllium-carbon geometries.

The number of fissions produced per source neutron is shown in Fig. 5. The figure shows the nonlinearity of the thermal neutron interrogation response. The differential response decreases with increasing plutonium loading because the surface plutonium shields the interior plutonium. The plutonium wt% in thermal reactors does not typically exceed 5%. The $\approx 7 \times 10^{-5}$ fissions/source neutron at zero wt% plutonium comes from fissions in the natural uranium that makes up the fuel. Two other important effects are shown. The fast neutron interrogation response is very close to linear. The greater penetrability of the fast neutrons allows them to reach the inside of the fuel rod and interrogate all the fissile material. Fast neutron interrogation is particularly important for fast reactor fuel because of the high fissile fraction in the fuel. The fast neutron response is also much smaller than the thermal response because of the lower fast fission cross section. Thermal reactor fuel can be most efficiently measured using thermal neutrons.

The flux profiles in a prototype rod scanner could be verified using a fission chamber small enough to fit in the fuel rod through-hole. The flux values could be measured to near the accuracy of the strength of a known calibration source.

Fig. 5. Number of fissions induced per source neutron for increasing wt% of plutonium.

V. SHIELDING MATERIALS

The detectors used to measure the delayed gamma rays must be well shielded against both neutrons and gamma rays. The time delay between inducing a fission and measuring the delayed gamma rays needs to be minimized to reduce the decay of the delayed gamma rays. This requires a shield of minimum thickness.

The most significant shielding problem comes from fast neutrons that are escaping the rod scanner. The gamma rays from neutron captures also contribute heavily to the dose. A layered shield should be designed to stop both gamma rays and fast neutrons.

Good shielding for the combined gamma-ray and neutron flux is provided by layers of tungsten and borated polyethylene. Tungsten has a high inelastic-scattering cross section, which down-scatters high-energy neutrons to lower energies where the neutrons can be more easily thermalized and absorbed. Tungsten also stops gamma rays very effectively. Nickel and iron also down-scatter fast neutrons well but make relatively poor gamma-ray shields. Lead is a good gamma-ray shield but it is poor at down-scattering fast neutrons or absorbing thermal neutrons. The disadvantage of tungsten is that it costs more than lead. If tungsten was not used, layers of lead, iron, and borated polyethylene could be used.

Borated polyethylene thermalizes neutrons well and removes thermal neutrons. The boron produces lower-energy capture gamma rays than hydrogen, which reduces gamma-ray buildup and shielding requirements.

VI. SHIELDING DESIGN AND PERFORMANCE

The fast and thermal neutrons escaping from the lead shield on the face of the rod scanner were tallied using MCNP to judge the shielding effectiveness. A tally was also made on the number and spectra of the escaping gamma rays. The buildup of gamma rays from neutron capture is included in the model.

A composite layer of gamma-ray shielding made from tungsten and borated polyethylene is shown on the face of the moderator in Fig. 6.

The performance of a shielding layer for gamma rays is shown in Fig. 7. The gamma rays from neutron capture are approximately 90% of the escaping gamma rays. The composite-shielding layer allows just over twice as many gamma rays to escape as would a solid lead shield of equal thickness. The composite-shielding layer was designed mainly to stop neutrons, helping to account for the reduced performance against gamma rays. The performance of the shields with respect to neutrons is shown in Table III.

The composite-shielding layer reduces the number of thermal neutrons escaping the shield by a factor of more than 5000 when compared with the lead shield. Boron in the polyethylene easily absorbs the thermal neutrons. Fast neutrons are harder to eliminate because they must first be down-scattered to energies where hydrogen can thermalize the neutrons easily. Tungsten in the layered shield down-scatters the neutrons, which are then thermalized by the polyethylene and finally absorbed by the boron. The layered shield performs much better with fast neutrons than the lead does. The high expense and lower neutron performance make a solid tungsten shield a less attractive option than the composite shield. The reduction in neutron flux implies that a single set of shielding layers would be adequate if backed by a heavy-metal gamma-ray shield around the detectors.

VII. GAMMA-RAY DETECTOR SELECTION

Large NaI(Tl) and bismuth germanate (BGO) detectors were evaluated for counting the delayed gamma rays from the induced fissions. The primary reasons for selecting NaI(Tl) detectors for the current design are their robust nature and large available sizes. The detectors are capable of dealing well with physical shock and thermal changes. The detectors can also be bought with built-in gain stabilization.

Fig. 6. Composite shielding geometry for the NaI detectors used to count the induced gamma-ray activity.

Fig. 7. Escaping ²⁵²Cf gamma rays from rod scanner as a function of shielding material.

BGO, the most likely competitor to the NaI detector, is much more sensitive to temperature changes. The light output of the BGO depends strongly on temperature. Changing the temperature would change the detector resolution. The change in resolution would shift the number of useful counts measured even if no change in energy calibration occurred. Most other detector materials are too small, or too sensitive to shock, or both. The conservative choice is NaI(Tl) detectors with BGO detectors being a possible alternative. Detector properties are listed in Table IV.

Selecting the size of the detector for high efficiency is straightforward. Large detectors collect more of the energy of a gamma ray than do small detectors because their volume reduces the probability of partial energy escape. The choice of detectors is complicated, however, by a desire to have good spatial resolution. The spatial resolution of the detectors affects how many counts are

Table III. Neutrons Escaping From Shielding Layers		
Shield Type	Thermal Neutrons Escaping per Source Neutron	Fast Neutrons Escaping per Source Neutron
Layered Shield	3.09×10^{-9}	3.91×10^{-7}
Tungsten Shield	5.95×10^{-8}	7.54×10^{-7}
Lead Shield	1.76×10^{-5}	1.61×10^{-6}

Table IV. Scintillator Detector Properties[9,10]		
Detector Material	Advantages	Disadvantages
NaI	Rugged High Effective Light Output Thermally Stable Large Volume Crystals	Lower Stopping Power Low Neutron Exposure Limit
BGO	Good Stopping Power Large Volumes Possible Low Afterglow More Radiation Resistant than NaI	Thermally Unstable Low Effective Light Output
CsI	Good Stopping Power Very Rugged Easy to Fabricate Good Light Output More Radiation Resistant than NaI	Thermally Less Stable Slow Time Response
CdWO ₄	Good Stopping Power Good Light Output Thermally Stable	Brittle Small Volumes

clearly read from a particular fuel-rod zone for a rod with multiple zones. If the spatial resolution is poor, the counts from the small zones tend to merge with the larger adjacent sections.

Two NaI detectors (127 mm x 127 mm) with through-holes were modeled in these calculations. They are reasonably efficient and provide a clear signal from the different fuel zones. The very high count rate from x-rays and gamma rays with energies less than 150 keV requires a 1-mm tungsten shield within the through-hole to lower the total count rate to acceptable levels. Most of the low-energy counts are 60-keV gamma rays from americium.

A drawing of the rod scanner system without the biological shield is shown Fig. 8. The biological shield is a 1-in.-thick layer of borated polyethylene to shield neutrons without creating a significant buildup of gamma rays. With the borated polyethylene shield in place, the surface dose rate from neutrons and gamma rays is less than 0.4 mrem/h.

VIII. ROD SCANNER SYSTEM MODEL

The complete system, including the moderator, shield, and detectors, was modeled so that the total performance could be estimated. We calculated the number of fissions per source neutron ratio for different weight percentages of plutonium, allowing a source normalization. The axial flux profile allows the number of fissions induced at a point along the rod scanner axis to be calculated. The decay behavior of the gamma rays and the energy spectra is known from Reference 2. The

Fig. 8. Schematic diagram of final rod scanner with moderation materials, shielding layers, and NaI detectors.

response of the NaI detectors to the delayed gamma rays was simulated using a source in the detector at different positions along the through-hole. A distributed source within a small length of the fuel rod was modeled to account for self-shielding. The response to different gamma-ray energy groups is required to correctly account for down-scatter of the delayed gamma rays from the neutron interrogation and the presence of the natural-decay gamma rays from the plutonium. These gamma-ray emissions were used to produce a time-dependent model that could be used to estimate the number of detected counts above a 1-MeV energy threshold.

In the time-dependent model, we break the neutron moderator, detectors, and the fuel rod into discrete spatial elements that are small compared to both the flux changes in the interrogator and the gamma-ray detector's response profile. The modeled rod section is positioned in the first spatial element in the neutron moderator where the flux is assumed constant over the element. The program calculates how many fissions are created in the first time interval while the rod element is within a moderator element. The activated fuel-rod element is then moved to the detectors where the induced signal is calculated for each of the elements within the detector pair. The recorded signal is the product of the detector response function and the activity of the fuel-rod element. The decay of the delayed gamma rays is taken into account. The known rod speed allows the detector responses to be calculated as a function of time and rod speed.

Once the first activation step has been completed, the fuel-rod element is advanced one position in the rod scanner and another activation and detector transport step is completed. After the fuel rod element has passed through the detector, the next fuel-rod element is started through the rod scanner. In this way, the contribution from each fission in each fuel-rod element is simulated

for each position in the rod scanner and gamma-ray detectors. The rod speed, plutonium loading, time step, fuel-rod element size, and rod scanner dimensions can all be changed to study the effect on the signal and background counts.

A trace of the calculated counts/second recorded in the detectors is shown in Fig. 9 for a 1×10^6 n/s- ^{252}Cf source and a MOX fuel rod with 2.1 wt% plutonium and natural uranium. The middle line in Fig. 9 is the signal from the natural-plutonium-decay gamma rays and down-scattered delayed gamma rays with detected energies below 1 MeV. Most of the plutonium decay gamma rays and all of the plutonium x-rays occur in this energy region. The low-energy decay gamma rays make up the bulk of the total signal. The lowest line is from the detected 1.0–10-MeV delayed gamma rays, which correspond to the signal used to measure the fissile content. The top line is the total signal and includes all the detected gamma and x-rays. The 1-MeV energy threshold is used to separate the induced gamma rays from the natural-plutonium-decay gamma rays in the MOX fuel. Without the clear energy discrimination between decay and delayed gamma rays, the signal-to-noise ratio would make the problem much more difficult.

The step down in the count rate at 80 seconds is created by three off-specification fuel pellets of 1.8 wt% plutonium, a -0.3 wt% change. The lowered number of fissions reduces the signal in the detectors. The change in the signal is the largest in the total counts because the reduced plutonium load reduces both the decay and delayed gamma rays. The width of the response is controlled by the spatial resolution of the detectors and the size of the off-specification pellet group. The rise in the count rate at 170 seconds is created by three pellets of 2.4 wt% plutonium, a +0.3 wt% change, in the fuel rod.

The total counts from a rod as a function of rod speed are shown in Fig. 10. This graph shows the large number of counts produced using a source of 1×10^6 n/s and a 1.8-wt% plutonium and natural uranium fuel rod. The total number of counts recorded in a rod is a strong function of the rod speed through the rod scanner. The most efficient speed is determined by the decay

Fig. 9. Gamma-ray counting rate as a function of gamma energy for a rod transfer speed of 1 m/min.

Fig. 10. Integral delayed gamma-ray counts recorded in a fuel-rod zone for as a function of fuel-rod speed.

behavior of the induced gamma rays. The rod scanner design changes the magnitude of the curve but not its shape. The most efficient rod speed is desirable for the higher count rate and the smaller errors introduced by changing rod speeds. The final speed used will be partially determined by the throughput requirements of the fabrication facility.

Typical thermal reactor fuel pins have a higher enrichment at the ends of the rod than in the center to flatten the reactor power distribution. The rod scanner detects the changing plutonium loading in the same way the off-specification pellets were detected. The signal above 1 MeV in the detectors is proportional to the fissile loading.

A total of 40 000 integral counts above 1 MeV is required in each of the rod enrichment zones to achieve 0.5% counting statistics. A 1×10^6 -n/s- ^{252}Cf source (0.4 μg or 0.21 mCi) provides nearly 1 000 000 counts for the lowest fissile fraction expected in the fuel rods. This allows for 5 years (2 half-lives) with an excess of counts when measuring a fuel rod end zone only 25% of the rod length.

IX. CONCLUSIONS FOR ACTIVE NEUTRON SYSTEM

We used MCNP calculations to design an optimum neutron moderator for thermal neutron interrogation systems. Moderator materials including CH_2 , beryllium, and graphite were studied to determine the optimum size and position of the composite moderator. The moderator was designed to provide the maximum number of induced fissions per source neutron along an axial through-hole provided for the fuel rod. The final design increased the neutron interrogation efficiency by a factor of 2 when compared with a simple polyethylene design.

New moderator design approaches included the following:

- a. Placing polyethylene sleeves around the fuel rod through-hole for a gain of 17% when compared to a polyethylene-reflected beryllium moderator. Larger benefits were obtained by tapering the sleeves to be thinner for rod positions further from the ^{252}Cf , as illustrated in Fig. 2.

- b. Offsetting the ^{252}Cf source from the internal polyethylene moderator to the side away from the fuel rod channel as shown in Fig. 3.
- c. Replacing the outer 70% of the beryllium moderator with graphite to achieve a large cost savings with only a 12% reduction in the interrogation efficiency.

After the moderator was designed, we extended the calculations to include the delayed-gamma-ray detectors and the shielding in the direction of the ^{252}Cf source. To reduce the decay background from the plutonium in the MOX fuel rods, only high-energy (>1 MeV) gamma rays are counted. Large NaI through-hole detectors were selected to provide a high efficiency for the energetic gamma rays. The combination of high-efficiency gamma-ray detectors and an efficient moderator design made possible a significant reduction in the required ^{252}Cf source strength. This system, which measures fuel-rod loading zones, can use a source four orders of magnitude smaller than those currently used in commercial systems that resolve individual pellets. In the present design of the MOX rod scanner, the individual pellets would be resolved in the passive gamma-ray mode.

PHASE II: DESIGN OF PASSIVE NEUTRON SCANNER*

I. INTRODUCTION

MOX fuel rods can be measured using intrinsic passive neutron and gamma-ray emissions. The plutonium mass can be verified using spontaneous fission neutrons from the effective mass of ^{240}Pu where

$$^{240}\text{Pu}_{\text{Effective}} = 2.52 \cdot ^{238}\text{Pu} + ^{240}\text{Pu} + 1.68 \cdot ^{242}\text{Pu} \quad (1)$$

The isotopics of the plutonium sample must be determined to calculate the total plutonium mass from the $^{240}\text{Pu}_{\text{Effective}}$ mass. The plutonium isotopic ratios are measured with a high-resolution gamma-ray spectroscopy (HRGS) system using a high-purity germanium (HPGe) detector. Generally, longer counting times are required for the isotopic measurements.

The plutonium fissile content and pellet-to-pellet uniformity can be determined from the passive gamma-ray emissions primarily from ^{239}Pu , ^{241}Pu , and ^{241}Am . Low-resolution gamma-ray detectors can be used to detect the gamma-rays. An active gamma-ray transmission system can be used to check pellet density and observe gaps between pellets.

II. PASSIVE NEUTRON COINCIDENT COUNTING

Even-numbered plutonium isotopes spontaneously fission, emitting neutrons. The fuel rod is nearly transparent to fission neutrons, which allows all the plutonium within the rod to contribute to the neutron signal.

By counting coincident neutrons we can measure time-correlated fission neutrons by analyzing how the neutrons are detected as a function of time. During fission, several neutrons are emitted ($\bar{\nu} = 2.156$ for ^{240}Pu), thermalized, and detected within a short time. Coincident counts occur when multiple neutrons are detected during a time interval within the lifetime of the fission neutrons. Accidental coincident counts are produced when background, (α, n) , and fission neutrons are detected within the lifetime of the fission neutrons. The accidental coincident counts are not correlated to a fission event, and they occur at a constant rate regardless of when the correlation in time is made. The random coincident-count rate is subtracted from the total coincident-count rate, leaving the true coincident-count rate. The mass of plutonium is proportional to the true coincident-count rate. A correction based on the total and coincident count rates can be applied to account for neutron multiplication.¹¹

Coincidence counting allows samples to be analyzed without detailed information on the sample density or geometry. Plutonium samples of less than a gram up to several hundred grams, in many chemical forms, have been successfully measured.

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III. DESIGN GOALS

High efficiency, to minimize counting times, was the primary design goal for the rod scanner. Coincidence counters are especially sensitive to the counter efficiency because the probability of detecting a single neutron is proportional to the efficiency while the probability of detecting two neutrons from the same source is proportional to the efficiency squared.

The number of ^3He tubes was limited in the design to reduce the cost of the tubes and the required electronics. The use of expensive materials like beryllium was also eliminated for economic reasons.

IV. PREVIOUS WORK

The inventory sample III (INVS III) counter is the coincidence counter closest in design to the present rod scanner. [12] The INVS III is 42.4% efficient and was optimized to measure small plutonium samples. The high efficiency was achieved by using 18 ^3He tubes in 2 concentric rings in a polyethylene moderator.

The major differences between the INVS III and the passive rod scanner are imposed by the fuel rod size, which requires a pass-through design. The radius of the fuel rod is also smaller than the INVS III assay chamber radii. The rod scanner will measure MOX fuel rods that have several times the plutonium mass typically measured by the INVS III counter. These changes increase the counting rate, which reduces the required counting time.

V. DESIGN OPTIMIZATION

A. ^3He Tube Placement

The efficiency and neutron die-away times were calculated using MCNP. The code was benchmarked using data from the INVS III neutron counter.

The polyethylene that is the primary moderator of the rod scanner provides a large thermal-neutron-flux peak near the fuel rod where the ^3He tubes are placed. The use of an inhomogeneous polyethylene-beryllium moderator would produce a more efficient counter but at an excessive cost. The neutron absorption by polyethylene is not a critical problem in this design because of the large number of highly absorbing ^3He tubes and the small size of the moderator system. Graphite or teflon is used to reflect fast neutrons and to improve the neutron economy away from the fuel rod where the neutrons have been well thermalized. A neutron shield of polyethylene is required around the detector array to reduce background neutrons.

Eighteen ^3He tubes were selected to obtain good efficiency. Additional ^3He tubes would be difficult to justify because each additional tube contributes a smaller fraction of the total efficiency. The ^3He tubes are 15 in. long, 1 in. in diameter, and have a 6-atm fill pressure. Higher fill pressures are not useful because the incremental improvement in performance decreases rapidly. Figure 11 shows the effect of ^3He pressure on detector efficiency. A short tube length was selected to improve the spatial resolution of the rod scanner. Similar detectors have been successfully used in the INVS III counter.

The placement of the ^3He tubes required most of the design effort. The efficiency of the neutron counter as a function of ^3He tube placement is complicated by the interaction (neutron

Fig. 11. Efficiency for 18 ^3He tubes as a function of ^3He pressure.

shadowing) of the ^3He tubes. In this design, sets of ^3He tubes are arranged in three concentric rings about the fuel rod as shown in Fig. 12. More than one ring of tubes is used to reduce detector interactions. The compromise between the number of ^3He tubes per ring and the distance between rings and the fuel rod is affected by two conflicting factors. The ^3He tubes should be placed as close to the fuel rod as possible so the detectors can operate in the maximum neutron flux. The detectors should also be placed as far apart from each other as possible to minimize the flux depression of one tube affecting the other tubes. The first requirement pushes the detectors together while the second drives them apart.

Fig. 12. Helium-3 concentric ring geometries.

The total efficiency, with a centered ^{252}Cf neutron source for separate rings of tubes at increasing distances from the center of the fuel rod, is shown in Fig. 13. Each additional tube produces a smaller improvement in total efficiency. The optimum distance from the fuel-rod centerline is also moved farther out as more tubes are added. Both trends are related to the flux in which an individual ^3He tube operates.

A single tube would have the maximum efficiency per tube at the thermal neutron flux peak. Additional tubes must be separated for maximum efficiency, which increases the required ring radius. Both nine- and eight-tube rings have approximately the same maximum efficiency. More than nine tubes in a single concentric ring reduces the total efficiency. Figure 14 shows the efficiency per tube for different numbers of tubes in concentric rings at various distances from the fuel rod.

Fig. 13. Total efficiency for a single ring of tubes as a function of distance from fuel rod centerline.

The effect of a pair of six-tube rings on each other is shown in Fig. 15. The inner ring starts at 3 cm from the fuel rod centerline. At 3 cm, the ring is too close to the fuel rod and the neutrons are not well thermalized, which reduces efficiency. As the inner ring is pulled away from the fuel rod, it gains efficiency by having less neutron shadowing. Once the inner ring is past 5 cm from the fuel rod centerline, the efficiency drops with the neutron flux. The efficiency of the outer ring drops as the inner ring is moved out because the inner ring is absorbing more neutrons that would have reached the outer ring.

Figure 16 shows the addition of a middle ring to a system with an inner ring at 3 cm and an outer ring at 10 cm. As a middle ring moves from the closest position possible to the farthest position, the total efficiency goes through a maximum and then decreases.

The most efficient tube geometry was found by optimizing the efficiency of the inner ring of tubes. Then a middle ring was drawn in and both ring positions optimized.

Fig. 14. Efficiency per tube as a function of distance from fuel rod centerline for a single ring.

Fig. 15. Ring efficiency as the inner ring is expanded in radius to the outer ring; outer ring at 9 cm.

Fig. 16. Ring efficiency for middle ring expanded out to the outer ring, inner ring at 3 cm, outer ring at 10 cm.

The inner ring position needed to be slightly closer to the fuel rod to allow the middle ring to be moved closer to obtain a higher thermal-neutron flux. Finally, the third tube ring was brought in and all the rings optimized. Increasing the total number of tubes would require a change in the ring spacing and the number of tubes per ring. The final tube arrangement is shown in Fig. 17.

B. Neutron Reflector

The final feature added to the neutron counter was a graphite or Teflon reflector. An annulus of graphite or Teflon several centimeters thick was placed around the ^3He tubes. The inner and outer radii of the annulus were then varied to achieve an optimal performance. The graphite slightly reduces the die-away time of the neutron counter.

Fig. 17. Optimal ring spacing diagram for polyethylene moderator.

The neutron counter is shielded by 10 cm of polyethylene that thermalizes and absorbs most of the background neutrons that reach the neutron counter. The polyethylene also contributes to the reflection of source neutrons.

C. Performance Parameters

Figure 18 shows the full passive neutron counter with three rings of ^3He tubes, the graphite reflector, and the polyethylene shield. Table V gives the performance achieved with different geometries and materials.

Fig. 18. Completed rod scanner design.

Table V. Performance for Various Combinations of Reflector and Shield				
System Description	Reflector	Shield	^{252}Cf Efficiency	^{240}Pu Efficiency
Solid Polyethylene	Poly	-	48.8	50.7
Graphite Reflected	Graphite	-	47.30	48.9
Graphite + Polyethylene Reflector	Graphite Poly	-	51.3	53.0
Graphite + Polyethylene Shield	Graphite Poly	Poly	51.7	53.4
Teflon + Polyethylene Reflector	Teflon Poly	-	50.4	52.7
Teflon Reflector + Polyethylene Shield	Teflon Poly	Poly	51.8	52.7

The final total efficiency is 51.7% with a ^{252}Cf neutron source at the center of the rod scanner. The first, second, and third rings have efficiencies of 24.8%, 15.0%, and 11.9%, respectively. The addition of another ring of tubes would increase the total efficiency by only 4%–5% without a significant design change.

The neutron die-away time, t_d , within the rod scanner was calculated by fitting the neutron population as a function of time to the formula

$$N(t) = N(0)e^{-t/t_d}, \quad (2)$$

where $N(t)$ is the number of neutrons at time t and $N(0)$ is the initial number of neutrons. The die-away time of 41 μs is a typical value for polyethylene moderator with a high density of ^3He tubes and compares well with the 57- μs die-away time of the INVS III.

Using an optimized Teflon reflector in place of the graphite reflector would reduce the total efficiency by 1.5%. Teflon will produce a cost savings of 50% in the price of the reflector when compared to graphite.

The efficiency profile as a function of the distance from the axial center of the rod scanner is shown in Fig. 19. The full width at half maximum (FWHM) is approximately 40 cm. This is approximately half the length of the shortest anticipated fuel rod zone. The spatial resolution could be improved by reducing the length of the moderator materials. The efficiency-squared curve is a measure of the coincident-detection efficiency. The coincident-detection efficiency is about half of the total efficiency and has a FWHM of 30 cm.

VI. COUNTING PERFORMANCE FOR MOX FUEL

Given the calculated efficiency and the die-away time, we can estimate the neutron counter's performance. The first stage in the calculation is to compute the efficiency for detecting spontaneous-fission neutrons from MOX fuel rods. The ^{240}Pu spontaneous-fission neutrons have a

Fig. 19. Efficiency profile along axis of rod scanner.

lower average neutron energy than the ^{252}Cf , which increases the relative efficiency from 50.7% to 53.2% for a source of neutrons at the center of the neutron counter. The average detection efficiency over the entire volume of the rod scanner is ~40% with the fuel rod in place. The average efficiency is lower than the center-source efficiency because of losses toward the ends of the rod scanner.

Some multiplication occurs when thermal neutrons return to the fuel rod and induce fission and when fast-fission neutrons directly induce fission. The multiplication as a function of plutonium wt% is shown in Fig. 20. One wt% plutonium is equal to approximately 4.4 g of plutonium and 1.1 g of ^{240}Pu inside the detector. The addition of a 0.4-mm cadmium through-tube liner greatly reduces the multiplication and the slope of the multiplication curve as shown in Fig. 20.

The real counts from the rod scanner as a function of plutonium wt% are shown in Fig. 21. With the cadmium liner, the detector is much less sensitive to errors in the (α, n) rate estimate and the response is much more linear. The reals rate without the cadmium begins to diverge from linear because it depends on the multiplication squared.

The real counts rate can be corrected for the effects of multiplication to produce a linear response to plutonium. The correction of the reals rate per gram of $^{240}\text{Pu}_{\text{Effective}}$ to a zero multiplication is shown in Fig. 22. The required correction is much smaller for the cadmium-lined case. The accuracy of the correction for the oxide depends on how well the (α, n) rate is known. In all cases the correction is less sensitive when the multiplication is reduced, making the cadmium liner very useful.

The cadmium reduces the average efficiency from 40% to 37.2%. The drop in total efficiency is small because most of the neutrons absorbed by the cadmium would have been absorbed in the fuel rod.

Fig. 20. Neutron multiplication as a function of plutonium wt% in fuel rod scanner.

Fig. 21. Detected real neutrons as a function of plutonium wt% for oxide and metal alpha, n rate.

Fig. 22. The coincidence rate per gram of ^{240}Pu as a function of the plutonium wt% in the rod.

With the calculated die-away time of 41 μ s, the number of coincident, accidental, and total counts recorded using a shift register can be estimated. The total count rate is given by

$$T = \epsilon SM \quad , \quad (3)$$

where T is the total count rate, ϵ is the efficiency, S is the source strength in neutrons per unit time, and M is the multiplication.

The accidental coincidence count rate is given by

$$A = GT^2 \quad , \quad (4)$$

where A is the accidental coincident count rate and G is the shift register gate width in seconds.

When a 2 wt% plutonium and natural uranium MOX fuel rod is within the rod scanner, it produces a total count rate of 3900 counts per second, an accidental coincident count rate of 730 counts per second, and a real coincident count rate of 470 counts per second. The counting statistics as a function of measurement time are shown in Fig. 23. Only 30 s is required to reach a 2% error in the counting statistics and 1% counting statistic is reached in 95 s. A 75-cm fuel rod section will be within the rod scanner for 45 s at 1m/min, so the 2% statistical accuracy would be achieved for the zone.

VII. ROD SCANNER DESIGN CONCLUSIONS

The rod scanner can assay individual fuel-rod zones on line in a fabrication facility. A precision of 1% can be obtained for a 2 wt% plutonium MOX rod in 100 s. The high neutron-detection efficiency is the primary reason for the short measurement time.

Fig. 23. Relative error of assay for increasing count time for 2 wt% Pu MOX (24% ^{240}Pu).

The neutron counter is moderated by polyethylene to promote strong flux peaking near the fuel rod. Optimum placement of the ^3He tubes gives a peak efficiency of 53.4% and an average efficiency of 37.2% for a MOX fuel rod. A graphite reflector improves total performance. A 0.4-mm cadmium liner is used in the fuel-rod through-hole to reduce the effects of multiplication.

PHASE III: GAMMA-RAY MEASUREMENT SYSTEMS

I. INTRODUCTION

The passive neutron counter depends on accurate isotopic data to calculate the total mass of plutonium from the measured $^{240}\text{Pu}_{\text{effective}}$ mass. In a safeguards application, the plutonium isotopics are measured to ensure that the declared operator values are correct. A second advantage to independent isotopic measurements is the fuel-cycle transparency derived from the measurement. The burnup, age since plutonium separation, and basic reactor data can all be inferred from the plutonium isotopics.

Separate isotopic and neutron NDA systems would require the fuel-rod identification and measurement position to be verified in two instruments. A single, combined gamma-neutron assay system requires only one verification step and ensures the same material is measured while eliminating handling steps within the facility.

The previous passive neutron rod scanner design was modified to include an high-resolution, gamma-ray (HRGS) isotopic measurement system. The HRGS displaces some of the rod scanner's ^3He tubes, which reduces the neutron detection efficiency. The HRGS efficiency is maximized, at the expense of the neutron efficiency, to minimize the much longer isotopic measurement time.

II. ISOTOPIC CALCULATIONS

The ratio of the number of gamma rays detected at particular energies from plutonium isotopes can be correlated to the atom ratio of plutonium isotopes. A HRGS is required to adequately resolve the closely spaced gamma rays of plutonium. MOX fuel typically incorporates several elements with multiple detectable isotopes. Plutonium samples always have ^{238}Pu , ^{239}Pu , ^{240}Pu , ^{241}Pu , and ^{242}Pu . The freshly separated plutonium has no ^{241}Am or ^{237}U but both isotopes begin to buildup immediately as ^{241}Pu decays. The uranium used in the MOX has ^{234}U , ^{235}U , and ^{238}U . The total isotopics are also complicated by neptunium, additional isotopes of uranium, and the presence of some fission products depending on the reprocessing methods. High-burnup recycle plutonium usually has all of these isotopes, creating fuel with very complex isotopics.

The measurement of elemental isotopics is based on the equation

$$C(E_j^i) = \lambda^i N^i B_j^i \epsilon(E) \quad , \quad (5)$$

where $C(E_j^i)$ is the total peak counts detected at energy E for gamma ray j from isotope i , λ^i is the decay constant of isotope i , N^i is the number of atoms of isotope i , B_j^i is the branching ratio for isotope i and gamma ray j , and $\epsilon(E)$ is the efficiency with which gamma rays at energy E are detected. The branching ratio is the number of times gamma ray j is emitted by isotope i per decay. Several well-defined peaks within the detected spectra are fitted to get the efficiency curve and the peak-shape parameters. The peaks and background in the detected gamma-ray spectra are then fitted and the peak areas calculated to get the ratios between the number of atoms N^i and N^i . The

codes that perform the analysis are typically large and include many correlations to achieve accurate results. Measurements below 1% error are possible for the ^{240}Pu effective mass for long counting times.

III. DESIGN GOALS

The primary goal of the isotopic measurement system was to achieve the maximum efficiency to avoid excessive gamma-ray count times. Creating the minimum disruption to the neutron detector system is also important. Figure 24 shows a diagram of the combined neutron-gamma detector system. The gamma-ray system is inserted into the neutron-rod scanner and can see the fuel rod through a window in the polyethylene. The removal of polyethylene and repositioning of ^3He tubes reduces the neutron detection efficiency.

Fig. 24. Diagram of combined gamma-ray neutron rod scanner.

The highest emission rate, combined with reasonable penetrability of the gamma rays, occurs in the 100-keV region. In addition, all the measurable plutonium isotopes have gamma rays near 100 keV, making it the preferred energy region to analyze when few absorbers lie between the detector and sample. A window cut in the polyethylene allows the detector to see the fuel with a minimum of attenuation. Maintaining the neutron detector efficiency implies that a minimum of polyethylene should be removed and that the ^3He tube array should be displaced as little as possible. The thin aluminum walls of a single ^3He detector between the fuel rod and gamma-ray detector do not significantly affect the gamma-ray detector performance. A thin cadmium liner (0.4 mm) in the fuel rod through-hole is required to reduce the number of 60-keV ^{241}Am gamma rays detected. The cadmium gamma-ray filter is also used to minimize the rod scanner multiplication.

To maximize the gamma-ray detection efficiency and achieve a high resolution, a large 32-mm-diameter, 13-mm-thick HPGe planar detector is used. Most HPGe crystals have identical detector cans, which means no neutron performance is lost with a maximum-sized crystal.

The analysis of the 100-keV MOX gamma-ray spectra is a specialty of the analysis code MGA. The MGA code is optimized to calculate complex MOX fuel isotopics using HRGS detectors.

Two primary design values were studied: the position of the gamma-ray detector and the shape of the polyethylene window. MCNP calculations were performed to calculate the efficiency of both the neutron and gamma-ray detectors. Experiments were performed on a detector system mock-up to measure the counting rates expected from the gamma-ray system and the required measurement time for MGA analysis.

IV. RESULTS

The gamma-ray system was approximated in laboratory experiments. A drawing of the experimental setup is shown in Fig. 25. The source was a 0.45-g sample of high-burnup plutonium (24% ^{240}Pu) contained within a steel canister that approximated a small length of fuel rod. The MGA analysis results are shown in Tables VI and VII.

Fig. 25. Experimental setup to determine counting rates.

Table VI. True and MGA-Analyzed Plutonium Isotopics (Case 1, 0.45 g Pu, 30 min Count Time)				
	Isotopic Fraction	MGA Isotopic Fraction	Estimated MGA Error %	True Error %
^{238}Pu	0.895	0.8834	1.62	1.3
^{239}Pu	67.62	67.73	0.77	0.16
^{240}Pu	24.23	24.20	1.80	0.12
^{241}Pu	3.670	3.637	1.45	0.90
^{242}Pu	3.582	3.582	Known	-
^{241}Am	5.213	5.494	1.42	5.1

TABLE VII. True and MGA-Analyzed Plutonium Isotopes (Case 2, 0.23 gm Pu, 30 min Count Time)				
	Isotopic Fraction	MGA Isotopic Fraction	Estimated MGA Error %	True Error %
^{238}Pu	0.2283	0.2530	1.46	9.8
^{239}Pu	78.10	78.33	0.42	0.29
^{240}Pu	18.94	18.65	1.42	1.5
^{241}Pu	1.476	1.457	1.17	1.3
^{242}Pu	1.246	1.264	Known	-
^{241}Am	4.153	4.199	10.75%	1.1

The detector efficiency as a function of the width of the axial void-window at the fuel rod is shown in Fig. 26 for a detector 8 cm from the fuel rod. The void window can be relatively wide and can contribute significant additional counts. As the detector is brought toward the rod, the solid angle effect becomes larger.

To achieve 2.5% error in the estimate of the $^{240}\text{Pu}_{\text{Effective}}$ mass requires approximately 3000 s using the experimental system. The source used in the experiments closely approximates a 2-cm length of MOX fuel rod. The efficiency of a rod-scanner geometry that approximates the experimental setup was approximated using MCNP. Further calculations were then performed to account for the larger detector and the changing geometry. A graph of the gamma-ray and neutron detector efficiency as a function of the distance from the gamma-ray detector to the fuel rod is shown in Fig. 27. The gamma-ray efficiency was normalized to the experimental setup while the neutron efficiency was normalized to the passive neutron counter without the gamma-ray system. The neutron detection efficiency was not significantly affected by the introduction of the gamma-ray system until ^3He tubes were displaced by the HRGS detector. At the closest point, three ^3He tubes, two from the inner ring and one from the middle ring, have been removed. The neutron

Fig. 26. Detector efficiency as a function of void window width, normalized to 2-cm window.

Fig. 27. Relative efficiencies for gamma-ray and neutron detectors as a function of gamma-ray detector position.

efficiency at 5 cm rises because two inner-ring tubes can be replaced. The final tube is replaced at 7 cm but contributes little to the total performance because it is far from the fuel rod, located optimally, and is in the center of the void window.

The assay time for 2 wt% plutonium MOX was reduced from 3000 s to 600 s by reducing the cadmium filter thickness from 0.6 mm to 0.4 mm, using a higher-efficiency detector and moving the detector very close to the fuel rod (2 cm). The efficiency required for a 600-s assay can be achieved at the 2-cm position. This position significantly degrades the neutron counting performance because ^3He tubes have to be removed. Only a small volume of polyethylene needs to be removed to produce an effective void window when the detector is this close to the fuel rod. The neutron system performance has been reduced by 25% compared to the system without the HRGS system. This will add to the required neutron assay time.

The error as a function of time for the gamma-ray and neutron system is shown in Fig. 28. The relative error for the $^{240}\text{Pu}_{\text{Effective}}$ neutron assay is shown in detail in Fig. 29. The optimized

Fig. 28. Relative error as a function of count time for gamma-ray and neutron detector systems measuring 2-wt% plutonium MOX.

neutron system and the combined systems are shown. After adding the HRGS system, we increased the 2% relative error assay time from 30 s to 40 s while the 1% relative error assay time goes from 95 s to 135 s.

Fig. 29. Effect of reduced neutron detection efficiency on required measurement time.

V. CONCLUSIONS

The combined gamma-ray neutron rod scanner system will produce accurate measurements of the plutonium mass within fuel rods. Combining both gamma-ray and neutron detector systems into one system greatly simplifies verification and fuel-rod handling. The neutron detection efficiency was reduced by 13% absolute to provide an isotopic assay time of slightly less than 10 minutes. The neutron counting time is still one-fourth of the gamma-ray counting time. Reductions in isotopic assay time would require larger or multiple detectors. The isotopic count time of less than 10 minutes for a typical MOX rod should allow many fuel rods to be sampled.

VI. LOW-RESOLUTION GAMMA-RAY DETECTOR SYSTEMS

The low-resolution gamma-ray detector systems measure the conformity of the fuel rod to the design specifications, ensure that the fuel rods are being loaded as declared for safeguard purposes, and help make plant operations transparent. The operator can also gain valuable quality control information on the completed fuel rods. Two types of measurements are made: passive gamma rays from the plutonium, and the transmission of active gamma rays to measure the fuel pellet density and to detect any gaps between pellets.

Passive gamma-ray measurements verify the correct plutonium loading of each fuel pellet and identify individual out-of-specification fuel pellets by detecting gamma rays emitted by the plutonium and its daughters. The number of plutonium gamma rays detected is proportional to the

plutonium concentration in the fuel. The integrated fuel pellet data provide the plutonium content of the completed fuel rod. Passive plutonium gamma-ray measurements can be made over several gamma-ray energy regions. Two energy regions are of particular interest when measuring reactor-grade plutonium. The first energy range is centered on the strong source of ^{241}Am gamma rays at 60 keV. The second energy range is above 150 keV where multiple gamma rays are emitted directly from the plutonium. In particular, the 208-keV gamma ray from ^{241}Pu is a strong source in reactor-grade plutonium. The x-ray region around 100 keV is difficult to analyze because both uranium and plutonium contribute x-rays too close in energy to resolve with a NaI detector.

An active gamma-ray transmission densitometer can be used to measure fuel-pellet density and to look for pellet-pellet gaps in the fuel-pellet stack. The densitometers are available in commercial systems that include a gamma-ray source like ^{137}Cs at 660 keV with energy above the plutonium gamma rays.

VII. PASSIVE GAMMA-RAY COUNTING

The plutonium fuel loading pattern obtained from passive gamma-ray measurements verifies the declared operator values. Knowing that the HRGS measurement was performed on a representative section of the fuel rod is also useful in the assay. The measured americium content indicates which batches of plutonium are being manufactured into pellets, thereby allowing a correlation with other measurements at the fabrication facility. Additionally, the operator can verify that the pellet stack has been correctly assembled and that fuel pellets are not misplaced in the fuel rod, thereby degrading the fuel performance.

VIII. DESIGN GOALS

The primary design goals for the passive gamma-ray counting systems are getting a high detector efficiency for both plutonium and americium gamma rays and optimized spatial resolution for measuring fuel pellets. The significant energy differences between the americium and plutonium gamma rays make the gamma-ray detector design complex. The low-energy, 60-keV gamma rays from americium are easier to collimate and detect than the higher-energy 150–400-keV gamma rays from plutonium. Americium is measured when high spatial resolution or the plutonium batch information is required.

The higher penetrability of the plutonium gamma rays is helpful in two ways. First, more gamma rays reach the detector, which allows a much higher measurement efficiency per source gamma ray when compared to americium gamma rays. Second, plutonium gamma rays from inside the fuel pellet contribute to the assay, improving the measurement sampling.

The number of americium gamma rays measured is not directly correlated with the plutonium content in the fuel. The americium-to-plutonium ratio in a fuel sample depends on several factors including the time since the plutonium was separated, the burnup, and the reactor type. Rather than correlating the americium to the plutonium content, the plutonium is directly measured.

Several design options are available for the passive gamma-ray system. The simplest and least expensive design would use a single high-efficiency through-hole detector with an internal collimating sleeve. Higher-energy plutonium gamma rays would pass through the relatively thin collimator and then be detected with good efficiency and good spatial resolution. The low-energy americium gamma rays would reach the detector only through the collimator opening because of their low energy and penetrability. The americium signal could potentially have a very good spatial

resolution with a narrow collimator. The mixed signal must be analyzed over separate energy regions representing ^{241}Am and plutonium.

A problem may exist because of the potential for very high count rates in a single detector measurement. The unattenuated americium count rate from a collimator approximately the size of a fuel pellet can reach 100 kHz for 4 wt% Pu-MOX. The plutonium in the MOX would contribute additional counts to the system at a frequency greater than 100 kHz if left unshielded. Reducing the count rates to reasonable levels will require significant attenuating filters, narrower americium collimation, or a thicker shield.

A second design uses two separate NaI detectors optimized independently for each energy region. With separate detectors, the count rates in each detector can be significantly lower than those in a single detector and still be usable. The additional cost of the detectors and the electronic hardware is justified to make the system more adaptable.

Americium gamma rays would be measured with a small detector that allows for optimized spatial collimation and a high americium count rate. The total count rate would be controlled with absorbing filters of cadmium or tin. Higher-energy gamma rays from the plutonium would be detected at a lower efficiency than in the ^{241}Am detector system, thereby reducing the size and complexity of the detector background. A separate, larger high-efficiency detector is used to measure the plutonium gamma rays. The disadvantage of plutonium gamma rays is their higher penetrability, which makes effective collimation difficult within the confined space of a through-hole detector. A thin, 1–2-mm attenuating layer of tungsten eliminates the very numerous gamma rays from americium and the x-rays from uranium and plutonium. By shielding out the low-energy photons, we reduce the count rate in the larger detector and the higher-energy gamma rays from plutonium can be counted.

The NaI detectors can be fabricated as cylinders with a through-hole for the fuel rods or as a slab with a hole through the side. Through-hole detectors can be made in various thicknesses and sizes; the detectors would be made equal to the width of the optimized collimator. A single photomultiplier may be adequate if the energy regions of interest do not require good resolution. Increasing the side dimensions of the crystal increases the total efficiency of the detector; however, in noncylindrical detectors or detectors with large through-holes, two photomultiplier tubes must be used to improve light collection and peak resolution. Current systems can be purchased with matched response photomultipliers and balancing circuits to minimize the complications of using two photomultipliers.

IX. DESIGN DETAILS

Figure 30 shows front and side views of the single (dual function) detector system. The single detector is 5.08 x 5.08 x 1.3 cm. This detector would have filters and collimators in the through-hole to attenuate x-rays and americium gamma rays. The two-detector system shown in Figure 31 uses a small 2.54 x 2.54 x 1.3-cm crystal and a 5.08 x 5.08 x 1.3-cm crystal. The small detector in the system measures americium gamma rays with high efficiency and plutonium gamma rays at low efficiency, thereby reducing the total count rate. The total count rate in the small detector can be further controlled with cadmium or tin filters. The larger detector in the paired system will measure the higher energy gamma rays from plutonium. Changing the thickness of the internal tungsten shield controls the total count rate.

Fig. 30. Single plutonium gamma-ray detector, filters included.

X. PERFORMANCE CALCULATIONS

The performance of the single detector system was estimated using the Monte Carlo code. Detector calculations that included the fuel rod, collimator, detector crystal, and shielding were used to estimate the detector efficiency for various geometries. The single detector will count the low-energy gamma rays from ^{241}Am as well as the higher-energy gamma rays from plutonium.

Fig. 31. Two-detector passive system with a large tungsten-shielded detector and a smaller filtered detector.

XI. COLLIMATOR

The optimal collimator width for observing a fuel pellet is a compromise between the rapid change in the signals for a pellet possible with tight collimation and the larger number of counts detected with wide collimation. The expected optimal collimator width should be near the width of the fuel pellet.

The optimal collimator was designed using a dynamic model of the passive gamma-ray system. We simulated a fuel rod with an off-specification pellet passing through the detector. The predicted counts from the moving fuel rod were calculated in sequential time bins. The number of counts in each bin was evaluated with normal statistics to account for statistical fluctuations. The average counts and the variance of the counts over all time bins for the rod were then calculated. The counts recorded by the off-specification pellet were then compared with the average counts from a uniform portion of the rod. The model was run several hundred times to estimate the number of standard deviations the off-specification pellet was from the average and the variance of the number of standard deviations from the average. This simulation accounts for the counting statistics, rod speed, plutonium loading, and collimator width with a single off-specification pellet.

The americium results from the model are shown in Fig. 32 for increasing collimator widths with 2-wt% plutonium fuel pellets, a 1.8-wt% off-specification pellet, and a fuel-rod speed of 1 m/min. Figure 32 shows that the optimal collimator width is very close to the pellet length of 1.3 cm. A smaller collimator provides fewer counts and is less accurate in detecting the off-specification pellet. Larger collimators perform worse because the off-specification pellet makes less difference in the count rate, which reduces the signal contrast.

The same collimator width of 1.3 cm would be used for the plutonium gamma-ray detector system but equivalent collimation is provided using a 1.3-cm-wide crystal because the plutonium gamma rays penetrate the thin collimating sleeve that is used for the ^{241}Am gamma rays. The crystal is made reasonably efficient by using a dimension of 5.08 cm, perpendicular to the rod axis as shown in Fig. 30. When the dynamic model is run using 208-keV gamma rays and different

Fig. 32. Average number of standard deviations from the average counts from the total fuel rod for an off-specification pellet (60-keV gamma rays, increasing collimator widths).

size detectors, the 1.3 x 5.08 x 5.08-cm detector (see Fig. 33) is a good compromise between total efficiency, count rate, and spatial resolution. The number of standard deviations that the off-specification fuel pellet is from the average will be compared using this model.

XII. AMERICIUM DETECTOR

The americium detector design is controlled almost exclusively by the collimator width. The americium gamma rays contribute very little to the detector response outside the collimated region. The thickness of the detector is relatively unimportant because the 60-keV gamma rays have low penetrability. The response of the detector to a fuel rod passing through the detector is shown in Fig. 34. The change in the count rate occurs predominately in the 1-mm width of the collimator.

Fig. 33. Average number of standard deviations from the average counts from the total fuel rod for an off-specification pellet (208-keV gamma rays, increasing detector sizes).

The total americium signal is related to the width of the collimator. The effect of a fuel pellet passing through the detector with 1.8-wt% plutonium instead of the desired 2.0-wt% plutonium is shown in Fig. 35. The spatial resolution is good and the detector can be used to detect the off-specification fuel pellet. The americium detector can also be used to measure the length of fuel-rod zones and the total length of the fuel-rod stack. A large collimator opening of 1.3 cm would allow too many low-energy gamma rays to reach the detector, thereby creating pulse pileup and high-rate counting problems. The count rate should be linearly dependent on the width of the collimator. For 2-wt% Pu-MOX fuel, the peak count rate from americium is approximately 50 kHz with a collimator open to 1.3 mm with no filters.

Fig. 34. Fuel-rod end passing through americium detector with 1-mm collimator.

Fig. 35. Response of low plutonium-loading pellet passing through americium detector.

XIII. PLUTONIUM GAMMA-RAY DETECTOR

The design of the larger plutonium gamma-ray detector is largely controlled by the desired spatial resolution of the detector. The higher energy of the plutonium gamma rays makes effective internal collimation of the through-hole detector difficult. The thickness of the rectangular detector replaces the collimator width while maintaining good efficiency. The response profile of the plutonium detector is shown in Fig. 36. The FWHM is very near the width of the detector. The thin collimator in the detector prevents x-rays and low-energy gamma rays from entering the detector.

The efficiency as a function of detector crystal size is shown in Fig. 37. Once the detector wall thickness is greater than 2 cm, the total efficiency improves only slightly with the detector thickness. The peak count rate with a 1.3-mm-thick collimator is approximately 40 kHz with 2-wt% Pu-MOX. The attenuation of 208-keV gamma rays with increasing thickness of the internal shield is shown in Fig. 38. A 2-mm space should be left around the fuel rod for the shield and filters. The ability to change the internal shielding thickness will allow the detector system to perform well using a variety of fuel loadings.

XIV. DESIGN RECOMMENDATION

The high performance of the plutonium and ^{241}Am detectors allows many design choices. The plutonium can be assayed using a single rectangular detector. The detector can be tuned to produce ideal count rates by changing the thickness of the internal tungsten shield. The addition of a second smaller detector to measure the americium is not justified because of the added complexity. Instead, the single-detector option should be used where a small tungsten collimator is

Fig. 36. Response profile of the plutonium detector to 208-keV gamma rays.

Fig. 37. Peak and total efficiency as a function of detector wall thickness for 208-keV gamma rays.

used to count the ^{241}Am . The width of the collimator should be 1–2 mm to keep the total count rates down. The collimator should be adjustable to allow for optimization, and space should be left for additional filters. A 1.3 x 5.08 x 5.08-cm rectangular detector with a 2-cm-diameter through-hole meets these requirements. Additional tungsten on the outside of the crystal shields the detector from sources outside the crystal.

XV. ACTIVE GAMMA-RAY DENSITOMETER

Fuel pellet density can be measured using an active gamma-ray transmission system. Transmission gauges are a well-established commercial technology. The system used must accurately measure the density of the completed fuel rod and be able to detect gaps between fuel pellets. The transmission system is used for gap checking because it is difficult to measure small changes in the passive-system signal caused by small gaps between pellets.

Fig. 38. Attenuation of 208-keV gamma rays with increasing tungsten shield thickness.

A simple transmission and solid angle calculation indicates that a ^{137}Cs source of 0.33 mCi would be the minimum source strength required to identify a 0.3-mm gap in a rod moving 1 m/min at a three-sigma level. The collimator used in the calculation was a 1-mm collimator tube 4 cm long. A detector-collimator system using a fan beam instead of a tube collimator could use a smaller source to reach the three-sigma level.

A ^{137}Cs source is a good choice for the transmission source because the gamma rays can penetrate the fuel rod but are also reasonably attenuated. The minor complication in using cesium is the presence of a small source of 662-keV gamma rays from americium in the MOX rod. The maximum strength of a ^{137}Cs source is largely controlled by the shielding and handling requirements of the source.

XVI. CONCLUSIONS FOR THE PASSIVE GAMMA-RAY DETECTOR

Active and passive measurements of the completed MOX fuel rods are useful in verifying the composition and design of the fuel rod. This provides a useful safeguard verification in the fabrication process. The measured data can also be used to increase the transparency of the fuel fabrication process.

The suggested passive gamma-ray detector should be a 1.3 x 5.08 x 5.08-cm rectangular crystal with a 2-cm-diameter through-hole. The through-hole allows an internal shield to be used. The shield attenuates ^{241}Am gamma rays, keeping the total count rate reasonable. The large number of plutonium 208-keV gamma rays from ^{241}Pu provides an adequate measure of the plutonium loading on-line. The ^{241}Am is detected using a small 1-2-mm-wide collimator in the internal tungsten shield. The collimator will allow tracking of the batch of plutonium used in the MOX fabrication. The use of a single combined function detector system will provide the simplest system that can give both the ^{241}Am and the plutonium uniformity.

Combined Fuel-Rod Scanner. The combined rod scanner would be needed for complex fuel that contained plutonium and varied in ^{235}U enrichment. To predict the fuel's performance in the reactor, it is necessary to know the fuel composition in detail. For safeguards, it is necessary to verify the plutonium and enriched uranium masses before fabricating the fuel assemblies and making the interior rods inaccessible.

Table VIII lists various components of the complete rod scanner together with the material that is being measured. The combination of the plutonium isotopics and the passive neutron

TABLE VIII. MOX Rod Scanner Components	
Component	Measurement Target
^{252}Cf Active Assay	^{235}U , ^{239}Pu , ^{241}Pu
Passive Neutron Coincidence	$^{240}\text{Pu}_{\text{Effective}}$
HRGS	Plutonium isotopics, ^{241}Am
NaI (60 keV)	^{241}Am pellet uniformity
NaI (150-500 keV)	Plutonium pellet uniformity
^{137}Cs Transmission Source	Pellet gaps, density

measurement gives the total plutonium mass in the rod. Knowing the plutonium content from the passive measurements makes it possible to solve for the ^{235}U content from the active neutron measurement.

The HPGe detector would be operated in the attended mode eliminating the complexity of developing a liquid-nitrogen-cooled HPGe system that runs unattended. The NaI detectors could be operated in the unattended mode for the fuel-rod quality control. The commercial densitometer could also be operable unattended. A conceptual drawing of the composite passive assay system is shown in Fig. 39.

Fig. 39. Proposed arrangement of the components of a fuel-rod scanner system to be used by both the operator and safeguards inspectors.

The plutonium mass is measured with the passive neutron coincidence counter. The number of plutonium neutrons is correlated with the mass of plutonium using the isotopic ratios. The isotopics can be directly measured using the HRGS concurrently with the neutron measurements for completely independent results. The passive gamma-ray detectors ensure the fabricated fuel rod matches the declared plutonium loading and fuel pin loading profiles. This information is also valuable to the operator for the fuel-rod quality control. The active gamma-ray transmissions are needed to verify that there are no gaps between pellets and that the completed fuel rods are consistent.

Figure 39 shows a splitting of the signals from the passive components of the MOX rod scanner for joint use by the inspectors and the facility operator. The IAEA would have access to those parts of the system that are of safeguards interest; the plutonium isotopics measurement with the HRGS and the ^{240}Pu neutron coincidence measurement. These measurements would be performed in the attended mode. The operator would have routine unrestricted use of the full system between the inspection visits.

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