338
4-19-79

OR. 2457

LA-7645-MS

Informal Report
(ISPO-47)

Passive Neutron Assay of Irradiated Nuclear Fuels

University of California

PROGRAM FOR
TECHNICAL ASSISTANCE
TO IAEA SAFEGUARDS

USA
IAEA

Department of Energy
Safeguards & Security

LOS ALAMOS SCIENTIFIC LABORATORY
Post Office Box 1663 Los Alamos, New Mexico 87545

LA-7645-MS
Informal Report
(ISPO-47)

UC-15
Issued: February 1979

Passive Neutron Assay of Irradiated Nuclear Fuels

S. T. Hsue
J. E. Stewart
K. Kaieda*
J. K. Halbig
J. R. Phillips
D. M. Lee
C. R. Hatcher

*Visiting Staff Member. Japan Atomic Energy Research Institute, Tokai Research Establishment, Tokai-mura, Naka-gun, Ibaraki-ken, Japan.

PASSIVE NEUTRON ASSAY OF IRRADIATED NUCLEAR FUELS

by

S. T. Hsue, J. E. Stewart, K. Kaieda,
J. K. Halbig, J. R. Phillips, D. M. Lee, C. R. Hatcher

ABSTRACT

Passive neutron assay of irradiated nuclear fuel has been investigated by calculations and experiments as a simple, complementary technique to the gamma assay. From the calculations we have found that the neutron emission arises mainly from the curium isotopes, the neutrons exhibit very good penetrability of the assemblies, and the neutron multiplication is not affected by the burnup. From the experiments on BWR and PWR assemblies, we have found that the neutron emission rate is proportional to burnup raised to 3.4 power. Our investigations indicate that the passive neutron assay is a simple and useful technique to determine the consistency of burnups between assemblies.

I. INTRODUCTION

Nondestructive assay (NDA) methods for the determination of the burnup and cooling times of irradiated nuclear fuels are needed to safeguard against diversion of fissile materials to weapon use. Because irradiated fuels represent a possible source of plutonium, it is essential that the International Atomic Energy Agency (IAEA) have a measurement capability for effective international safeguards. For easy implementation the preferred NDA method should use simple, portable equipment that can be carried from site to site for rapid and reasonably accurate assays.

Gamma-spectroscopy has been investigated as a measurement method for determining nondestructively the burnup and cooling time of irradiated fuels.[1-4] The measurement is usually performed in a hot cell or in a cooling pond. The method is based on measurement of the concentrations of radioactive burnup monitors (^{95}Zr, ^{137}Cs, or ^{144}Ce-^{144}Pr) or, in recent years, the activity ratios ($^{134}Cs/^{137}Cs$ or $^{154}Eu/^{137}Cs$). The problems related with gamma-spectroscopic measurements have recently been reviewed.[1-3] The major problems are the fission products (or their precursors) may migrate at high fuel temperature (^{106}Ru-^{106}Rh, ^{134}Cs, and ^{137}Cs); the intensities of gamma rays from the fission products in the interiors of assemblies are substantially attenuated by the relatively dense fuel rods (density ~ 10 g/cm^3); these activity ratios are sensitive to the epithermal-to-thermal flux ratio in addition to the fuel burnup. Burnup measurements cannot be performed by gamma assay on fuels shortly after discharge because the gamma activity is dominated by relatively short-lived isotopes (e.g., ^{140}La), reflective of recent reactor power level. Gamma-spectroscopic measurement also requires high-resolution germanium detector and multichannel analyzer (MCA), both of which necessitate well-trained personnel to obtain meaningful results. Though it is foreseeable that MCA with automatic features can be developed, thus minimizing the need for comprehensive training of the operator, a basic problem of the gamma assay method remains--it is only sensitive to the outer layers of rods of a fuel assembly. An alternative NDA method more sensitive to interior rods of an assembly is highly desirable for spent fuel inspection.

In addition to the various gamma rays, irradiated fuels also emit neutrons. Neutrons are less subject to self-absorption in the fuel assembly than are gamma rays. Passive neutron assay has been identified as a potentially useful inspection assay method of spent fuel both by the recent review[2,3] and in the IAEA Advisory Group Meeting.[4]

At the Los Alamos Scientific Laboratory (LASL) we have performed calculations and experiments to investigate the merits of passive neutron assay. From the calculations we have found that the neutron emission arises mainly from the curium isotopes,[5] and that the neutrons exhibit very good penetrability of the assemblies.[6] From the experiments on boiling water reactor (BWR) and pressurized water reactor (PWR) assemblies,[7] we have found that the neutron emission rate is proportional to burnup raised to ~ 3.4 power. This report will summarize our findings with passive neutron assay.

II. ORIGINS OF THE PASSIVE NEUTRONS

In a spent fuel assembly, neutron emission arises from spontaneous fissioning of the even isotopes of plutonium and curium and from the (α,n) reaction in oxide arising from the alpha emission of the various isotopes of plutonium, americium, and curium. Figure 1 shows the major pathways of production of the transuranium isotopes in fuels irradiated in light water reactors (LWR). While it is possible to calculate the buildup of the transuranium isotopes, the uncertainty of such a calculation would be large because the neutron capture cross sections of some of these isotopes are not well established.

The following calculations are based on the results of a postirradiation examination study on Trino Vercellese reactor fuel.[8] In this study, which was done in Italy in 1976, the burnups of each of the fuel samples were determined by mass-spectrometric measurements of ^{148}Nd. The uranium, plutonium and americium isotopic concentrations were determined by mass spectrometry with typical precision of better than 0.5%. The curium isotopic contents were determined by means of alpha-spectrometry measurement. The standard deviations of the measurements of the curium concentrations are better than 5%. All the isotopic compositions were referred to reactor shutdown, except for ^{241}Am, which was expressed at the time of measurement.

To calculate the passive neutron emission rate arising from the spontaneous fission and (α,n) reaction in oxide, we used the nuclear data information shown in Table I.[9] Using the isotopic composition of the fuel samples as determined by destructive analysis, we calculated the neutron emission rate at discharge and at subsequent cooling time for burnups ranging between 13 000 and 27 000 MWD/MTU. Figure 2 shows the calculated neutron emission rate for a 12 859 MWD/MTU burnup fuel sample; Fig. 3 shows the neutron yield for a 26 884 MWD/MTU burnup fuel. The percentage contributions from the various fissionable isotopes to passive neutron emission rates are shown in Table II. Curium isotopes are the main contributors of the passive neutrons for burnups that exceed 15 000 MWD/MTU, accounting for at least 70% of the neutrons up to a cooling time of seven years. For cooling times of less than two years, the two curium isotopes dominate the neutron emission (greater than 70%) with ^{242}Cm the major contributor at short cooling times. Only

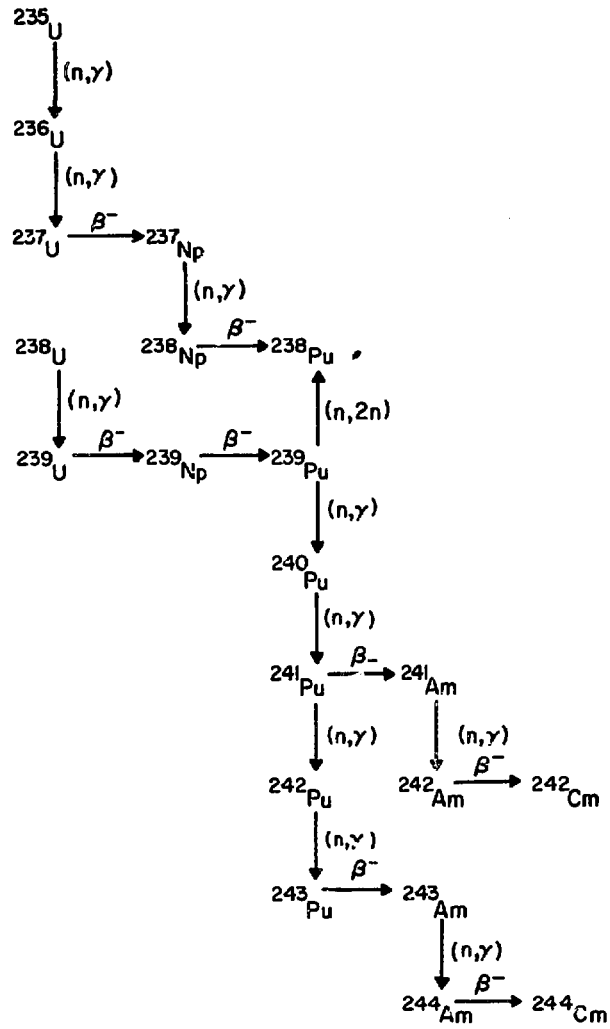

Fig. 1.
Major pathways of production of transuranium isotopes.

TABLE I

SPONTANEOUS FISSION AND ALPHA-INDUCED NEUTRON YIELD

Source	Total Half-life (years)	Yield neutron g s		
		Spontaneous fission	(α,n) in oxide	Total
^{235}U	7.10×10^{8}	5.12×10^{-4}	1.11×10^{-3}	1.62×10^{-3}
^{238}U	4.51×10^{9}	1.14×10^{-2}	1.33×10^{-4}	1.15×10^{-2}
^{238}Pu	8.78×10^{1}	2.51×10^{3}	1.84×10^{4}	2.09×10^{4}
^{239}Pu	2.44×10^{4}	2.16×10^{-2}	5.38×10^{1}	5.38×10^{1}
^{240}Pu	6.55×10^{3}	0.14×10^{2}	2.01×10^{2}	1.11×10^{3}
^{241}Pu	1.47×10^{1}	1.10×10^{-2}	1.96	1.97
^{242}Pu	3.87×10^{5}	1.68×10^{3}	2.87	1.68×10^{3}
^{241}Am	4.33×10^{2}	5.79×10^{-1}	3.64×10^{3}	3.64×10^{3}
^{242}Cm	4.46×10^{-1}	2.13×10^{7}	4.75×10^{6}	2.60×10^{7}
^{244}Cm	1.81×10^{1}	1.10×10^{7}	1.03×10^{5}	1.12×10^{7}

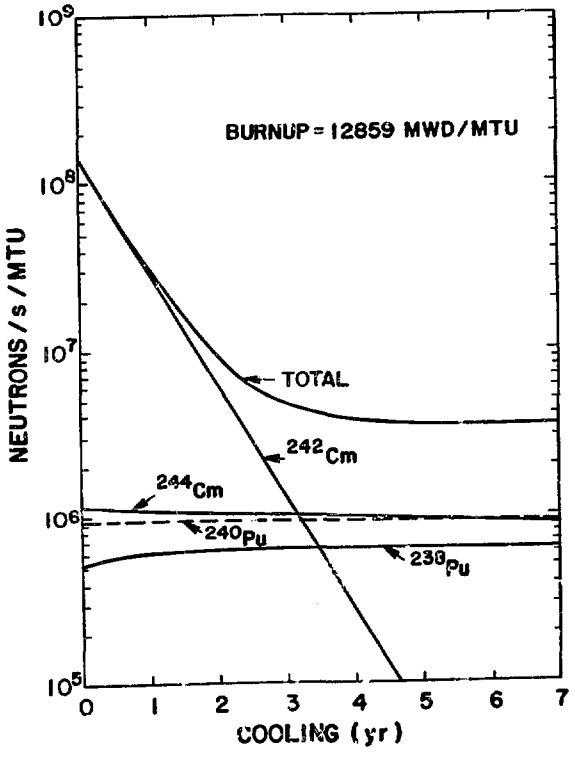

Fig. 2.
Neutrons per second from ^{238}Pu, ^{240}Pu, ^{242}Cm, and ^{244}Cm isotopes at a burnup of 12 859 MWD/MTU. The total includes contributions from the other uranium, plutonium, and americium isotopes of the Trino reactor fuel.

TABLE II

PERCENTAGE CONTRIBUTION TO PASSIVE NEUTRON EMISSION RATE FROM THE FISSIONABLE ISOTOPES

Cooling Time (years)	^{235}U	^{238}U	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	^{242}Cm	^{244}Cm	^{241}Am*
BU=12859 MWD/MTU										
0	2.3-5**	8.0-3	3.8-1	1.8-1	6.8-1	5.7-4	5.6-2	97.6	8.3-1	2.2-1
1	9.9-5	3.5-2	1.9	7.7-1	2.9	2.4-3	2.4-1	89.5	3.5	1.2
2	3.3-4	1.2-1	6.6	2.6	9.9	7.6-3	8.2-1	64.0	11.3	4.6
3	6.7-4	2.3-1	13.4	5.2	19.8	1.5-2	1.6	27.2	21.8	10.6
5	9.0-4	3.1-1	17.9	7.0	26.5	1.8-2	2.2	1.6	27.0	17.4
7	9.0-4	3.2-1	18.0	7.0	26.7	1.6-2	2.2	7.3-2	25.1	20.5
BU=15170 MWD/MTU										
0	1.4-5	53.-3	3.5-1	1.2-1	5.4-1	4.9-4	5.9-2	94.0	4.7	1.5-1
1	5.3-5	2.0-2	1.6	4.7-1	2.1	1.8-3	2.3-1	77.1	17.7	7.3-1
2	1.4-4	5.3-2	4.2	1.2	5.4	4.4-3	5.9-1	42.2	44.0	2.3
3	2.1-4	8.1-2	6.5	1.8	8.3	6.4-3	9.0-1	13.6	64.7	4.0
5	2.5-4	9.7-2	7.8	2.2	10.0	7.0-3	1.1	7.3-1	71.9	6.1
7	2.6-4	1.0-1	8.2	2.3	10.5	6.7-3	1.1	3.4-2	70.0	7.7
BU=20602 MWD/MTU										
0	4.4-6	2.1-3	3.2-1	5.8-2	3.2-1	3.3-4	5.6-2	88.8	10.3	1.2-1
1	1.5-5	7.0-3	1.3	2.0-1	1.1	1.1-3	1.9-1	63.3	33.5	5.0-1
2	3.0-5	1.4-2	2.7	4.0-1	2.2	2.1-3	3.8-1	27.3	65.8	1.2
3	4.0-5	1.9-2	3.5	5.3-1	2.9	2.6-3	5.0-1	7.6	83.1	1.8
5	4.6-5	2.1-2	4.0	6.0-1	3.3	2.7-3	5.8-1	3.9-1	88.5	2.6
7	4.9-5	2.3-2	4.3	6.5-1	3.5	2.6-3	6.2-1	1.9-2	87.6	3.2
BU=26884 MWD/MTU										
0	1.8-6	1.2-3	2.9-1	3.2-2	2.2-1	2.4-4	6.1-2	83.1	16.2	9.7-2
1	5.3-6	3.4-3	1.0	9.4-2	6.6-1	6.8-4	1.8-1	51.9	45.8	3.4-1
2	9.2-6	6.0-3	1.8	1.6-1	1.1	1.1-3	3.1-1	19.1	76.8	7.0-1
3	1.1-5	7.3-3	2.2	2.0-1	1.4	1.3-3	3.8-1	4.9	89.9	9.7-1
5	1.3-5	8.2-3	2.5	2.2-1	1.6	1.3-3	4.3-1	2.5-1	93.7	1.3
7	1.3-5	8.8-3	2.7	2.4-1	1.7	1.3-3	4.6-1	1.2-2	93.2	1.7

*^{241}Am content is expressed at three years after irradiation at the time of measurement.

**2.3-5 is defined to be 2.3x10-5.

for low burnup (less than 15 000 MWD/MTU) and for long cooling times (greater than two years) are the contributions from the Pu isotopes significant to the neutron emission rate. In the ^{242}Cm and ^{244}Cm decay, spontaneous fission is the major source of the passive neutron emission (see Table I).

If all the assemblies being verified have cooling times longer than two years, then cooling time corrections can be neglected because most of the ^{242}Cm activity ($T_{\frac{1}{2}}$ = 163 days) will have decayed away, and the other neutron emitting isotopes

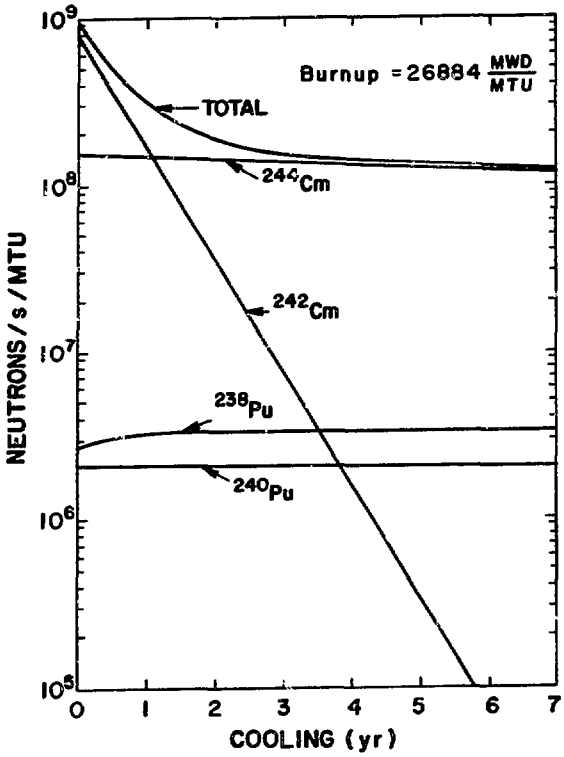

Fig. 3.
Neutrons per second from ^{238}Pu, ^{240}Pu, ^{242}Cm, and ^{244}Cm isotopes at a burnup of 26 884 MWD/MTU. The total includes contributions from the other uranium, plutonium, and americium isotopes of the Trino reactor fuel.

have half-lives longer than 14 years. For cooling times less than six months (burnup less than 27 000 MWD/MTU), the dominant neutron emission source decays with the 163 day half-life. Between these two limits, the neutron emission rate depends on the burnup and cooling times.

These calculations indicate that passive neutron emission from irradiated fuels depends on the accumulation of the ^{242}Cm and ^{244}Cm isotopes, which in turn depends on the burnup.

III. PROPERTIES OF PASSIVE NEUTRON ASSAY

For inspection purposes, the spent fuels normally will be assayed as whole assemblies submerged in water. To explore the properties of passive neutron assay such as response versus distance, assay penetrability, and the multiplication effect, we performed the following calculations.

A Monte Carlo transport code, MCNP,[10] was used for the calculations. A Trino reactor spent fuel assembly was mocked up as shown in Fig. 4. For ease in computation, the assembly was

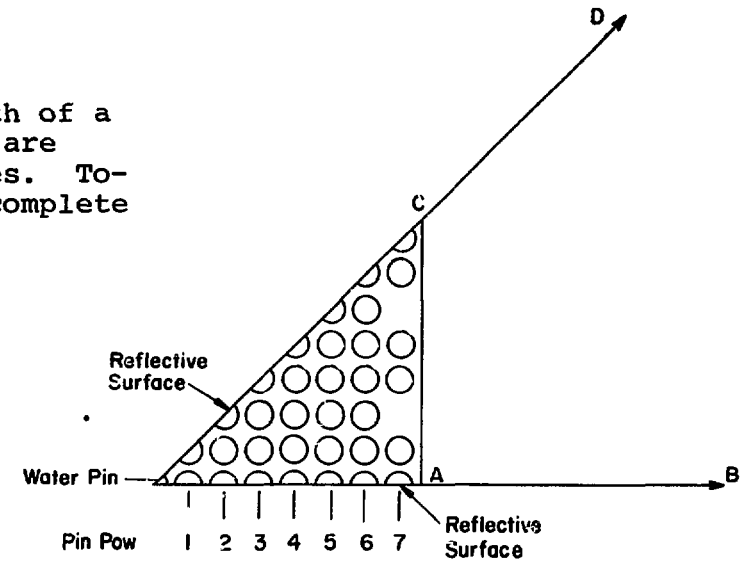

Fig. 4.
Cross section of one-eighth of a Trino assembly; AB and CD are neutron reflective surfaces. Together they simulate the complete spent fuel assembly.

assumed to be infinite in length, and the fuel pin cladding was mixed with the fuel. These assumptions were not expected to affect results significantly. The neutron sources for different burnups within the fuel rods were taken from the results of the previous calculations in Sec. II and based on postirradiation destructive examinations. The neutron sources in the assembly were generated assuming a cooling time of one year. A water temperature of 54°C was assumed.

A. Neutron Response Versus Distance

The purpose of this calculation is to optimize the location of the neutron detector and to determine the sensitivity of passive neutron assay to variation in detector-to-source distances.

Figures 5 and 6 represent spatial flux distributions (at the assembly midplane) in water outside one assembly. The burnup value chosen for the plot is 20 602 MWD/MTU. In Fig. 5, distance is measured along line AB (midplane normal as referred to in Fig. 4). In Fig. 6, distance is along line CD (midplane diagonal).

In general, the midplane normal flux is higher than the midplane diagonal flux at the same distance from the assembly surface. Notice that for the thermal neutrons (energy less than

$5x10^{-7}$ MeV) the flux exhibits a smaller slope through the region 0-3 cm than in the region further from the assembly (see Fig. 5) due to the slowing down of fast neutrons. A thermal neutron detector so positioned with respect to the assembly can minimize the effect of distance variations as well as maximize the neutron response.

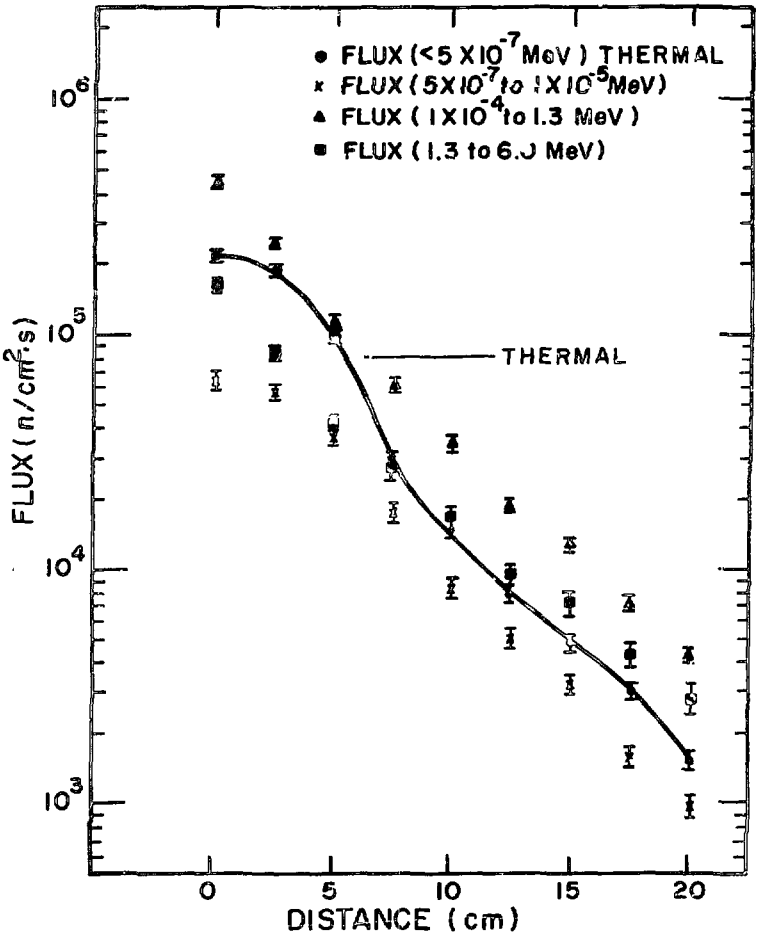

Fig. 5.
Flux along midplane normal from the assembly.

B. Assay Penetrability

The purpose of this calculation is to investigate the spatial distribution of the importance of fuel pins in contributing to the detector signal. Or, in other words, how well does a neutron from inside the assembly escape from the assembly? An intermediate burnup level (20 602 MWD/MTU) assembly was chosen.

In the first calculation, source neutrons were started uniformly in the volume of the first fuel pin row; in the second calculation rows 1 through 3 (see Fig. 4 for the definition of

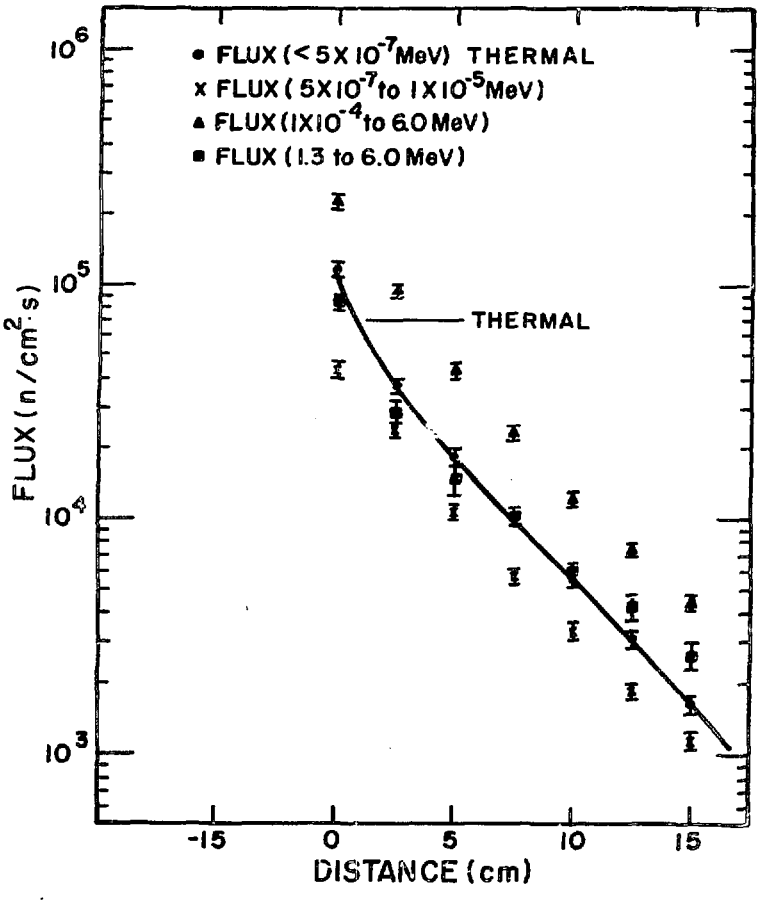

Fig. 6.
Flux along midplane diagonal from the assembly.

row); in the third calculation, rows 1 through 5; and in the fourth calculation, rows 1 through 7 (all fuel pins). Note that the first fuel pin row is actually the second row of pins, the first being a central water pin. These calculations allowed the determination of the relative flux contribution of various combinations of fuel pin rows. The results of the calculation are summarized in Table III. In this table the flux contribution per pin from all the fuel pins is normalized to one. If the flux contribution per pin is greater than one, that means these rows of fuel pins will make an above-average contribution to the signals of passive neutron assay. The standard deviation (S.D.) in Table III is a measure of the uniformity of flux contribution from the various rows of fuel pins, the smaller the standard deviation, the more uniform the contributions.

The results indicate a relatively uniform flux contribution within the calculational precision of about 8%. Thus, passive neutron assay "sees" all the fuel pins about equally well. The reason that the neutron signals originating in the inner fuel pin rows can penetrate the assembly is because the source neutrons from the inner rows slow down in the water and induce fissioning in the outer fuel rows. The neutrons from the induced fissions can then penetrate the assembly. In this respect the passive neutron assay is significantly different from the gamma assay, which detects signals from the outer layers of the assembly.

C. Multiplication Effect

Next, the neutron multiplication effect was examined. While the Cm isotopes (the origin of most of the passive neutrons) within the fuel rods may be proportional to burnup as indicated in the post irradiation examination,[8] the passive neutron assay response may be different due to neutron multiplication dependence on burnup.

In this calculation, the neutron sources for different burnups within the fuel rods were taken from previous calculations. The results are shown in Table IV. From the table, the system multiplication factor is rather constant (standard deviation

TABLE III

RELATIVE FLUX CONTRIBUTION TO PASSIVE NEUTRON ASSAY FROM DIFFERENT FUEL PIN ROWS

THERMAL FLUX ALONG MIDPLANE NORMAL

Position*	5	7.5	10	12.5	15	17.5	20
Pin Row							
1	.95	1.20	.95	1.07	.86	.79	.83
2,3	1.00	.92	1.16	1.05	1.17	1.01	1.04
4,5	.72	1.08	.84	.87	.79	.75	1.01
6,7	1.23	.95	1.06	1.08	1.11	1.22	.99
S.D.	.21	.13	.14	.099	.19	.22	.094

FAST FLUX (1×10^{-4} to 1.3 MeV) ALONG MIDPLANE NORMAL

Position*	5	7.5	10	12.5	15	17.5	20
Pin Row							
1	1.05	1.20	.96	1.16	.88	.97	.87
2,3	1.14	.92	1.03	1.00	.86	.92	1.00
4,5	.86	.98	1.09	1.02	.99	1.18	.95
6,7	1.00	.94	.91	.97	1.09	.89	1.05
S.D.	.12	.13	.079	.084	.11	.13	.068

THERMAL FLUX ALONG MIDPLANE DIAGONAL

Position*	0	2.5	5	7.5	10	12.5	15
Pin Row							
1	.92	.86	.84	.85	.88	.94	.98
2,3	.85	.88	1.00	.95	.80	.80	.84
4,5	.69	1.03	.90	1.09	1.06	1.14	.97
6,7	1.33	1.04	1.09	.96	1.06	.99	1.09
S.D.	.29	.097	.11	.098	.13	.14	.10

FAST FLUX (1×10^{-4} to 1.3 MeV) ALONG MIDPLANE DIAGONAL

Position*	0	2.5	5	7.5	10	12.5	15
Pin Row							
1	.81	.84	.90	.96	.99	.91	.96
2,3	.80	.69	.88	.79	.89	.79	.75
4,5	.96	1.53	.95	.94	1.10	1.02	1.04
6,7	1.15	.72	1.10	1.14	.97	1.10	1.08
S.D.	.16	.39	.099	.14	.087	.14	.15

*Distance from the surface or corner of the assembly in cm.

1.8%) with burnups ranging from 13 000 to 27 000 MWD/MTU. This illustrates the multiplication factor is not significantly affected by isotopic differences over the range of burnup considered. The multiplication factor is mainly a function of the fuel density within the rods, and the spatial distribution of the fuel rods in the assembly. If the purpose of the assay is to determine the consistency between burnup and neutron emission rate, the multiplication effect can be ignored.

TABLE IV

MULTIPLICATION FACTOR FOR VARIOUS BURNUPS

Burnup (MWD/MTU)	Multiplication
12 859	1.57
15 170	1.54
19 208	1.52
20 602	1.54
23 557	1.49
25 258	1.50
26 884	1.51
S. D. (%)	1.8

IV. MEASUREMENT OF PASSIVE NEUTRONS FROM IRRADIATED FUEL

We have performed two separate passive neutron measurements of irradiated fuels: a BWR facility and a PWR facility. The BWR has a power rating of 63 MWe and has been in operation since 1962. The PWR has a power rating of 1100 MWe and has been in operation since 1973.

A. Response Versus Distance

At the BWR facility, we have measured the neutron response as the detector-to-assembly distance was increased. The detector, a fission chamber containing 58 mg ^{235}U deposit, was submerged in a tube at approximately the midplane of an assembly. The electronics used consisted of an amplifier, a discriminator with the discrimination level set above the alpha signals, and a scaler. The spent fuel assembly, BRP-6, has a declared burnup of 17 814 MWD/MTU and a cooling time of 842 days. The detector position was fixed with the spent fuel assembly being moved.

The results, together with the calculated values for thermal neutrons, are shown in Fig. 7. The calculated neutron count at a distance of 12 cm was normalized to the measured value. The calculated precisions ranged from 6 to 9%. It is observed that for thermal neutrons, the measured diffusion length of 4.3 cm

agrees well with the calculation. In general, the passive thermal neutron counting rate decreases by an order of magnitude when the detector-assembly distance is increased by 10 cm. In Fig. 7 the relative response from an air ion chamber measured on the same assembly was also shown. In comparison, the ionization chamber signal reduces by a factor of ten when the distance is increased by $\sim$30 cm. These results illustrate that the passive neutron signal has a shorter range in water than the ionization chamber assay and also that the neutron contribution from the (γ,n) reaction on water (or other low-Z material in the pool) is negligible compared with the neutrons emitted from the spent fuel. The results also show that fission chambers are insensitive to gamma radiations.

B. Neutron Emission Rate Versus Burnup

At the BWR facility, we have measured the neutron emission rate from six spent fuel assemblies with burnups ranging from 4356 to 18 804 MWD/MTU. The measurements were performed with a 58-mg ^{235}U fission chamber located in a tube placed $\sim$27 cm from the midplane of the assembly. The neutron counting rates for various burnups are listed in Table V and also shown in Fig.

TABLE V

PASSIVE NEUTRON ASSAY OF
BOILING WATER REACTOR SPENT ASSEMBLIES

Assembly	Burnup (MWD/MTU)	Cooling Time (Days)	Counting Rate (cps)
BWR 1	4356	842	.038 $\pm$ .010
BWR 2	8883	1452	.373 $\pm$ .031
BWR 3	13332	843	.981 $\pm$ .037
BWR 4	15264	843	3.296 $\pm$ .069
BWR 5	17122	844	4.064 $\pm$ .076
BWR 6	17814	842	4.125 $\pm$ .102

8. We found that the passive neutron counting rate is proportional to burnup raised to 3.38 power. No cooling time correction has been made because all the assemblies had long and similar cooling times.

We have also performed axial scans of BRP spent fuels with fission chamber. Since we found that the neutron emission rate is proportional to $(\text{burnup})^{3.38}$, the $(\text{neutron emission rate})^{1/3.38}$ should then be proportional to burnup. The axial neutron scan can then be compared with the axial gamma scan using the Ge detector. Figure 9 shows such a comparison. We

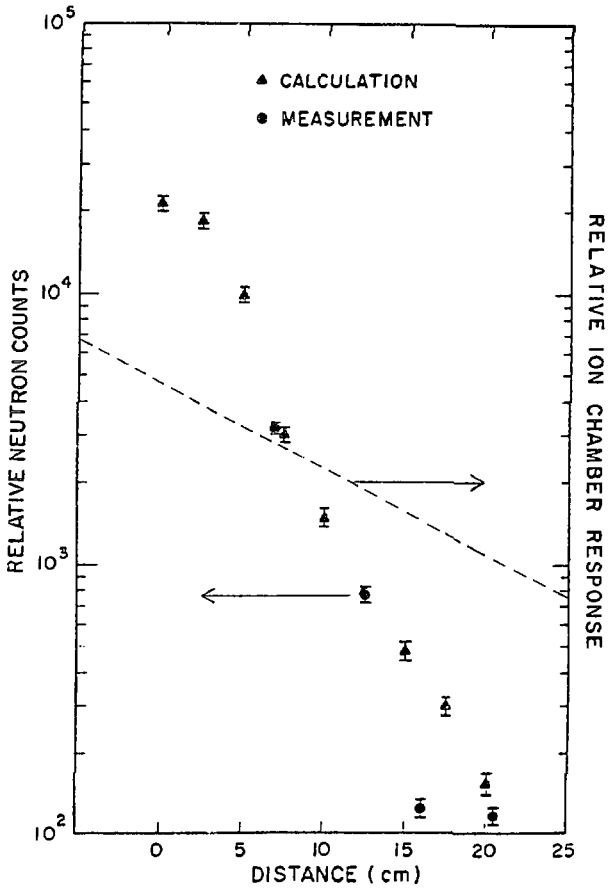

Fig. 7.
Comparison of the measured and calculated relative thermal neutron response as a function of increasing detector-to-assembly distance.

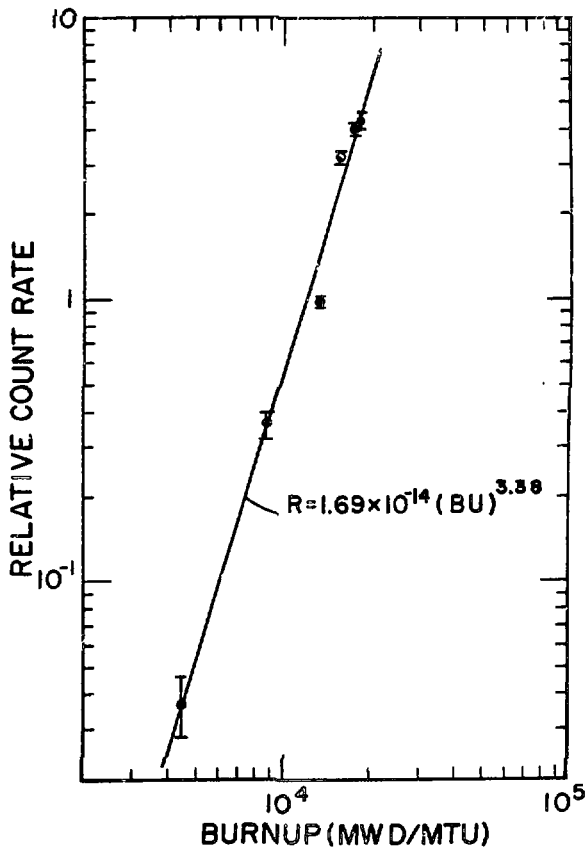

Fig. 8.
Relative fission chamber response versus burnup for six spent fuel assemblies from a BWR.

found that the neutron and gamma scan agree well, supporting further that the neutron emission rate is proportional to burnup raised to 3.38 power.

Because the neutron count rate at the BWR exercise was relatively low (maximum counting rate 4 cps), for the PWR exercise we used a fission chamber with higher ^{235}U loading (135 mg) and the chamber was placed closer to the assemblies than the previous exercise (12 cm from midplane). The neutron counts for a 300 s assay on 9 assemblies is shown in Table VI and Fig. 10. The error bars of the PWR measurement are smaller than the data points. We found that the fission chamber counting rates are all proportional to burnup raised to the 3.35 power. A cooling time correction has not been made. To make the cooling

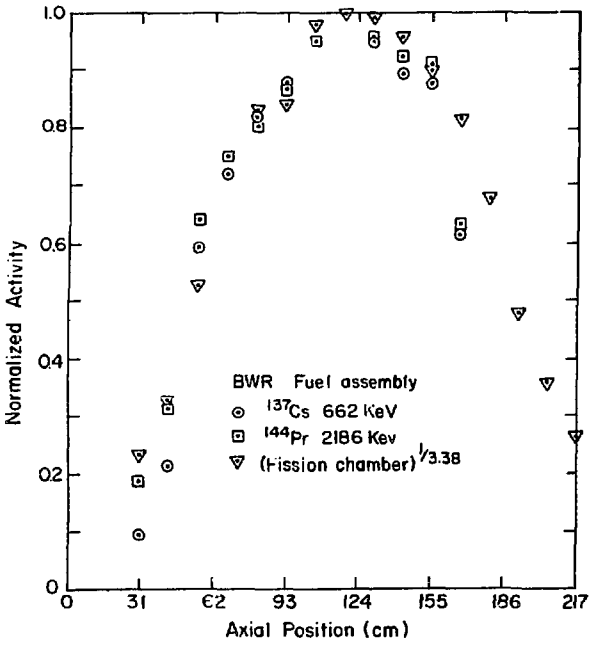

Fig. 9.
Axial profile of fission chamber response raised to 1/3.38 power compared with ^{137}Cs and ^{144}Pr activity profile.

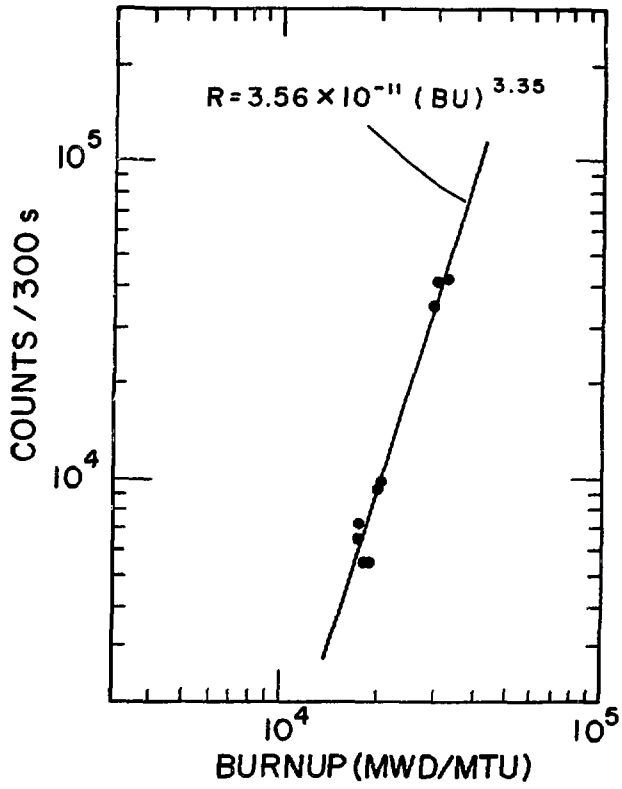

Fig. 10.
Relative fission chamber response versus burnup for nine spent fuel assemblies from a PWR.

TABLE VI

PWR SPENT ASSEMBLIES
MEASURED BY PASSIVE NEUTRON ASSAY

Assembly	Burnup (MWD/MTU)	Cooling Time (Days)	Counts/300 s
PWR 1	17404	528	7195
PWR 2	17776	832	6493
PWR 3	18279	528	5569
PWR 4	18723	837	5501
PWR 5	20066	527	9361
PWR 6	20252	527	9754
PWR 7	29129	140	36255
PWR 8	31851	279	41292
PWR 9	32185	279	41436

correction, it is necessary to know the curium isotopic ratio for burnups ranging from 17 000 to 32 000 MWD/MTU. Although we have some indications of this isotopic ratio up to a burnup of 27 000 MWD/MTU for Trino reactor fuel, we do not know the curium buildup beyond this burnup.

V. CONCLUSIONS

The agreement of the counting rate dependency on burnup between the two exercises is rather surprising since a boiling water reactor and a pressurized water reactor may have rather different actinide buildup. This agreement may be incidental; however, the results indicate that the passive neutron assay is a useful technique to determine consistency of burnups between assemblies.

Several salient features of passive neutron assay emerge from our study.

1. Passive neutron assay uses simple room-temperature detector (fission chamber) requiring no liquid nitrogen as in Ge

detectors. The simplicity of the assay electronics (amplifier, discriminator, and scaler) enhances the reliability of apparatus. The measurement and data processing are straightforward.

2. Passive neutrons are more penetrating than gamma rays. The inner rods of an assembly can be detected in neutron assay.

3. A possible diversion scenario is the substitution of the spent fuel assembly with, for example, ^{60}Co rods, which can be easily produced in a reactor. Such a diversion can be easily detected by passive neutron assay since the activated rod has no neutron signal.

4. The neutron counting rate is reasonable and can be easily increased by reducing the assembly-detector distance.

While our study indicates that passive neutron assay is a useful assay technique and can well compliment the gamma assay of spent fuels, further investigations will clarify the nature of passive neutron assay: (1) the effect of cooling time on the neutron signals, and (2) investigate detectors other than fission chamber that can be used in passive neutron assay. Fission chambers contain fissile material. This may cause difficulty in shipping because of legal problems.

ACKNOWLEDGMENTS

We would like to express our thanks to R. Voll, H. Brazydlo, L. Monshor of Consumers Power Co., and D. O'Boyle, G. Perdikis, D. Walden, and M. Peterson of Commonwealth Edison for their help in the experimental portion of this study. The design and installation of apparatus by S. Beach of LASL Group Q-1 is also appreciated.

REFERENCES

1. T. N. Dragnev, "Experimental Techniques for Measuring Burn-up Nondestructive Techniques: Gamma Spectroscopy," International Atomic Energy Agency report IAEA/STR-48 (October 1970).

2. S. T. Hsue, T. W. Crane, W. L. Talbert, and J. C. Lee, "Nondestructive Assay Methods for Irradiated Fuels," Los Alamos Scientific Laboratory report LA-6923 (January 1978).

3. S. T. Hsue, "Methods for the Nondestructive Assay of Irradiated Fuels for Safeguards," Atomic Energy Review 16, No. 1, 89-128 (1978).

4. "Advisory Group Meeting on the Nondestructive Analysis of Irradiated Power Reactor Fuel," International Atomic Energy Agency report AG-11 (April 1977).

5. S. T. Hsue, "Passive Neutron Assay of Irradiated Fuel," Los Alamos Scientific Laboratory report LA-6849-PR (August 1977).

6. J. E. Stewart, S. T. Hsue, and K. Kaieda, "The Neutron Assay of Spent PWR Fuel Assemblies," to be published in a Los Alamos Scientific Laboratory progress report.

7. S. T. Hsue, K. Kaieda, J. E. Stewart, and J. K. Halbig, "The Neutron Assay of Spent Fuel Assemblies," to be published in a Los Alamos Scientific Laboratory progress report.

8. P. Barbero, G. Bidoglio, M. Bresesti, R. Chevalier, D. D'Adamo, S. Facchetti, A. Federici, G. Guzzi, L. Lezzoli, F. Mannone, F. Marell, P. R. Trincherini, G. Buscaglia, A. Drago, R. Faccelli, A. Frigo, E. Ghezzi, R. Klersy, K. H. Schrader, A. Schuerenkaemper, R. Dierckx, J. Biteau, G. Cottone, A. Cricchio, L. Koch, R. Bannella, M. Paoletti Gualandi, and P. Peroni, "Post Irradiation Examination of the Fuel Discharged From the Trino Vercellese Reactor After the 2nd Irradiation Cycle," Joint Nuclear Research Centre (Ispra and Karlsruhe) and Ente Nazionale for l'Energie Electtrica-Rome (Italy) report EUR-5605e (1976).

9. Personal communication from M. Bresesti, Joint Nuclear Research Centre (Ispra, Italy) 1977.

10. "MCNP - A General Monte Carlo Code for Neutron and Photon Transport," Los Alamos Scientific Laboratory report LA-7396-M (July 1978).

☆ U.S. Government Printing Office: 1979 — 677-013/27