CUMULATIVE FISSION YIELD OF $^{148}$Ce PRODUCED BY THERMAL-NEUTRON FISSION OF $^{235}$U

Submitted in partial fulfilment of the requirements of the Master's Degree of Nuclear Engineering at the Department of Electrical Engineering, College of Engineering, King Saud University, Riyadh, Saudi Arabia

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Rabi Awwal - 1405
December - 1984
CUMULATIVE FISSION YIELD OF $^{148}\text{Ce}$ in $^{235}\text{U}(n_{\text{th}},f)$

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This thesis was discussed on 19/03/1405
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تم في هذا المشروع قياس النسبة التراكمية للسيريوم 148 ذات العمق المنفتي 56 ثانية وبعض النظائر الأخرى ذات العمر المنفتي القصير وذلك باستخدام طريقة التحليل الطيفي (سبكتروسكوبية) للأشعة بواسطة جهاز كشف الجيرمانيوم ليثيوم (6،11) وتم تحمية القياسات في المفاعل النووي بجامعة تنجيرو وتم بعد تجميع عينات بورانيوم 235 (3 - 4 مغ) ذات التشعيب العالي (519٪) وتم الحصول على نسبات أطياف للأشعة الناتجة عن الانحلال الإشعاعي لنواتج الانشطار ذات العمر المنفتي بين 229 و 239 ثانية بعد تجميع لمدة 10 ثواني في المفاعل المذكور وتم تحديد أثاثة جاما في الطيف للعناصر بواسطة طاقة الأشعة والعمر المنفتي وأمكن بعد ذلك إيجاد النسبة التراكمية بعد حساب النشاط الإشعاعي وتحقيق كفاءة الكشف ونسبة التفرع لأشعة الجاما للعنصر ثم بالقسمة على محدد الانشطارات الحالية في العينة علما بأن سرعة الانشطار تم قياسها بواسطة طريقة حساب مساهمات الانشطار.

ولقد تم قياس النسبة التراكمية لكل من العناصر الآتية: السيريوم 148 الكربون 12، الأيونين 139، واللانشاتوم 144 والكريبتون 89 والزيتيدون (Crouch) في النتائج ثم قبعت بنتائج كروش (Crouch) والسيريوم 137 و136 و140 و135 و134 و133 و132 و131 و130 و129 و128 و127 و126 و125 و124 و123 و122 و121 و120 و119 و118 و117 و116 و115 و114 و113 و112 و111 و110 و109 و108 و107 و106 و105 و104 و103 و102 و101 و100 و99 و98 و97 و96 و95 و94 و93 و92 و91 و90 و89 و88 و87 و86 و85 و84 و83 و82 و81 و80 و79 و78 و77 و76 و75 و74 و73 و72 و71 و70 و69 و68 و67 و66 و65 و64 و63 و62 و61 و60 و59 و58 و57 و56 و55 و54 و53 و52 و51 و50 و49 و48 و47 و46 و45 و44 و43 و42 و41 و40 و39 و38 و37 و36 و35 و34 و33 و32 و31 و30 و29 و28 و27 و26 و25 و24 و23 و22 و21 و20 و19 و18 و17 و16 و15 و14 و13 و12 و11 و10 و9 و8 و7 و6 و5 و4 و3 و2 و1 و0.

ومن المفيد أن نشوه أن قياس نسبة السيريوم 148 تم للمرة الأولى وذلك بعد استقاء المعلومات التي نشرت قليلا عن مخطط الانحلال الإشعاعي لهذا العنصر.
ABSTRACT

CUMULATIVE FISSION YIELD OF $^{148}$Ce PRODUCED BY THERMAL-NEUTRON FISSION OF $^{235}$U

The cumulative fission yield of $^{148}$Ce and some other fission products produced by thermal-neutron fission of $^{235}$U and having half-life of 56 sec for $^{148}$Ce and between (32.3 sec) and (229 sec) for the others have been determined using Ge(Li) spectroscopic methods. The measurements were done at Tsing-Hua Open-pool Reactor (THOR) using 3 ~ 4 mg 93.15% enriched $^{235}$U samples. Spectra for 9 gamma rays emitted in the decay of the fission products after a 10 sec irradiation were obtained. Gamma rays were assigned to the responsible fission products by matching gamma-ray energies and half lives. Cumulative fission yields were then obtained from the data, after determining the appropriate gamma-ray activity after irradiation, correcting for detector efficiency and gamma-ray branching ratio and finally dividing by the number of fissions created in the sample. Fission rate was determined by fission track method.
The cumulative fission yields of $^{148}$Ce, $^{90}$Kr, $^{136mI}$I, $^{144}$La, $^{89}$Kr, $^{136}$Xe, $^{137}$Xe and $^{140}$Cs were determined and compared with previous evaluation by Crouch and showed good agreement. Uncertainties assigned to the present results range between 4 and 31.6%.

This first time measurement of the cumulation fission yield of $^{148}$Ce was possible due to the recent published information of the decay scheme for $^{148}$Ce.
### TABLE OF CONTENTS

<table>
<thead>
<tr>
<th>Section</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>ABSTRACT (ARABIC)</td>
<td>iii</td>
</tr>
<tr>
<td>ABSTRACT (ENGLISH)</td>
<td>iv</td>
</tr>
<tr>
<td>TABLE OF CONTENTS</td>
<td>vi</td>
</tr>
<tr>
<td>LIST OF TABLES</td>
<td>ix</td>
</tr>
<tr>
<td>LIST OF FIGURES</td>
<td>x</td>
</tr>
<tr>
<td>ACKNOWLEDGEMENT</td>
<td>xi</td>
</tr>
<tr>
<td>CHAPTER 1 INTRODUCTION</td>
<td>1</td>
</tr>
<tr>
<td>CHAPTER 2 EXPERIMENTAL FACILITIES AND MEASUREMENTS</td>
<td>5</td>
</tr>
<tr>
<td>2.1 Description of &quot;THOR&quot; Reactor</td>
<td>5</td>
</tr>
<tr>
<td>2.2 Description of Measurement System</td>
<td>10</td>
</tr>
<tr>
<td>2.2.1 Description of Pneumatic Tube System</td>
<td>11</td>
</tr>
<tr>
<td>2.2.2 Description of Ge(Li) Detector</td>
<td>13</td>
</tr>
<tr>
<td>2.2.3 Description of Multichannel Analyzer (TN-1710)</td>
<td>16</td>
</tr>
<tr>
<td>2.3 Target Preparation</td>
<td>18</td>
</tr>
</tbody>
</table>
2.4 Nuclear Fission Track Method of Measurement

2.4.1 Track Formation Mechanism
2.4.2 Track Registration Efficiency
2.4.3 Track Counting Methods

2.5 Experimental Procedures

2.5.1 Apparatus Set-up
2.5.2 Energy Calibration
2.5.3 Efficiency Calibration
2.5.4 Irradiation Process
2.5.5 Data collection

CHAPTER 3 DATA ANALYSIS

3.1 Decay and Mass Correction
3.2 Irradiation Time Determination
3.3 Fission Rate Measurements
3.4 Data Analysis Using a Computer Program

CHAPTER 4 DISCUSSION

4.1 Comparison with Previous Work
4.2 Conclusion and Remarks
### APPENDICES:

<table>
<thead>
<tr>
<th>Appendix</th>
<th>Title</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>Appendix-A</td>
<td>Data Analysis Code</td>
<td>64</td>
</tr>
<tr>
<td>A-1</td>
<td>The Program Yield Print-out.</td>
<td>64</td>
</tr>
<tr>
<td>A-2</td>
<td>The Program Yield Output Results.</td>
<td>68</td>
</tr>
<tr>
<td>Appendix-B</td>
<td>Growth and Decay of Fission Products.</td>
<td>75</td>
</tr>
<tr>
<td>Appendix-c</td>
<td>Case of 218 KeV Data (doublet).</td>
<td>82</td>
</tr>
<tr>
<td>Appendix-D</td>
<td>Figures of the Decay of the Measured Isotopes.</td>
<td>85</td>
</tr>
</tbody>
</table>

### REFERENCES

93
<table>
<thead>
<tr>
<th>Table</th>
<th>Description</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.</td>
<td>Classification of fission products by half-life.</td>
<td>1</td>
</tr>
<tr>
<td>2.</td>
<td>Number of experimental IFY for different isotopes.</td>
<td>2</td>
</tr>
<tr>
<td>3.</td>
<td>Radionuclides used for efficiency calibration.</td>
<td>34</td>
</tr>
<tr>
<td>4.</td>
<td>Data for calculation of absolute efficiency.</td>
<td>36</td>
</tr>
<tr>
<td>5.</td>
<td>Number of counted tracks per detector, 15sec exposure time.</td>
<td>43</td>
</tr>
<tr>
<td>6.</td>
<td>Average counted tracks per detector with different exposure time.</td>
<td>43</td>
</tr>
<tr>
<td>7.</td>
<td>Main data to evaluate the CFY for different fission products.</td>
<td>55</td>
</tr>
<tr>
<td>8.</td>
<td>Cumulative fission yields for short-lived fission products produced in $^{235}$U $\left(n_{th,f}\right)$ and comparisons to other works.</td>
<td>59</td>
</tr>
</tbody>
</table>
# LIST OF FIGURES

<table>
<thead>
<tr>
<th>Figure</th>
<th>Description</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.</td>
<td>Core assembly array</td>
<td>8</td>
</tr>
<tr>
<td>2.</td>
<td>Experimental facilities in high power section</td>
<td>9</td>
</tr>
<tr>
<td>3.</td>
<td>Pneumatic tube system</td>
<td>12</td>
</tr>
<tr>
<td>4.</td>
<td>Basic configuration of a lithium drifted p-i-n junction</td>
<td>14</td>
</tr>
<tr>
<td>5.</td>
<td>Typical Ge(Li) detector within its vacuum cryostat</td>
<td>15</td>
</tr>
<tr>
<td>6.</td>
<td>Configuration of track formation and etching</td>
<td>24</td>
</tr>
<tr>
<td>7.</td>
<td>Block diagram of the experimental set-up</td>
<td>28</td>
</tr>
<tr>
<td>8.</td>
<td>Typical calibration curve for a detector</td>
<td>30</td>
</tr>
<tr>
<td>9.</td>
<td>Efficiency calibration for the detector</td>
<td>38</td>
</tr>
<tr>
<td>10.</td>
<td>Timing and Irradiation schedule for the experiment</td>
<td>40</td>
</tr>
<tr>
<td>11.</td>
<td>Exposure time vs track number</td>
<td>48</td>
</tr>
<tr>
<td>12.</td>
<td>Successive radioactive decay process</td>
<td>75</td>
</tr>
</tbody>
</table>
ACKNOWLEDGEMENT

The author wishes to express deep gratitude toward Prof. Dr. Sümer Şahin, King Saud University, for his instructions and encouragement and also to Prof. Dr. Chien-Chung, National Tsing Hua University, for both his instructions during the whole thesis work especially the experimental part and his encouragement. Deep gratitude goes also to Mr. L.J. Yuan, head of reactor group for his many helpful discussions and suggestions during this work, and to Mr. C.L. Tseng for his valuable help during the experimental work.

Thanks is also to Institute of Nuclear Science, National Tsing Hua University for providing the experimental facilities. Great Appreciation is also to Saudi Arabian National Center for Science and Technology for their great encouragement and financial support.

Thanks also to graduate students, Mr. J.K. Pan and Mr. K.P. Chen for their much help in calculational work and information support, and also to Mr. Muneer for typing this thesis and Mr. Rana for preparing the graphs.
Since the discovery of fission about 40 years ago, thousands of papers have appeared about the physics of fission process. Despite many efforts made by experimenters in the past, there still exist wide blank areas, such as the precise information about the cumulative fission product yield (CFY) and independent fission product yield (IFY) data. One of the most decisive and theoretically important but least understood phases in fission is the rapid descent from saddle to scission points. Investigation of IFY and CFY is one of the approaches to elucidate the true evolution.

There are about 500 fission products with half-life $T_{1/2} > 0.5$ sec, which we can classify in table (1) and table (2).

**Table 1: Classification of Fission Products by Half-life**

<table>
<thead>
<tr>
<th>Half-life Range (sec)</th>
<th>Number of Fission Products</th>
</tr>
</thead>
<tbody>
<tr>
<td>Stable</td>
<td>28</td>
</tr>
<tr>
<td>$T &gt; 100$</td>
<td>145</td>
</tr>
<tr>
<td>$1 &lt; T &lt; 100$</td>
<td>161</td>
</tr>
<tr>
<td>$0.5 &lt; T &lt; 1$</td>
<td>$\sim 170$</td>
</tr>
<tr>
<td>Total</td>
<td>$\sim 500$</td>
</tr>
</tbody>
</table>
Table 2: (n,f) Experimental Data

<table>
<thead>
<tr>
<th>Fission Reaction</th>
<th>Experimental Data</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>No. of IFY</td>
</tr>
<tr>
<td>U-235 (n_{th},f)</td>
<td>239</td>
</tr>
<tr>
<td>U-233 (n_{th},f)</td>
<td>68</td>
</tr>
<tr>
<td>Pu-239 (n_{th},f)</td>
<td>70</td>
</tr>
<tr>
<td>U-238 (n_{R},f)</td>
<td>36</td>
</tr>
<tr>
<td>Th-232 (n_{R},f)</td>
<td>25</td>
</tr>
</tbody>
</table>

It is apparent from these tables that even for thermal induced fission of $^{235}$U which is the subject of our investigations the lack of reliable data of short-lived fission products ($T < 100$ sec) is stringent.

Data of IFY and CFY are the basic input to nuclear data file which in turn serves as data bases for reactor safety, fuel cycle physics, and decay heat calculations. Systematic predictions have been developed to bridge existing blank areas or to allow an extrapolation to unknown regions. Results of the systematics, however, could yield a dangerous and misleading information to end users if not checked by experimental measurements. A typical example is demonstrated in a recent paper\(^{(1)}\) the experimental IFY of ($^{148}$Cs) in $^{235}$U (n_{th},f) fission is a
factor of 100 higher than those predicted by adopted systematics. Thus an ample room still exist in the fission product yield studies using reactor neutron.

Until recently, measurements of CFY were hampered by the limited knowledge of absolute gamma branchings in the fission product decay scheme. With rapid expansion of knowledge on fission product decay scheme, it became possible only recently to identify CFY without radio chemical separation using large solid-state detectors as done at Oak Ridge by J.K. Dickens, and coworkers\(^2\).

Recently, a series of decay schemes among fission products were published by Dr. Chien, Chung and co-workers. One of them is the decay scheme of \(^{148}\text{Ce}\) with half-life of 56 sec\(^3\). This in turn encouraged this study using nuclear spectroscopy method.

The overall objective of this project has been to measure the cumulative fission yield of \(^{148}\text{Ce}\) (\(T/2 = 56\) sec.) and some other short-lived fission products, then to compare with the empirical predicted values by Crouch\(^4\). The study has been done for fission yield measurement for thermal neutron induced fission events. The target material was highly purified 93.15% enriched \(^{235}\text{U}\) metal foils of average weights of 3 ~ 4 mg.
The whole experiment and study were done at 1 MW Tsing-Hua Open-pool Reactor (THOR) with peripheral equipments. The targets were irradiated using pneumatic tube system at thermal flux of about $2 \times 10^9$ n/cm$^2$sec and reactor power at 500 watts. Different set-ups had been tried and developed to find the best possible counting system, which concluded with the counting system available in Counting Room A at Radioisotope Laboratories at National Tsing-Hua University. Data in the present work were obtained from three different irradiations of three different samples of $^{235}\text{U}$ with same enrichment. Using a 50 cm$^3$ Ge(Li) detector the gamma-ray spectra for each sample were collected. A computer program was developed to process the experimental data to evaluate the cumulative fission yields. Finally a conclusion was given in chapter 4.
CHAPTER 2

EXPERIMENTAL FACILITIES AND MEASUREMENTS

The experimental part of this thesis work was done at "THOR" reactor. The irradiation process was done through the pneumatic tube system in the reactor building, where the counting of each irradiated sample was performed at counting room "A" in the radioisotope building.

In the following sections we will describe (a) the reactor used for irradiation "THOR", (b) Pneumatic tube system, (c) Ge(Li) detector and (d) The TN-1710 multichannel analyzer.

2.1 Description of "THOR" Reactor(5):

Tsing-Hua Open-pool Reactor (THOR) is a heterogeneous, water moderated and graphite reflected reactor, fueled with 20% enriched rod-type and 93% enriched plate-type fuels, with total weight of fuel of about 18.86 Kg containing \(^{235}\text{U}\) of about 6.40 Kg (\(= 33.93\%\)).

The reactor pool is composed of a high power section, a low power section and a fuel storage section.
1- The high power section is approximately 8.5 ft in diameter and 34.5 ft deep, with walls and floor pierced by experimental facilities and coolant pipes.

2- The low power section of the pool is rectangular in plan, approximately 19 ft long, by 12 ft wide by 29.5 ft deep. Refueling and other service operations are performed in this section.

3- The fuel storage section is the same width and depth as the low power section from which it is separated by a submerged wall, 4 ft thick and 12 ft high. Spent fuel elements are stored in racks supported on mounting brackets located in this section.

The reactor arrangement is based on a nine by seven array of modules, each of 3" (7.62 cm) square. Core elements are supported and enclosed in a grid box on four sides. The central core region consists of 28 fuel elements formed a 5 x 6 array, with only three fuel elements in the 6th row. Each fuel element comprises two aluminum side plates and ten equally spaced flat fuel plates. The fuel plates are uranium-aluminum alloy, sandwiched between aluminum cladding on each side. When assembled in the fuel element, the plates are separated by 0.20"
(0.508 cm) gap for water passage. Each fuel plate is 0.099" (0.251 cm) thick, 2.79" (7.09 cm) wide and 25" (63.5 cm) long in size.

The reflector elements surround the central core region, each contains graphite in a 3 inch square aluminum can. The addition of aluminum cased graphite reflector elements together with a neutron source enlarges the core to 7 x 8 array. The control plates are located between the first and second row and the fourth and fifth row of fuel elements. Five radiation baskets fill the ninth row of the array. A schematic diagram of the core arrangement is shown in Figure (1).

The pool enclosure is a reinforced concrete structure which provides biological shielding against radiation.

The experimental facilities make the radiation produced by the reactor available for experimental work without jeopardizing the safety of operating personnel. As shown in Figure (2), experimental facilities provided in high power section of the pool include a thermal column, two 8-inch and four 6-inch beam ports, one 6-inch through port, and one 2-inch pneumatic tube. Twenty eight radiation baskets, 3-inch square, are available for use in the core. The fuel storage section of the pool may be
1st generation plate type (93%) (Metal)

2nd " " " (93%) (Metal)

3rd " Rod Type (20%) (UO₂)

Graphite plate for Moderate Neutron Source.

Total weight of fuel 18.86 Kg.

Figure 1. Core Assembly Array for "THOR"
FIGURE 2  TYPICAL OPEN POOL REACTOR EXPERIMENTAL FACILITIES.
used as a gamma-radiation facility as spent fuel elements become available.

Many types of experiments can be performed with open-pool reactor. These include applications in the fields of physics-diffraction, neutron optics, solid state physics, reactor physics, radiation effects, production of tracers, biomedical researchs, food sterilization, large equipment irradiation, agriculture studies, nuclear chemistry, radiation chemistry, education & engineering training.

2.2 Description of the Measurement System:

During experimental work the measurements were completed after irradiating the samples through the pneumatic tube system and then gamma-ray spectra were collected using a 50 cm$^3$ Ge(Li) detector. Analysis of the spectra was accomplished through TN-1710 multi channel analyser. Final results of cumulative fission yields were evaluated by using a computer code constructed for this purpose. In this section descriptions are given for the major parts of the measuring system.
2.2.1 Description of Pneumatic Tube System(5):

As our irradiation work had been performed at this experimental system of "THOR", a brief description of the system is required. Other experimental facilities are left for readers as their interest from the references.

The pneumatic tube system, as shown in Figure (3) is provided for rapid movement of small experimental specimens to and from the high flux region adjacent to the core in the high power section of the pool adjacent to the reactor core. The system includes a sending-receiving station in the laboratory, and a reactor terminal in the pool. An electric motor-driven turbo exhauster provides the driving force for the tube. The transit tube between the face of the shield and the reactor consists of a pair of concrete aluminum tubes. The 2.5" inner-tube guides the 1.5" diameter, 6-inch long specimen carrier. The 3" outer tube is connected through suitable solenoid air-control valves to the exhauster. The maximum carrier speed is 50 ft/sec. with a maximum load of two pounds. The speed can be lowered by reducing the vacuum in the system. The air from the system is discharged from the exhaust pump into an exhaust duct.
Figure 3. PNEUMATIC TUBE SYSTEM
Because of the uncertainty of the carrier speed, the real time traveling for the carrier from starting point to reactor core was experimentally determined by using fissiontrack method. More details and calculations will be given in chapter 3.

2.2.2 Description of Ge(Li) Detector

Ge(Li) detector is a lithium drifted germanium detector, Figure (4) shows the basic configuration of a lithium drifted p-i-n junction detector. The germanium is cooled to liquid nitrogen temperature to fix the lithium in position. These lithium atoms perfectly compensate the excess acceptor concentration, producing a depletion layer essentially equivalent to intrinsic germanium, and the resulting devices are sometimes referred as p-i-n detector.

Figure (5) shows the diagram for a typical Ge(Li) detector, showing the location of the Ge(Li) detector within its vacuum cryostat.

One of the advantages of Ge(Li) detectors is that they give very good energy resolution [small values of full-peak width at Half Maximum (FWHM)]. Another asset is
Figure 4. Basic Configuration of a lithium drifted p-i-n junction.
Figure 5. Typical Ge (Li) Detector within its Vacuum Cryostat.
their linearity over wide energy ranges\(^{(6)}\). Careful measurement with Ge(Li) detector have demonstrated a linearity of better than 46 ppm between 0.5 - 10 MeV gamma ray energy \(^{(7)}\). Detectors with equal efficiency will result in equal areas under the peak, but those with good energy resolution produce a narrow but tall peak which may then rise above the statistical noise of the continuum\(^{(8)}\). The combination of good linearity and high resolution makes these detectors into excellent spectrometers when they are combined with appropriate electronic instrumentation \(^{(6)}\).

The germanium lithium detectors are clearly preferred for the analysis of complex gamma ray spectra involving many energies and peaks\(^{(8)}\).

For this experiment a 50 cm\(^3\) Princeton Gamma Tech Ge(Li) detector was used, with energy resolution of 3.1 keV at 1332 keV for a \(^{60}\)Co source. Energy calibration and efficiency calibration were performed directly before the irradiation process, where detailed description will be in another section of this thesis.

2.2.3 Description of the Multi Channel Analyzer(TN-1710)\(^{(9)}\):
A multi-channel analyzer (MCA) has many functions for many kinds of analysis, for example: pulse height analysis (PHA) and multi channel scaling (MCS).
The MCA (TN1710) has four selection positions: The 1/1 position selects full memory; 1/2 selects the first half of the memory, while 2/2 selects the second half; 1/4 specifies the first quarter, 2/4 the second, 3/4 the third and 4/4 the fourth quarter of memory.

The data memory is available in configuration of 1024, 2048, 4096 and 8192 channels, each channel holding 1,048,576 counts.

The TN-1710 allows for selection, setup, display and output up to fifty regions of interest. Each region of interest can be integrated (area under curve) and gross count and net count may be memorized and driven as an output as required.

Counting dead time is internally corrected by the TN-1710 MCA. Clock time indicates the real passed time, where live time compensates the detection dead time. So, no dead time correction by the user is required.
2.3 Target Preparation:

Although target or sample preparation for reactor irradiation is generally straightforward, some special considerations do enter\((6)\). For example, containers for samples to be exposed in high-flux reactors have to be carefully chosen, with due regard to neutron flux, ambient temperature, and length of irradiation. A high flux of neutrons and gamma rays causes decomposition of many substances, and in many cases there are valency charges\((10)\). Boron contained vessels should be avoided because of their high neutron-capture cross section. For irradiations of order of minutes or less in modest fluxes of many research reactors like "THOR", plastic vials (PVC) are often satisfactory, and they had given rise to rather low activity levels. Aluminum wrappers are not advised for this work because they need to be cooled (half-life of $^{28}\text{Al} = 2.3$ min) and it is rather long compared with half-life of $^{148}\text{Ce}$ which is 56 second.

Personnel exposure and contamination hazards should be also concerned. A survey meter and consulting of health physicist are required for safety precautions. Thermal stability of the substance to be irradiated must be considered. For this work the irradiated material $^{235}\text{U}$ metal foil is sealed in P.V.C. bags and then put
into irradiation vessels to prevent releasing of fission fragments. Irradiation of aqueous solutions creates special problem like decomposition of water which will result in dangerous pressure problems. Self-shielding is another kind of problem. To prevent this, or at least to minimize it, the irradiated materials are tried to be made as thin as possible.

The technique for preparation of a target depends on many factors: thickness; cost; target backing; size required and other many factors.

Usually an irradiated target needs to be chemically separated to prepare a known reaction product free from other radioactive contaminants or even other inactive impurities. But with use of the high-resolution Ge(Li) gamma-ray spectroscopy makes possible the analysis of even very complex mixtures without applying the time consuming chemical processes.

Because of rapid expansion of knowledge of fission product decay scheme, as mentioned before, it became possible only recently to identify cumulative fission yield without radio-chemical separation.
The target material in this work was uranium-235. During different stages of this work, both natural and enriched, powder form and foils were used.

Natural uranium in powder form of weight \( \approx 15 \) mg sealed in PVC bags was firstly tested, but they gave no clear distinguishable data and peak. So rejected.

Enriched powder form of weight \( \approx 0.1 \) mg sealed in PVC bags was used, but rejected too because of the high counting rate resulted which caused 100% dead time to the counting system. Cooling was required, but the short half life of \(^{148}\text{Ce} \) \( (T_{1/2} = 56 \) sec.\) did not allow that. The above two cases were irradiated under reactor full power of 0.75 Mwatt for \( \approx 10 \) sec.

Because of availability of metal foils of highly enriched uranium-235 (93.15%), this work was completed by using those metal form foils, which were easy to handle easy to seal and showed less health hazard. And by changing some other factors reasonable data were observed and collected.

Each sample of uranium foil was weighted by using an analytical balance and permanently sealed in PVC bags under inspection and supervision of the health physicist. A fission track detector is also sealed together with each sample of uranium foil for later use of calculations. Detailed track method is described in section 2.4.
During irradiation, the sealed foils were put into rabbit tube (carriers) and sent to reactor core for irradiation. Before counting the foils were again pulled out from the rabbit tube in a controlled hood and put in front of the detector for counting. After measurements were completed, the spent foils were put at a controlled shielded place for cooling down and later handled as low level nuclear waste by health physicist.

2.4 Nuclear Fission Track Method of Measurement

If a sheet of mica, covered with uranium, is irradiated in a reactor core, fission fragments will traverse and produce radiation damage in it. When this piece of mica is immersed in a suitable reagent such as hydro fluoric acid (HF), the tracks of these fission fragments are very selectively attacked, fine hollow channels are formed along the particle paths while the rest of a material is almost untouched.

Studies of this effect in various materials, i.e. glass, mica, lexan had been tried in another thesis work\(^{(11)}\) done with the same experimental facilities at "THOR". Mica was proved to be the easiest for recognition of the fission tracks in it. Glass showed high background tracks and lexan proved not easy to be recognized.
In general, since the fission track density is directly proportional to the total fission reactions, fission rate and travelling time of specimen carrier to and from reactor core are easily and inexpensively determined.

2.4.1 Track Formation Mechanism

There are two main mechanisms to explain the formation of etchable tracks in dielectric materials:

a) Ion Explosion Spike\(^{(12)}\): This theory is applicable to inorganic crystals. In this model a positive region is produced along the path of heavy charged particles due to ejection of electrons, and atomic displacement in the crystal lattice happens due to mutual repulsion of remaining heavy positive ions. This disordered imperfect region becomes more easy to be etched.

b) Radiochemical Damage Mechanism\(^{(13)}\): This theory is applicable to organic polymers. It states that the tracks are formed by radiolytic scission of long polymer chains into shorter fragments and products are more easily etched than the other undamaged bulk plastic.

There are some other mechanisms but play little role, such as thermal spike, displacement spike, or ionization spike.
2.4.2 Track Registration Efficiency: (14)

Tracks may be formed in almost any sort of insulating material, such as glass, mica, lixan and polymers, but have not been seen in good conductors.

The fission fragments from the fissionable material attached to the track detector is considered to be isotropic one. So, an assumption was made that half of the fragments emit toward the hemisphere in $2\pi$ solid angle extended from the point source. Hence fission fragments strike the detector surface at different angles from $0^\circ$ to $90^\circ$ with respect to the normal to the surface. But not all tracks resulted can be etched or read by optical microscope. It depends on the detector material as well as the chemical solution. Etching process is uniform at the same time rapid for the damaged regions. But only tracks emitting a surface at angle greater than some critical value may be observed. If the detector material dissolved at a rate $G$, the attack at a more rapid rate $T$, the behaviour may be shown in Figure (6), then for each detector material exists a critical angle $\theta_c$, given by

$$\theta_c = \sin^{-1} \frac{G}{T} \quad \ldots \quad (1)$$

If the general attack moves normal to the original surface more rapidly than the attack along the track, the track is erased and can not be observed.
Figure 6 Configuration of Track formation & Etching.
For the case of mica, the fresh cleavage planes are not attacked by the etching reagent, i.e. hydro flouric acid (HF), hence tracks, at all angles should be revealed, the critical angle is thus considered as zero.

Track formation is affected by the choice of etchant, its concentration, etching temperature, and etching time\(^\text{(15)}\). For our case HF was the most effective reagent for etching tracks. Mica detectors were etched in 15% HF for one hour at 70°C. These conditions were chosen because they cause tracks large enough to identify but small enough not to be overlap appreciably\(^\text{(11)}\).

When the etching process was completed, fission tracks were counted through an optical microscope with 500 magnification for all mica detectors.

2.4.3 Track Counting Methods:
Counting techniques of the tracks became more and more advanced and easy using the automated systems, which reduces the time and the manpower.

For example an X-Y desitometer can be used to read the number of the tracks and then fed to a computer. The method is done by obtaining an image of each detector in a screen coupled to the computer and the spot track density for each detector is counted by computer\(^\text{(16)}\).
Standard image analysis systems, which are widely used in many fields, are commercially available for computerized measurements of nuclear tracks, but they are rather expensive and unflexible\(^{(17)}\).

To overcome this problem, two inexpensive microprocessor based systems were developed\(^{(17)}\), with different resolution. A real time video digitizer was used to digitize the video pictures of particles seen through a microscope and the picture analysis was done by software. The microscopes are equipped with stages driven by stepping motors, which are controlled by separate microprocessors. A PDP 11/03 supervises the operation of all microprocessors and stores the measured data on its mass storage devices.

In Tsing-Hua University Laboratories, a similar device is now under developing, but not yet finished the time this work was completed. This encouraged doing the countings manually by using a video camera and a T.V. screen. Each screen represents a random spot on the detectors and the whole screen was counted as one reading.
2.5 **Experimental Procedure:**

2.5.1 **Apparatus set-up**

With rapid expansion of knowledge on fission product decay schemes, it became possible to identify CFY without radiochemical separation using solid state detectors, i.e. Ge(Li) detector in our case, without involving too much with complicated timing and electronics.

A simple counting system set-up was used, shown by the block diagram in Figure (7).

**Apparatus used:**

1. High voltage power supply (AEL-5000B Power Design Inc).
2. Ge(Li) detector (50 cm$^3$ Princeton Gamma Tech).
3. Pre-amplifier
4. Linear amplifier (CANBERRA model 1413).
5. Multi channel analyzer (TN-1710).
6. Teletype.
Figure 7  Block diagram of the experimental setup.

1 High voltage power supply AEL-5000 B power Design Inc.
2 Ge (Li) detector 50 cc. Princeton gamma Tech.
3 Linear Amplifier Canberra model 1413
4 Multichannel Analyzer TN-1710
2.5.2. Energy calibration

In gamma ray spectroscopy with Ge(Li) detectors, the pulse height scale must be calibrated in terms of absolute gamma ray energy if various peaks in the spectrum are to be properly identified.

Calibration should be done using some known gamma ray sources which gives peaks of known energy and intensity in the spectrum, preferably standard sources with gamma energies that are not widely different from those to be measured in the known spectrum. Linearity of response with energy is not to be taken for granted, although highly desirable and calibration with several sources over the entire energy range of interest is therefore available\(^{(6)}\). The precision to which the centroid of a peak in a Ge(Li) spectrum can be localized is dependent on the spectrometer system resolution and its stability over the period of the measurement\(^{(8)}\). When all the energy calibration points have been established over the whole range of interest, a calibration curve relating energy to channel number is normally derived by using least square fitting. A typical calibration curve is shown in Figure (8).

Internally built calibration system is also possible like in our case the TN-1710 MCA, where only two calibration
Figure 8 Typical Calibration Curve for a Detector
values were required, a lower energy value and a upper energy value. Calibration procedure was as follows:

1. Connect the preamplifier of Ge(Li) detector to the input of the amplifier module which in turn connect output to the ADC input.

2. Find the required suitable gain, lower level discrimination, upper level discrimination and adjust the zero level of ADC. The procedures of step 2 are clearly described in our reference: (THE OPERATION MANUAL FOR TN-1710 MCA).

3. Choose two standard sources which were in our case, Cs-137 (gamma peak approximately 662 KeV) and Co-60 (upper gamma peak at approximately 1332 KeV) and place them at a fixed reproduceable geometrical position. Since in high precision measurements, geometry and position of source to detector are very important, because shifts up to 110 eV have been reported in aparent peak position as the source of gamma rays was moved around the detector(8).

4. Press AQUIRE, when the spectrum is clearly defined, insert the value of low energy (662 KeV) at the channel number where the low energy peak is located. Similarly, insert the value of the high energy (1332
Similarly, insert the value of the high energy (1332 KeV) at the channel number where the high energy peak is located. These calibration values are inserted by using the BUG/REGION switch, OMNI knob and the calibration switch.

5. Now the system is already calibrated. The energy of any channel located by the cursor position is simply printed on the CRT screen of the MCA.

2.5.3 Efficiency calibration

Measurement of absolute (and even of relative) intensities of radiations are appreciably more difficult than energy determination and requires the knowledge of the detector efficiency. In Ge(Li) gamma ray spectroscopy, an efficiency based on the area under the single or double escape peak is sometimes used in place of that based on the full-energy peak\(^8\). Users will normally carry out their own periodic efficiency calibration, although they can be estimated from published measurements or calculations for detectors of similar size, but the accuracy of results based on those values will not be better than 10 ~ 20 percent.

The calibration should be done by users using sources calibrated by some other means. Any error at that time
will be applied both to the calibration and actual measurement, and will not affect the accuracy of activity measurements.

The calibration is normally done to a range enough to cover all the range of interest to allow construction of an empirical efficiency versus energy curve. Table (3) lists radionuclides used for efficiency calibration, together with decay data necessary to compute gamma ray yields from absolute activity. $^{152}$Eu recently gained popularity because of its convenient half-life (13Y) and the wide range of gamma ray energies produced in its decay.

Once the efficiency of a detector has been measured at several energies using calibrated sources, it is useful to fit a curve to these points which describes the detector efficiency over the entire energy range.

Summing effect of two low energy peaks might be of importance but can be minimized by reducing the solid angle or the source-detector spacing. On the other hand this depends on the activity of the source. So, a compromise should be done between summing inaccuracies and statistical errors in the peak.

Other errors in peak area measurement can arise from inaccurate treatment of system dead time and from pulse
Table 3  Radionuclides used for efficiency calibration

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>$T_{1/2}$</th>
<th>$E$ [keV]</th>
<th>$I$ [%]</th>
<th>$\Delta I/I$ [%]</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{22}$Na</td>
<td>2.60 y</td>
<td>1274.5</td>
<td>99.95</td>
<td>0.0</td>
</tr>
<tr>
<td>$^{24}$Na</td>
<td>15.0 h</td>
<td>1368.5</td>
<td>100.0</td>
<td>0.0</td>
</tr>
<tr>
<td>$^{46}$Sc</td>
<td>83.7 d</td>
<td>889.2</td>
<td>99.98</td>
<td>0.0</td>
</tr>
<tr>
<td>$^{54}$Mn</td>
<td>312.5 d</td>
<td>834.8</td>
<td>99.98</td>
<td>0.0</td>
</tr>
<tr>
<td>$^{57}$Co</td>
<td>272 d</td>
<td>14.4</td>
<td>9.6</td>
<td>1.0</td>
</tr>
<tr>
<td>$^{60}$Co</td>
<td>5.27 y</td>
<td>1173.2</td>
<td>99.88</td>
<td>0.0</td>
</tr>
<tr>
<td>$^{85}$Sr</td>
<td>64.8 d</td>
<td>13.4</td>
<td>50.7</td>
<td>1.5</td>
</tr>
<tr>
<td>$^{85}$Y</td>
<td>106.6 d</td>
<td>14.2</td>
<td>52.5</td>
<td>1.5</td>
</tr>
<tr>
<td>$^{95}$Nb</td>
<td>35.15 d</td>
<td>765.8</td>
<td>99.80</td>
<td>0.0</td>
</tr>
<tr>
<td>$^{113}$I</td>
<td>115.2 d</td>
<td>24.1</td>
<td>79.5</td>
<td>2.0</td>
</tr>
<tr>
<td>$^{131}$I</td>
<td>8.02 d</td>
<td>364.5</td>
<td>82.4</td>
<td>0.5</td>
</tr>
<tr>
<td>$^{134}$Cs</td>
<td>2.06 y</td>
<td>604.5</td>
<td>97.5</td>
<td>0.2</td>
</tr>
<tr>
<td>$^{137}$Cs</td>
<td>30.0 y</td>
<td>31.8/32.2</td>
<td>5.64</td>
<td>2.0</td>
</tr>
<tr>
<td>$^{139}$Ce</td>
<td>137.6 d</td>
<td>33.0/33.4</td>
<td>64.1</td>
<td>2.0</td>
</tr>
<tr>
<td>$^{141}$Ce</td>
<td>32.5 d</td>
<td>35.6/36.0</td>
<td>12.6</td>
<td>2.0</td>
</tr>
<tr>
<td>$^{140}$La</td>
<td>40.27 h</td>
<td>1596.6</td>
<td>95.6</td>
<td>0.3</td>
</tr>
<tr>
<td>$^{198}$Au</td>
<td>2.696 d</td>
<td>411.8</td>
<td>95.53</td>
<td>0.1</td>
</tr>
<tr>
<td>$^{203}$Hg</td>
<td>46.6 d</td>
<td>70.8/72.9</td>
<td>10.1</td>
<td>1.5</td>
</tr>
<tr>
<td>$^{241}$Am</td>
<td>432 y</td>
<td>59.5</td>
<td>36.0</td>
<td>1.0</td>
</tr>
</tbody>
</table>

From Debertin, et al.37

$I$: Gamma ray photon yield per disintegration
$\Delta I/I$: Uncertainty in yield figure
pile-up effects, but both are important only when the pulse rates are relatively high. Although dead-time is already treated in our measuring device i.e. the TN-1710 MCA.

The full-energy of peak efficiency of Ge(Li) detectors can be quoted in several different ways. The quantity most often directly measured is the absolute efficiency, defined as the ratio of counts in the full-energy peak divided by the number of gamma rays emitted by the source. It can be represented by equation (2) if counting time $t_c$ is much shorter than the half-life of the sample:

$$\varepsilon = \frac{C}{I_\gamma A_0 e^{-\lambda t_d}}$$

... (2)

where $\varepsilon = \text{absolute efficiency}$

$C = \text{total counts}$

$I_\gamma = \text{Gamma ray branching ratio}$

$A_0 = \text{activity of source}$

$\lambda = \text{decay constant}$

$td = \text{decay time}$

Table (4) shows all the data for calculating the absolute efficiency and the calculated absolute efficiency. Using least square fitting (computer code) the
Table (4): Data for calculation of absolute efficiency

<table>
<thead>
<tr>
<th>$A_Z$</th>
<th>$E_\gamma$ (KeV)</th>
<th>$T_{1/2}(Y)$</th>
<th>$A_0$ ($\mu$Ci)</th>
<th>Total count</th>
<th>$T_c$(Sec)</th>
<th>$T_d$(Y)</th>
<th>$I_\gamma$</th>
<th>Absolute Efficiency</th>
</tr>
</thead>
<tbody>
<tr>
<td>133Ba</td>
<td>276</td>
<td>10.66</td>
<td>10.37</td>
<td>10743</td>
<td>500</td>
<td>6.942</td>
<td>0.07006</td>
<td>0.0132814</td>
</tr>
<tr>
<td></td>
<td>302</td>
<td></td>
<td></td>
<td>23890</td>
<td></td>
<td></td>
<td>0.18104</td>
<td>0.0114296</td>
</tr>
<tr>
<td></td>
<td>356</td>
<td></td>
<td></td>
<td>65752</td>
<td></td>
<td></td>
<td>0.62</td>
<td>9.18562x10^{-3}</td>
</tr>
<tr>
<td></td>
<td>383</td>
<td></td>
<td></td>
<td>8533</td>
<td></td>
<td></td>
<td>0.0899</td>
<td>8.22117x10^{-3}</td>
</tr>
<tr>
<td>60Co</td>
<td>1173</td>
<td>5.2719</td>
<td>1.11</td>
<td>13654</td>
<td>400</td>
<td>7.225</td>
<td>1</td>
<td>2.14838x10^{-3}</td>
</tr>
<tr>
<td></td>
<td>1332</td>
<td></td>
<td></td>
<td>11577</td>
<td></td>
<td></td>
<td>1</td>
<td>1.82158x10^{-3}</td>
</tr>
<tr>
<td>22Na</td>
<td>1257</td>
<td>2.6022</td>
<td>1.07</td>
<td>11349</td>
<td>100</td>
<td>7.208</td>
<td>0.9993</td>
<td>1.9559x10^{-3}</td>
</tr>
</tbody>
</table>
straight line part of the efficiency calibration curve can be found, since our energy interest falls under straight line region which is between 121 ~ 600 KeV, as shown in Figure (9).

The efficiency calibration curve is also represented in a formula form (eq. 3) later used in the constructed computer program for evaluating the cumulative fission yields. The equation is as below:

\[ E_{eff} = 10[1.07 - 1.22 \times \log(\text{Energy})] \quad (3) \]

with \(200 \text{ KeV} < E < 2000 \text{ KeV}\).

2.5.4 Irradiation Process

The sample to be irradiated which includes a 93.15% enriched \(^{235}\text{U}\) metal foil and a \(^{238}\text{U}\) (natural uranium) fission track detector in an aluminum wrapper, was sealed into the carrier and placed in the laboratory terminal. The controls were set, and the differential pressure created in the transit tube dispatched the carrier to reactor-core terminal. After the desired exposure time had elapsed, the air pressure in the transit tube was reversed, and the carrier was returned to the laboratory terminal.
Figure 9  Efficiency Calibration Curve for the Detector
After exposure, carrier was discharged directly from the laboratory terminal into shielded receptacles, where the carrier was reopened and transferred immediately to the sample holder in front of the detector for counting.

Since the available counting system with TN-1710 MCA could only store in 4 quarters (memories), the complete desired counting of 11 minutes was done using three different samples of weights $3 \sim 4 \text{ mg}$. The problem of difference in weights was solved later by multiplying with the normalization factor. Figure (10) shows the timing schedule of the three samples, from the moment of pressing the carrier sending button to the end of the counting process.
Figure 10  Timing & Irradiation Schedule for the Experiment
2.5.2 Data Collection

After each exposure the carriers are discharged into the shielded receptacles, then opened. Each exposed sample contains two parts: (a) the uranium-235 (93.15% enriched) metal foil and (b) the fission track detector (mica). The $^{235}$U foil is sent immediately for counting and the fission track detectors are sent to another laboratory for etching and preparing for counting.

(a) Data Collection for Uranium-235 Foils

As mentioned previously, three samples were used and irradiated at different time intervals. This was because of the limitations on the counting system used (TN-1710 MCA), since it was limited with maximum four data memories for spectra storing, each data memory containing 2048 channels.

After exposure the first sample was allowed to cool for 67 seconds (the time needed to open the carrier and place the sample foil in front of the detector) and then counted for 4 times with counting times of 23, 22, 21, and 21 seconds; the second sample was allowed to cool for 147 seconds and then counted for 4 times with
counting times of 43, 42, 42, and 41 seconds; where the third sample was allowed to cool for 307 seconds after exposure and then counted for 4 times with counting times of 82 seconds each.

This procedure was done to establish a complete describe of the decaying process for the fission products after the $^{235}\text{U}$ metal foil was being exposed to thermal neutron flux. Longer counting time was not necessary because all fission products we do concern at present work were short lived with maximum half-life of $T_{1/2} = 229$ seconds for $^{137}\text{Xe}$ ($E_\gamma = 455.5$ keV).

To relate the three sets of counting done with three different weighted samples, a normalization factor was introduced to be multiplied with the second and third group data and normalized to the first group data. The normalization factor is later described.

(b) Data Collection for the Fission Track Detectors:

After all exposures had been done, the fission track detectors were sent for etching, then they were counted under an optical microscope with 500 magnification together with a video camera to make the counting process easier. Different spots for each detector were
selected randomly and counted for several times to lead to good statistical information. Detector countings are shown in Table (5) and Table (6).

Table 5: Number of Counted tracks per detector with 15 sec exposure time.

<table>
<thead>
<tr>
<th>Track detector</th>
<th>First average count</th>
<th>Second average count</th>
<th>Total count</th>
<th>Count%</th>
</tr>
</thead>
<tbody>
<tr>
<td>T15A</td>
<td>106.4</td>
<td>99.8</td>
<td>1031</td>
<td>+ 3.1</td>
</tr>
<tr>
<td>T15B</td>
<td>115</td>
<td>116.8</td>
<td>1159</td>
<td>+ 2.9</td>
</tr>
<tr>
<td>T15C</td>
<td>101.6</td>
<td>104.6</td>
<td>1028</td>
<td>+ 3.1</td>
</tr>
</tbody>
</table>

Table 6: Average counted tracks per detector with different exposure time

<table>
<thead>
<tr>
<th>Track detector</th>
<th>Exposure time (sec)</th>
<th>Average count</th>
</tr>
</thead>
<tbody>
<tr>
<td>T 10</td>
<td>10</td>
<td>56.2</td>
</tr>
<tr>
<td>T 20</td>
<td>20</td>
<td>152.2</td>
</tr>
<tr>
<td>T 25</td>
<td>25</td>
<td>200.2</td>
</tr>
</tbody>
</table>
3.1 Decay and Mass Corrections:

As mentioned before, due to limitation of data memories in the counting system, three samples were used and related together by a normalization factor. It was difficult to cut identical weight of samples by simply using a cutting instrument. One reason was that the samples were very small metal foils in the order of milligrams, although they were easy to cut but in other hand the amount to be cut was very difficult to control. Another reason was too much cutting may result in cracking of the sample foil. That meant loosing of sample material, which was very precious (93.15% enriched $^{235}\text{U}$). A third reason was that too much handling of the sample and working on it may cause some health hazards since uranium is an (alpha emitter) from powder and fine fragments resulted from cutting and handling.

To solve this problem, similar sized samples were choosen and their countings were corrected in turn by introducing normalization factor. The weight for the three samples were 3.2, 3.6 and 4.3 milligrams respectively.
The activity after irradiation for time $t$ seconds was

$$\text{Activity} = N\phi \sigma (1-e^{-\lambda t}) \text{ dis/sec.}$$

$$w = \frac{-(\text{Av. No})\phi \sigma (1-e^{-\lambda t})}{A} \quad \ldots \quad (4)$$

where $N =$ number of atoms of the target

$\phi =$ irradiation flux, n/cm$^2$ sec.

$\sigma =$ cross-section for the formation of a certain product, cm$^2$.

$\lambda =$ decay constant of that product, sec$^{-1}$

$w =$ sample weight, gm

$A =$ atomic weight

$t =$ irradiation time sec

Thus:

$$\frac{\text{Activity 1}}{\text{Activity 2}} = \frac{w_1\phi_1}{w_2\phi_2} = \frac{C_1}{C_2} \quad \ldots \quad (5)$$

where $A_1 = A_2$ & $\phi_1 = \phi_2$

$C_1$ and $C_2$ were the counts.

Here $\phi_1$ and $\phi_2$ could not be assumed equal especially for high precision measurements, although the power level were at a constant value during the whole experiment. Because
fine oscillation of the flux occured in the core during the experiment which could not be avoided. So, the fission track detectors counting ratio were used as indicators of the flux ratios:

\[ \frac{\phi_1}{\phi_2} = \frac{T_1}{T_2} \quad \ldots \quad (6) \]

where \( T_1 \) & \( T_2 \) are track countings of first and second irradiations. These tracks were irradiated simultaneously with the samples in the same carrier at the same position. The fluxes subjected to the sample foil equals the fluxes subjected to fission track detectors.

Thus:

\[ \frac{C_1}{C_2} = \frac{w_1 T_1}{w_2 T_2} \quad \ldots \quad (7) \]

For second group:

\[ \frac{C_1}{C_2} = \frac{3.2\pm3.1\%}{3.6\pm2.8\%} \cdot \frac{1031\pm3.1\%}{1159\pm2.9\%} = 0.7907\pm5.96\% \]

For third group:

\[ \frac{C_1}{C_3} = \frac{3.2\pm3.1\%}{4.3\pm2.3\%} \cdot \frac{1031\pm3.1\%}{1028\pm3.1\%} = 0.7464\pm5.84\% \]

By multiplying these normalization factors with second and third group countings, they were normalised and referred to the first group data.
3.2 Irradiation time determination:

In the present work, the samples were irradiated at the pneumatic system of "THOR" facilities, the real exposure time could not be directly measured. Since the real travelling time of the sample carrier was not clearly given by the designers of this irradiation system. The only known information given in the user's manual was that the maximum speed of this carrier that could be reached was 50 ft/sec. And due to wear off of the blower, blower voltage variation and variation of the vacuum in the system, the carrier speed could not be calculated precisely from the informations given by the manual. The real exposure time must then be measured experimentally.

Since number of tracks, that were formed in the fission track detector, was related proportionally to the exposure time, several fission track detectors were used and exposed for different times: 10, 20 and 25 seconds to construct a relationship between the exposure time and number of tracks resulted. The results were shown in table (6), and plotted in figure (11). Using least square fitting by a computer code, a straight line relation was as following:

\[ Y = -39.4 + 9.58X \]  \hspace{1cm} (8)
Figure 11 Exposure Time vs Track Number
where \( Y \) the number of fission tracks recorded and \( X \) exposure time in seconds.

Solving equation (8) by substituting \( Y \) (number of fission tracks) with zero, then \( X \) (time) was found as 4.1 seconds, which was the time spent for the carrier to travel from sending station to reactor core terminal. The total time spent, from the moment the carrier left the sending station to the moment it left reactor core after irradiation was completed, was controlled by the experimenter which was 15 seconds. To find the exact exposure time:

\[
\text{Exposure time} = \text{total time} - \text{time of flight} \quad \cdots \quad (9)
\]

\[
= 15 - 4.1
\]

\[
= 10.9 \text{ sec}
\]

So, the samples were irradiated exactly for 10.9 seconds.

*from pushing the sending botton till pushing the sending back botton*
3.3 Fission Rate Measurement:

During the experiment flux was evaluated by fission track detector countings and some previously measured data. Actual flux at the time of irradiation could not be accurately adopted through control room readings.

In a previous normalization (calibration) experiment in thermal column between track detectors and activation foil detectors at a 30 hr run, the flux was evaluated as \(4.74 \times 10^6\) n/cm\(^2\)sec. A straight line relation was made with slope of \(1.826 \times 10^4\).

\[
\text{Slope } m_D = \frac{Y_D}{X_D} = \frac{\phi t}{X} \quad \ldots \ldots \quad (10)
\]

where \(\phi\) the flux, \(t\) irradiation time in seconds and \(X_D\) the number of tracks per grid (7788 tracks), where subscript \(D\) stands for depleted uranium fission track.

A normalization factor \((= 12.1)\) was introduced to relate the 93% enriched \(^{235}\text{U}\) data to the depleted uranium track detector data. The normalization factor was defined as the ratio between the concentration of 93% enriched \(^{235}\text{U}\) and the concentration of depleted \(^{235}\text{U}\).

Thus for our case \(Y = \phi t = mX = \frac{m_D X_D}{12.1} \quad \ldots \ldots \quad (11)\)
\[ \phi t = 1.826 \times 10^4 \times \frac{7788}{12.1} = 1.17528 \times 10^7 \text{n/cm}^2 \]

So,

\[ \phi = \frac{1.17528 \times 10^7 \times 3600}{30} = 1.41033 \times 10^9 \text{n/cm}^2 \text{ sec} \quad \ldots (12) \]

For thermal cross-section of $^{235}\text{U}$ in THOR, we should consider that the "thermal" neutron in THOR is a little hotter than 25°C, i.e. 34.2°C).

Correction was thus needed, were

\[ \sigma_{\text{THOR}} = \sigma_{\text{th}} \left( \frac{273 + t_{\text{THOR}}}{273 + t_{25}} \right)^{-1} \quad \ldots (13) \]

\[ = 580 \left( \frac{273 + 34.2}{273 + 25} \right)^{-1} \]

\[ = 571.2 \text{ barns} \]

Then from (12) and (13) fission rate ($\sigma \phi$) was determined as $8.05583 \times 10^{-13} \text{ fission/sec}$
3.4 Data Analysis Using a Computer Program:

In order to generalize the fission yield calculations, for future use, upto two parent isotopes were adopted in this program for the cases where the grand parent could not be simply eliminated due to its relatively long half-life. This lead to numerous calculational steps. Human error do highly interfere with this sort of calculations, beside the manpower and time consumed.

To solve this, a computer program in FORTRAN language was developed on a PDP/11 computer. The program can be devided to three major parts.

First part was the read-in part, where all different data for the parents and daughter were inserted, e.g. $T_{1/2}$, $\lambda$, IFY, ... etc. All other data were also inserted, like target weight, number of data points, error bars, ... etc.

Second part was the calculational part, where detector absolute efficiency, number of atoms for the target material, and mostly important, the cumulative fission yield and the corresponding error bar.

The third part was the print-out part, where all required results were printed out for reference.

Complete listing of this program and the results print out could be seen in the appendices.
CHAPTER 4

DISCUSSION

4.1 Comparison With Previous Work

The purpose of this experiment was to measure the cumulative fission yield of $^{148}$Ce and some other short-lived fission products created from thermal-neutron induced fission of $^{235}$U. Yield of $^{148}$Ce had been obtained for the first time experimentally. Total of nine short lived fission products were measured in this work with minimum half-life of 32.3 sec for $^{90}$Kr ($E_\gamma = 121.5$ KeV) and maximum half-life of 229 sec for $^{137}$Xe ($E_\gamma = 455.5$ KeV).

At the beginning this experiment was done using a natural uranium sample (~1 mg) and irradiated at reactor power of 0.75 MW, but this caused 20% dead time even at farthest source to detector distance (~40 cm). Then the experiment was tried near the reactor core using natural uranium sample (~10 mg) irradiated at reduced reactor power of 500 watt, this time the source (sample) was placed at a much shorter source to detector distance (15 cm) and only 0.2% dead time was registered. But the problem of high electronic noise and high counting background existed, which were later explained by the location of the Ge(Li) detector directly close to the
reactor pool area and also due to the length of signal
cable (~100 m) used to connect the detector to the ADC
and MCA in the counting room at a nearby building. It
was very difficult to construct a shield that time on
the cable to prevent the high frequencies interrupting the
signals sent from the detector received by the counting
room.

The work was finally decided to be completed in the
radioisotope laboratories of THOR low background
counting room "A" using a 50 cm$^3$ Ge(Li) detector together
with TN-1710 multi channel analyzer.

The sample adopted was 93.15% enriched $^{235}\text{U}$ metal foils
with weights 3 ~ 4 mg, irradiated at reduced reactor
power of 500 watt. The experimental set-up was giving
very low dead time and correction was directly compen-
sated by the system. The complete data were shown in
Table (7).

Cumulative fission yield was calculated using Appendix
(B), data were collected from different sources$^{18}$&$^{19}$.

A doublet case was recorded for the case of $E_\gamma = 218$ KeV
and decomposition was required and shown in Appendix (c).

The counts for each required energy peak was stored in an
TN-1710 multi channel analyzer and printed out using a
Table 7 Main Data to Evaluate the CFY for Different Fission Products

<table>
<thead>
<tr>
<th>A₂</th>
<th>Eγ (KeV)</th>
<th>T₁</th>
<th>T₂</th>
<th>T₃</th>
<th>T₁^</th>
<th>Td</th>
<th>Tc</th>
<th>Count ± ΔC%</th>
<th>IFV(1)</th>
<th>CFY(2)</th>
<th>Iγ ± I%</th>
</tr>
</thead>
<tbody>
<tr>
<td>148Ce</td>
<td>269.52 (KeV)</td>
<td>0.6</td>
<td>1.02</td>
<td>56</td>
<td>10.9</td>
<td>67</td>
<td>23</td>
<td>349 ± 22.9%</td>
<td>0.00028</td>
<td>0.00351</td>
<td>0.231 ± 3.9%</td>
</tr>
<tr>
<td></td>
<td>291.7 (KeV)</td>
<td>0.6</td>
<td>1.02</td>
<td>56</td>
<td>10.9</td>
<td>67</td>
<td>23</td>
<td>466 ± 18.7%</td>
<td>0.242 ± 3.8%</td>
<td></td>
<td></td>
</tr>
<tr>
<td>90Kr</td>
<td>121.5 (KeV)</td>
<td>-</td>
<td>1.9</td>
<td>32.3</td>
<td>10.9</td>
<td>67</td>
<td>23</td>
<td>3354 ± 3.5%</td>
<td>0.06136</td>
<td>0.355 ± 8.5%</td>
<td></td>
</tr>
<tr>
<td>136mI</td>
<td>381.4 (KeV)</td>
<td>0.82</td>
<td>17.5</td>
<td>44.8</td>
<td>10.9</td>
<td>67</td>
<td>23</td>
<td>692 ± 11.1%</td>
<td>0.00141</td>
<td>0.02617</td>
<td>0.998 ± 5.5%</td>
</tr>
<tr>
<td>144La</td>
<td>397.4 (KeV)</td>
<td>1</td>
<td>11.9</td>
<td>39.7</td>
<td>10.9</td>
<td>67</td>
<td>23</td>
<td>2658 ± 3.7%</td>
<td>0.00293</td>
<td>0.04157</td>
<td>0.9 ± 5.6%</td>
</tr>
<tr>
<td>$A_Z$</td>
<td>$E_Y$(KeV)</td>
<td>$T_{(1)}$</td>
<td>$T_{(2)}$</td>
<td>$T_{(3)}$</td>
<td>$T_1$</td>
<td>$T_d$</td>
<td>$T_c$</td>
<td>Count ± ΔC%</td>
<td>IFY(1)</td>
<td>CFY(2)</td>
<td>$I_y$ ± $I$ %</td>
</tr>
<tr>
<td>-------</td>
<td>------------</td>
<td>-----------</td>
<td>-----------</td>
<td>-----------</td>
<td>-------</td>
<td>-------</td>
<td>-------</td>
<td>-------------</td>
<td>--------</td>
<td>--------</td>
<td>----------</td>
</tr>
<tr>
<td>137$^\text{Xe}$</td>
<td>455.5 (KeV)</td>
<td>4</td>
<td>24.5</td>
<td>229</td>
<td>10.9</td>
<td>67</td>
<td>87</td>
<td>2281 ± 6%</td>
<td>0.00289</td>
<td>0.02618</td>
<td>0.31 ± 9%</td>
</tr>
<tr>
<td>140$^\text{Cs}$</td>
<td>602.3 (KeV)</td>
<td>0.8</td>
<td>14</td>
<td>65.5</td>
<td>10.9</td>
<td>67</td>
<td>23</td>
<td>1087 ± 6.4%</td>
<td>0.00203</td>
<td>0.03770</td>
<td>0.72 ± 4.2%</td>
</tr>
<tr>
<td>89$^\text{Kr}$</td>
<td>220.9 (KeV)</td>
<td>0.41</td>
<td>4.4</td>
<td>190.8</td>
<td>10.9</td>
<td>67</td>
<td>23</td>
<td>856 ± 40.3%</td>
<td>0.00265</td>
<td>0.01630</td>
<td>0.201 ± 12%</td>
</tr>
<tr>
<td>139$^\text{Xe}$</td>
<td>218.6 (KeV)</td>
<td>0.00001</td>
<td>2.3</td>
<td>39.7</td>
<td>10.9</td>
<td>67</td>
<td>23</td>
<td>2677 ± 44.7%</td>
<td>0.00001</td>
<td>0.00125</td>
<td>0.5 ± 6%</td>
</tr>
</tbody>
</table>
teletype connected to the system. The countings and the counting error bars were analyzed manually by a hand calculator. The main source of uncertainty in final results of calculations was referred to the counting uncertainty. So, many other small error bars were not considered in cumulative fission yield uncertainty calculations, for example for half-lives, independent fission yield and weight.

For determination of the detector absolute efficiency $\varepsilon$, a set of calibrated source were used to find the detector efficiency with different energies of interested energy range. Least square fit was applied using a computer code and a relation (eq. 3) between gamma ray energy and detector absolute efficiency was found, and the uncertainty in that was 4.54%.

Branching ratios and half lives were determined from table of isotope$^{(18)}$, except for the case of $^{148}$Ce, the data were obtained from a recent publication$^{(3)}$ of decay scheme of $^{148}$Ce. The uncertainty of branching ratios were included in our overall uncertainty calculation.

For independent fission yield of the precursors the information were obtained from Crouch$^{(4)}$ by multiplying
the chain yield by the fractional independent fission yield \((IFY = CY \times FIFY)\).

Finally, the timings were obtained either by built in clock in the counting system, experimental measurement, or calculation as the case for irradiation time. Although for the first glance the timing and half-life uncertainty might be considered, but in actually the degree of involvement was very small as seen in Append. (c). On the other hand, considering them may cause much sophistication to uncertainty calculations.

In table (7) the measured results of cumulative fission yield and recommended values from Crouch\(^{7}\) were listed. The measured values were compared to Crouch's since all independent fission yields were taken from it. The measured cumulative fission yields showed good agreement with cumulative fission yields calculated from Crouch's data.

This work was concentrated on the studying of the short-lived \(^{148}\)Ce \((T_{1/2} = 56\) sec\) fission product, which was encouraged to be done experimentally for the first time by the recent publication of the decay scheme for \(^{148}\)Ce by N.K. Aras, et. al\(^{3}\).
Table 8: Cumulative fission yield for short-lived fission products produced in $^{235}\text{U}(\text{n}_{th},\text{f})$ and comparisons to other works:

<table>
<thead>
<tr>
<th>Fission Product</th>
<th>$E_Y$ (KeV)</th>
<th>CFY(%) this work</th>
<th>CFY(%) other work</th>
<th>References</th>
<th>CFY(%) from (4)</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{89}\text{Kr}$</td>
<td>220.9</td>
<td>$5.26 \pm 0.85$</td>
<td>$4.61 \pm 0.11$</td>
<td>(20)</td>
<td>$4.47 \pm 0.08$</td>
</tr>
<tr>
<td>$^{90}\text{Kr}$</td>
<td>121.5</td>
<td>$5.04 \pm 0.27$</td>
<td>$5.06 \pm 0.18$</td>
<td>(20)</td>
<td>$5.22 \pm 0.09$</td>
</tr>
<tr>
<td>$^{136m}\text{I}$</td>
<td>381.4</td>
<td>$1.50 \pm 0.10$</td>
<td></td>
<td></td>
<td>$0.90 \pm 0.47$</td>
</tr>
<tr>
<td>$^{137}\text{Xe}$</td>
<td>455.5</td>
<td>$6.94 \pm 0.37$</td>
<td>$6.12 \pm 0.17$</td>
<td>(21)</td>
<td>$6.11 \pm 0.08$</td>
</tr>
<tr>
<td>$^{139}\text{Xe}$</td>
<td>218.6</td>
<td>$5.47 \pm 1.73$</td>
<td>$5.33 \pm 0.19$</td>
<td>(20)</td>
<td>$6.22 \pm 0.29$</td>
</tr>
<tr>
<td>$^{140}\text{Cs}$</td>
<td>602.3</td>
<td>$3.70 \pm 0.15$</td>
<td></td>
<td></td>
<td>$5.82 \pm 0.07$</td>
</tr>
<tr>
<td>$^{144}\text{La}$</td>
<td>397.4</td>
<td>$4.63 \pm 0.19$</td>
<td></td>
<td></td>
<td>$5.32 \pm 0.05$</td>
</tr>
<tr>
<td>$^{148}\text{Ce}$</td>
<td>269.5</td>
<td>$2.40 \pm 0.26$</td>
<td></td>
<td></td>
<td>$1.50 \pm 0.02$</td>
</tr>
<tr>
<td></td>
<td>291.7</td>
<td>$2.44 \pm 0.26$</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
The measured cumulative fission yield for $^{148}$Ce of $E_\gamma = 269.5$ KeV was 0.0240 with uncertainty of 11.0% and for $E_\gamma = 291.7$ KeV was 0.0244 with uncertainty of 10.7% where Crouch calculated cumulative fission yield was $0.0167 \times 0.9 = 0.0150$. 62.67% difference was recorded. The best agreement was in the case of $^{90}$Kr ($E_\gamma = 121.5$ KeV) with only 3.45% difference.

The minimum uncertainty was for $^{144}$La ($E_\gamma = 397.4$ KeV) which was $\pm 4.0\%$ and the maximum uncertainty was for $^{139}$Xe ($E_\gamma = 218.6$ KeV) with uncertainty of 31.6%. This data was obtained after decomposition of a doublet case (Appendix), where a short lived product $^{139}$Xe ($T_{1/2} = 39.7$ sec) and a longer lived product $^{89}$Kr ($T_{1/2} = 190.8$ sec) fell in the same energy range. So, 31.6% of uncertainty was still acceptable. For normal data the uncertainty for $^{148}$Ce ($E_\gamma = 269.5$ KeV) was maximum which was 11.0%. It was clearly seen that the uncertainty of this work was quite acceptable.
4.2 Conclusions and Remarks

The primary data of this experiment were the independent fission yields for parents taken from Crouch's data. Using literature values for gamma-ray branching ratios and radionuclide half-lives, observed gamma yields were ascribed to decay of specific fission products, and cumulative fission product yields were determined for $^{148}$Ce and some other fission products resulted from thermal fission of $^{235}$U. Where for $^{148}$Ce it was first time experimentally measured data. The only comparison made was to literature data for $^{148}$Ce cumulative fission yield calculated from IFY of Crouch. The cumulative fission yields for isotopes reported in this report were generally in good agreement with the literature values.

The major weakness of this method was the need to rely on nuclear data, especially the gamma-ray branching ratios, any one of which may be incorrect by more than the assigned uncertainty, except for $^{148}$Ce case where the data were taken from a current publication for these ratios. Another drawback was elimination of uncertainties for some sources of data, as they were very small compared with the counting uncertainties. On the other hand, envolving those small uncertainties will result in tedious calculations as seen in Appendix (B).
One of the advantages in the present work was using a high resolution gamma ray spectroscopy to eliminate chemical separations prior to measurements. Although better results may be reached by using a HPGe detector. A computer code was also constructed, which was generalized for future use of similar experimental measurements. As long as the experimental set-up has been constructed, the whole experiment can be done for other fission product measurement. Other target materials can also be used and measurement of cumulative fission yield can be accomplished easily. For example other fission products from thermal or fast neutron fission of Uranium 235 or other element like $^{238}$U or $^{232}$Th can be done by the same set-up with some improvements.

Although almost all the estimated values in nuclear data were appearing as experimental results, still some differences occur. The error responsible for a wrong experimental yield value may be found a few generations of publications back in a constant influencing indirectly the evaluation of the measured data. To eliminate this concerned personnels, scientists, experimenters and publishers should have direct communication to revise all these data and to check whether if the available data were still the best choice, otherwise should be replaced by recent measurements from newly published papers.
APPENDIX - A
DATA ANALYSIS CODE

Appendix A-1 The Program "YIELD" Print-out:

PROGRAM YIELD
DIMENSION W2I(50), WCY (50), CFX (50), SERR2(50)
HL1=0.0
HL2=0.0
HL3=0.0
CC=0.0
TC=0.0
TD=0.0
II=0.0
FY1=0.0
CFY2=0.0
EFF=0.0
GA1=0.0
XNU=0.0
XRATE=0.0
WRITE(7,19)
!19 FORMAT (IX, 'INSERT: NUMBER OF POINTS')
READ(7,13) N
!13 FORMAT (I3)
WRITE(7,20)
!20 FORMAT (IX, 'INSERT: T(1/2) (1), T(1/2) (2), T(1/2) (3), FY1, CFY2')
READ(7,10) TH1, TH2, TH3, FY1, CFY2
!10 FORMAT (3F12.7, 2F12.7)
WRITE(7,600) TH1, TH2, TH3, FY1, CFY2
!600 FORMAT (IX, 5F12.7)
WRITE(7,21)
!21 FORMAT (IX, 'INSERT: TIME, ENERGY, BRANCHING RATIC, B. R. ERR. BAR,
C. LIGHT, FISSION RATE')
READ(7,11) TI, ENERG, GAM, EBI, WE, XRATE
!11 FORMAT (5F15.6, E15.4)
WRITE(7,500) TI, ENERG, GAM, EBI, WE, XRATE
!500 FORMAT (IX, 6E13.6)
XLEN=ALOG10(ENERG)
XLEFF=1.07-1.22*XLEN
EFF=10.0**XLEFF
EED=4.54
WRITE(7,105) ENERG, EFF, EED
!105 FORMAT (IX, 'THE EFFICIENCY', 'E11.5', 'REV', 'E11.5, 'EFFICIENCY
COUNTER BAR', 'E12.5/')
XNU=W*6.02217E23/235.0439*0.93509044
DO 101 I=1, N
!101 WRITE(7,22)
!22 FORMAT (IX, 'INSERT: COUNT NO., COUNT, ERR. BAR., DECAY TIME, COUNT 1
READ(7,12) CC, CEB, TD, TC
!12 FORMAT (F13.1, 3F12.5)
WRITE(7,700) CC, CEB, TD, TC
!700 FORMAT (IX, 'CC=', 'F9.1, 1X, 'CEB=', 'F8.3, 1X, 'TD=', 'F12.9, 1X, 'TC=', 'X','HL1=0.6931471/TH1
HL2=0.6931471/TH2
HL3=0.6931471/TH3
E1=EXP(-(HL1+TI))
E2=EXP(-(HL2+TD))
E3=EXP(-(HL3+TD))
ED1=EXP(-(HL1+TD))
ED2=EXP(-(HL2+TD))
ED3=EXP(-(HL3+TD))
XX=2FF*GAH*XNU*XRATE
G1=1-EI3
A1=(CFY2/(HL2*FY1))*(EI2-EI3)
A2=(EI2-EI3)/(HL2-HL1)
A3=(((HL2-HL3)*(EI1-EI3))/((HL2-HL1)*(HL3-HL1))
A4=(((HL2-HL3)*(1-EI1))/((HL1-HL3)*(HL2-HL1))
A5=1-EI1/((HL2-HL1) FO
A6=(CFY2/(HL2*FY1))*(1-EI2)
A7=(EI1-EI2)/(HL1-HL2)
A8=(((HL2-HL3)*(1-EI1))/((HL2-HL1)*(HL3-HL1))
A9=A1-A2-A3+A4-A5+A6+A7
DD=DD+7-A5
FF=HL2*HL3*FY1/(HL2-HL3)
AA=AA+FF/HL3*ED3*(1-EXP(-(HL3+TC)))*XX
DD=DD+ET/HL2*ED2*(1-EXP(-(HL2+TC)))*XX
B1=B1+FF/HL1*ED1*(1-EXP(-(HL1+TC)))*XX
G1=G1/HL3*ED3*(1-EXP(-(HL3+TC)))*XX
CFY3(I)=(CC-AA+DD-B1)/G1
WRITE(7,100)I,CFY3(I)
100 FORMAT(1X,'CFY.FISS.YIELD OF POINT(',I2,')=',E20.6//)
ERR=SQR(T(CED**2+EBR**2+EBI**2)
WEI(I)=1/ERR**2
101 CONTINUE
SFCFV=0.0
SWEI=0.0
DO 103 K=1,M
HCFY(K)=WEI(K)*CFY3(K)
SFCFV=SFCFV+HCFY(K)
SWEI=SWEI+WEI(K)
103 CONTINUE
ACFV=SFCFV/SWEI
WRITE(7,110)ACFY
110 FORMAT(4X,'WEIGHTED CFY =',F8.5)
SBAR=0.0
DO 102 J=1,M
SERR2(J)=1.0/SQR(T(WEI(J))
102 CONTINUE
PBAR=1.0/SQR(T(SWEI)
TBAR=PBAR*ACFY*0.01
WRITE(7,220)TBAR
220 FORMAT(4X,'THE ERROR BAR =',F7.5)
WRITE(7,999)PBAR
999 FORMAT(4X,'THE PERCENT ERROR BAR =',F8.5,'%//)
DO 104 L=1,M
WRITE(7,997)L,CFY3(L),SERR2(L)
997 FORMAT(4X,'CFY3(',I2,')=',E13.6,1X,'+/-',F10.5,'%)
104 CONTINUE
END
Appendix A-2 The Program "YIELD" Output Results:

RUN ASS
INSERT: NUMBER OF POINTS
6
INSERT: T(1/2)(1), T(1/2)(2), T(1/2)(3), F1, CFY2
0.0000001 1.0 932.30 0.0000001 0.061364
0.0000001 1.9000000 32.2979772 0.0000000 0.001345
INSERT: IRR. TIME, ENERGY, BRANCHING RATIO, D.R., ERR. BAR, WEIGHT FISSION RATE
10.9 121.5 0.395 8.5 0.0032 0.0550C-13
0.109000E+02 0.121500E+03 0.355000E+00 0.053300E+01 0.121500E-02 0.121500E-13
THE EFFICIENCY(0.121500E+03KEV)=0.33637E-01
EFFICIENCY ERROR BAR= 0.45400E+01

INSERT: COUNT NO., COUNT. ERR. BAR, DECAY TIME, COUNT TIME
3354.0 3.5 67.0 23.0
CC= 3354.0 CEB= 3.500 TD= 67.000000 TC= 23.000000
CUM. FISS. YIELD OF POINT( 1) = 0.45775E-01

INSERT: COUNT NO., COUNT. ERR. BAR, DECAY TIME, COUNT TIME
2191.0 4.8 90.0 22.0
CC= 2191.0 CEB= 4.800 TD= 90.000000 TC= 22.000000
CUM. FISS. YIELD OF POINT( 2) = 0.52467E-01

INSERT: COUNT NO., COUNT. ERR. BAR, DECAY TIME, COUNT TIME
1252.0 7.4 112.0 21.0
CC= 1252.0 CEB= 7.400 TD= 112.000000 TC= 21.000000
CUM. FISS. YIELD OF POINT( 3) = 0.49675E-01

INSERT: COUNT NO., COUNT. ERR. BAR, DECAY TIME, COUNT TIME
713.0 11.8 133.0 21.0
CC= 713.0 CEB= 11.800 TD= 133.000000 TC= 21.000000
CUM. FISS. YIELD OF POINT( 4) = 0.43707E-01

INSERT: COUNT NO., COUNT. ERR. BAR, DECAY TIME, COUNT TIME
1128.0 10.3 147.0 43.0
CC= 1128.0 CEB= 10.300 TD= 147.000000 TC= 43.000000
CUM. FISS. YIELD OF POINT( 5) = 0.57670E-01

INSERT: COUNT NO., COUNT. ERR. BAR, DECAY TIME, COUNT TIME
202.0 36.6 232.0 42.0
CC= 202.0 CEB= 36.600 TD= 232.000000 TC= 42.000000
CUM. FISS. YIELD OF POINT( 6) = 0.65420E-01

WEIGHTED CFY = 0.05040
THE ERROR BAR = 0.00270
THE PERCENT ERROR BAR = 5.34715%

CFY( 1)= 0.469906E-01 +/- 10.25237%
CFY( 2)= 0.52467E-01 +/- 10.74570%
CFY( 3)= 0.49675E-01 +/- 12.14976%
CFY( 4)= 0.43979E-01 +/- 15.23400%
CFY( 5)= 0.57670E-01 +/- 14.10500%
CFY( 6)= 0.65420E-01 +/- 37.04754%
STOP --
RUN ASS
INSERT: NUMBER OF POINTS
6
INSERT: T(1/2)(1), T(1/2)(2), T(1/2)(3), FY1, CFY2
0.6, 1.02, 56.0, 0.0028056, 0.0031360
0.6000000, 1.0200000, 56.0000000, 0.392283, 0.003513
INSERT: ERR, TIME, ENERGY, BRANCHING RATIO, D.R., ERR, BAR, HEIGHT, FISSION RATE
10.9, 369.5, 0.231128, 3.50, 0.0032, 0.0035, 13
0.109000E+02, 0.269520E+03, 0.231120E+00, 0.003200E+00, 0.003200E+00, 0.003500E+00
THE EFFICIENCY(0.26952E+03KEV) = 0.12730E-01
EFFICIENCY ERROR BAR = 0.04549E+01

INSERT: COUNT NO., COUNT, ERR, BAR, DECAY TIME, COUNT TIME
349.0, 22.9, 67.0, 23.0
CC = 349.0, CEB = 22.900, TD = 67.00000000
CUM. FISSION YIELD OF POINT (1) = 0.17443E+01

INSERT: COUNT NO., COUNT, ERR, BAR, DECAY TIME, COUNT TIME
425.0, 17.2, 90.0, 22.0
CC = 425.0, CEB = 17.200, TD = 90.00000000
CUM. FISSION YIELD OF POINT (2) = 0.27373E+01

INSERT: COUNT NO., COUNT, ERR, BAR, DECAY TIME, COUNT TIME
215.0, 31.2, 112.0, 21.0
CC = 215.0, CEB = 31.200, TD = 112.00000000
CUM. FISSION YIELD OF POINT (3) = 0.20311E+01

INSERT: COUNT NO., COUNT, ERR, BAR, DECAY TIME, COUNT TIME
314.0, 21.4, 147.0, 43.0
CC = 314.0, CEB = 21.400, TD = 147.00000000
CUM. FISSION YIELD OF POINT (4) = 0.35390E+01

INSERT: COUNT NO., COUNT, ERR, BAR, DECAY TIME, COUNT TIME
100.0, 61.1, 190.0, 42.0
CC = 100.0, CEB = 61.100, TD = 190.00000000
CUM. FISSION YIELD OF POINT (5) = 0.13736E+01

INSERT: COUNT NO., COUNT, ERR, BAR, DECAY TIME, COUNT TIME
61.0, 88.3, 232.0, 42.0
CC = 61.0, CEB = 88.300, TD = 232.00000000
CUM. FISSION YIELD OF POINT (6) = 0.14257E+01

WEIGHTED CFY = 0.02378
THE ERROR BAR = 0.00265
THE PERCENT ERROR BAR = 11.02702%

CFY3( 1) = 0.17443E-01 +/- 23.672E2
CFY3( 2) = 0.29393E-01 +/- 23.672E2
CFY3( 3) = 0.20311E-01 +/- 31.700E2
CFY3( 4) = 0.35390E-01 +/- 52.902E2
CFY3( 5) = 0.13736E-01 +/- 31.372E2
CFY3( 6) = 0.14257E-01 +/- 66.372E2
STOP --
RUN AC
INSERT: NUMBER OF POINTS
5
INSERT: T(1/2) (1), T(1/2) (2), T(1/2) (3) , FY1, CFY2
0.6 1.02 56.0 0.0002056, 0.00351309
0.000000 1.000000 56.0059999 0.0002056 0.00351309
INSERT: IRR. TIME, ENERGY, BRANCHING RATIO, B.R. ERR. BAR, WEIGHTED FISSION YIELD
10.9 291.7 0.2422 3.8 0.0032 0.0037137
0.109000E+02 0.291700E+03 0.242200E+03 0.362200E+03 0.511990E+02 0.109000E+02
THE EFFICIENCY (0.29170E+03KEV)=0.11528E-01
EFFICIENCY ERROR BAR= 0.45400E-01

INSERT: COUNT NO., COUNT. ERR. BAR, DECAY TIME, COUNT TIME
466.0 18.7 67.0 23.0
CC= 466.0 CEB= 18.700 TD= 47.000000 TC= 22.000000
CUM. FISSION YIELD OF POINT (1) = 0.292643E-01

INSERT: COUNT NO., COUNT. ERR. BAR, DECAY TIME, COUNT TIME
350.0 17.9 90.0 22.0
CC= 350.0 CEB= 17.700 TD= 70.000000 TC= 22.000000
CUM. FISSION YIELD OF POINT (2) = 0.254763E-01

INSERT: COUNT NO., COUNT. ERR. BAR, DECAY TIME, COUNT TIME
217.0 23.0 133.0 21.0
CC= 217.0 CEB= 23.000 TD= 133.000000 TC= 21.000000
CUM. FISSION YIELD OF POINT (3) = 0.277633E-01

INSERT: COUNT NO., COUNT. ERR. BAR, DECAY TIME, COUNT TIME
197.0 29.7 147.0 43.0
CC= 197.0 CEB= 29.700 TD= 147.000000 TC= 43.000000
CUM. FISSION YIELD OF POINT (4) = 0.167170E-01

INSERT: COUNT NO., COUNT. ERR. BAR, DECAY TIME, COUNT TIME
85.0 50.9 232.0 42.0
CC= 85.0 CEB= 50.900 TD= 232.000000 TC= 42.000000
CUM. FISSION YIELD OF POINT (5) = 0.219466E-01

WEIGHTED CFY = 0.02439
THE ERROR BAR = 0.00261
THE PERCENT ERROR BAR = 10.71204

CFY3( 1)= 0.245043E-01 +/- 19.31463
CFY3( 2)= 0.254306E-01 +/- 10.56571
CFY3( 3)= 0.279663E-01 +/- 23.74777
CFY3( 4)= 0.167170E-01 +/- 30.20435
CFY3( 5)= 0.219466E-01 +/- 51.24216

STOP --
### Run Ass

**Insert: Number of Points**

<table>
<thead>
<tr>
<th>Point</th>
<th>Count</th>
<th>Error</th>
<th>Decay Time</th>
<th>Count</th>
<th>Cumulative Fiss. Yield</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>692.0</td>
<td>11.1</td>
<td>67.0</td>
<td>23.0</td>
<td>0.12397E-01</td>
</tr>
<tr>
<td>2</td>
<td>463.0</td>
<td>14.5</td>
<td>70.0</td>
<td>22.0</td>
<td>0.12599E-01</td>
</tr>
<tr>
<td>3</td>
<td>365.0</td>
<td>16.2</td>
<td>112.0</td>
<td>21.0</td>
<td>0.14917E-01</td>
</tr>
<tr>
<td>4</td>
<td>347.0</td>
<td>15.0</td>
<td>133.0</td>
<td>21.0</td>
<td>0.17109E-01</td>
</tr>
<tr>
<td>5</td>
<td>373.0</td>
<td>17.4</td>
<td>147.0</td>
<td>43.0</td>
<td>0.19557E-01</td>
</tr>
<tr>
<td>6</td>
<td>172.0</td>
<td>32.0</td>
<td>190.0</td>
<td>42.0</td>
<td>0.22758E-01</td>
</tr>
<tr>
<td>7</td>
<td>156.0</td>
<td>30.1</td>
<td>232.0</td>
<td>42.0</td>
<td>0.23921E-01</td>
</tr>
<tr>
<td>8</td>
<td>89.0</td>
<td>49.4</td>
<td>274.0</td>
<td>41.0</td>
<td>0.24024E-01</td>
</tr>
</tbody>
</table>

**Weighted CFY = 0.01504**

**The Error Bar = 0.00102**

**The Percent Error Bar = 6.30472%**

CFY (1) = 0.12710E-01 +/- 13.1734E-02
CFY (2) = 0.12598E-01 +/- 16.1500E-02
CFY (3) = 0.14517E-01 +/- 17.7003E-02
CFY (4) = 0.19100E-01 +/- 16.0909E-02
CFY (5) = 0.14357E-01 +/- 13.0640E-02
CFY (6) = 0.13275E-01 +/- 32.7033E-02
CFY (7) = 0.23061E-01 +/- 30.7332E-02
CFY (8) = 0.25634E-01 +/- 47.7121E-02

**Stop --**
RUN ASS
INSERT: NUMBER OF POINTS
8
INSERT: I(1/2)(1)+T(1/2)(2)+T(1/2)(3)+Y1,CFY2
1.0:11.9,3.9,7.0,0.00292672,0.0415628
1.000000 11.899996 39.7000000 0.0027267 0.0415628
INSERT: I(TIME,ENERGY,BRANCHING RATIO,TOTAL ERR.BAR,WEIGHTED FISSION YIELD)
10.9,397.4,0.95,6.0,0.0032,0.0588E-13
0.109000E+02 0.377400E+03 0.900000E+00 0.540000E+12 0.229999E+02 0.900000E+12
THE EFFICIENCY(0.3974E+03KEV)=0.77240E-02
FFICIENCY ERROR BAR= 0.045400E+01

INSERT: COUNT NO., COUNT, ERR. BAR, DECAY TIME, COUNT TIME
258.0,3.7,67.0,0.033
CC= 258.0 CEB= 3.700 TD= 67.0000000 TC= 21.0000000
CUM. FISS. YIELD OF POINT( 1) = 0.407300E+01

INSERT: COUNT NO., COUNT, ERR. BAR, DECAY TIME, COUNT TIME
1728.0,4.9,90.0,0.220
CC= 1728.0 CEB= 4.900 TD= 90.0000000 TC= 22.0000000
CUM. FISS. YIELD OF POINT( 2) = 0.420090E+01

INSERT: COUNT NO., COUNT, ERR. BAR, DECAY TIME, COUNT TIME
1164.0,6.3,112.0,0.210
CC= 1164.0 CEB= 6.300 TD= 112.0000000 TC= 21.0000000
CUM. FISS. YIELD OF POINT( 3) = 0.440200E+01

INSERT: COUNT NO., COUNT, ERR. BAR, DECAY TIME, COUNT TIME
982.0,6.6,133.0,0.210
CC= 982.0 CEB= 6.600 TD= 133.0000000 TC= 21.0000000
CUM. FISS. YIELD OF POINT( 4) = 0.572220E+01

INSERT: COUNT NO., COUNT, ERR. BAR, DECAY TIME, COUNT TIME
1140.0,8.8,147.0,0.430
CC= 1140.0 CEB= 8.800 TD= 147.0000000 TC= 43.0000000
CUM. FISS. YIELD OF POINT( 5) = 0.467734E+01

INSERT: COUNT NO., COUNT, ERR. BAR, DECAY TIME, COUNT TIME
457.0,14.9,190.0,0.420
CC= 457.0 CEB= 14.900 TD= 190.0000000 TC= 42.0000000
CUM. FISS. YIELD OF POINT( 6) = 0.379620E+01

INSERT: COUNT NO., COUNT, ERR. BAR, DECAY TIME, COUNT TIME
153.0,37.3,232.0,0.420
CC= 153.0 CEB= 37.300 TD= 232.0000000 TC= 42.0000000
CUM. FISS. YIELD OF POINT( 7) = 0.207140E+01

INSERT: COUNT NO., COUNT, ERR. BAR, DECAY TIME, COUNT TIME
181.0,28.2,273.0,0.410
CC= 181.0 CEB= 28.200 TD= 273.0000000 TC= 41.0000000
CUM. FISS. YIELD OF POINT( 9) = 0.777120E+01

WEIGHTED CFY = 0.04425
THE ERROR BAR = 0.00104
THE PERCENT ERROR BAR = 3.70351%

CFY3( 1)= 0.437100E-01 +/- 0.10310
CFY3( 2)= 0.428694E-01 +/- 0.71074
CFY3( 3)= 0.440388E-01 +/- 0.71074
CFY3( 4)= 0.572225E-01 +/- 0.71074
CFY3( 5)= 0.467734E-01 +/- 0.71074
CFY3( 6)= 0.379620E-01 +/- 0.71074
CFY3( 7)= 0.207140E-01 +/- 0.71074
CFY3( 8)= 0.777120E-01 +/- 0.71074

STOP --
RUN ASS
INSERT: NUMBER OF POINTS
7
INSERT:T(1/2) (1), T(1/2) (2), T(1/2) (2), FY1, CFY2
4.0 24.5 229.0 0.000008 0.001479
4.0 24.5 229.0 0.000008 0.001479
4.0 24.5 229.0 0.000008 0.001479
INSERT: IRR. TIME ENERGY BRANCHING RATIO: BR. ERR. BAR. WEIGHT. Fission Rate
10.9 455.5 0.31 7.0 0.0032 0.0533E-13
0.109000E + 02 0.455000E + 03 0.310000E + 00 0.700000E + 01 0.320000E + 02 0.100000E + 12
THE EFFICIENCY (0.45500E + 03 KEV) = 0.67000E - 02
EFFICIENCY ERROR BAR = 0.45400E + 01

INSERT: COUNT NO., COUNT, ERR. BAR, DECAY TIME, COUNT TIME
2281.0 6.0 67.0 87.0
CC = 2281.0 CER = 6.000 TD = 67.000000 TC = 87.000000
CUM. FISS. YIELD OF POINT( 1) = 0.666610E - 01

INSERT: COUNT NO., COUNT, ERR. BAR, DECAY TIME, COUNT TIME
1363.0 8.1 147.0 85.0
CC = 1363.0 CER = 8.100 TD = 147.000000 TC = 85.000000
CUM. FISS. YIELD OF POINT( 2) = 0.654270E - 01

INSERT: COUNT NO., COUNT, ERR. BAR, DECAY TIME, COUNT TIME
1050.0 9.4 232.0 81.0
CC = 1050.0 CER = 9.400 TD = 232.000000 TC = 81.000000
CUM. FISS. YIELD OF POINT( 3) = 0.564270E - 01

INSERT: COUNT NO., COUNT, ERR. BAR, DECAY TIME, COUNT TIME
731.0 10.4 307.0 82.0
CC = 731.0 CER = 10.400 TD = 307.000000 TC = 82.000000
CUM. FISS. YIELD OF POINT( 4) = 0.500000E - 01

INSERT: COUNT NO., COUNT, ERR. BAR, DECAY TIME, COUNT TIME
744.0 9.6 397.0 82.0
CC = 744.0 CER = 9.600 TD = 397.000000 TC = 82.000000
CUM. FISS. YIELD OF POINT( 5) = 0.527770E - 01

INSERT: COUNT NO., COUNT, ERR. BAR, DECAY TIME, COUNT TIME
414.0 13.9 471.0 82.0
CC = 414.0 CER = 13.900 TD = 471.000000 TC = 82.000000
CUM. FISS. YIELD OF POINT( 6) = 0.527770E - 01

INSERT: COUNT NO., COUNT, ERR. BAR, DECAY TIME, COUNT TIME
378.0 14.1 553.0 82.0
CC = 378.0 CER = 14.100 TD = 553.000000 TC = 82.000000
CUM. FISS. YIELD OF POINT( 7) = 0.637110E - 01

WEIGHTED CFY = 0.06743
THE ERROR BAR = 0.00367
THE PERCENT ERROR BAR = 5.27231%

CFY3( 1) = 0.060810E - 01 +/- 11.7900%
CFY3( 2) = 0.054780E - 01 +/- 12.7347%
CFY3( 3) = 0.664270E - 01 +/- 11.2012%
CFY3( 4) = 0.580287E - 01 +/- 14.6217%
CFY3( 5) = 0.769642E - 01 +/- 13.9521%
CFY3( 6) = 0.537770E - 01 +/- 17.1697%
CFY3( 7) = 0.637110E - 01 +/- 17.3333%
STOP ---
**RUN ASS**

**INSERT: NUMBER OF POINTS**

9

**INSERT:**

<table>
<thead>
<tr>
<th>POINT</th>
<th>NUMBER OF POINTS</th>
<th>INSERT :T(l/2Xl)fT(l.'2)&lt;2) rT &lt; l/2&gt; (3&gt; rTYlr CTri:</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0.88300000 14.00000000 66.5000000</td>
<td>0.265212 0.03770225</td>
</tr>
</tbody>
</table>

**INSERT:**

<table>
<thead>
<tr>
<th>POINT</th>
<th>NUMBER OF TIME: ENERGY, BRANCHING RATIO: B: ERR: BAR: WEIGHT: FISSION YIELD</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>10.9602.40.72.4.2.0.0032.8.925E-13</td>
</tr>
<tr>
<td>2</td>
<td>0.109000E01 0.602400E01 0.720000E12 0.420000E01</td>
</tr>
</tbody>
</table>

**THE EFFICIENCY(0.602400E01KEV)=0.47702E-02**

**EFFICIENCY ERROR BAR= 0.45400E01**

**INSERT:**

<table>
<thead>
<tr>
<th>POINT</th>
<th>COUNT NO.</th>
<th>COUNT.ERR.BAR.</th>
<th>DECAY TIME</th>
<th>COUNT TIME</th>
<th>CUM. FISS. YIELD OF POINT</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>9078.0 6.467.0 23.0</td>
<td>CC= 907.0 CEB= 6.400 TD= 67.000000 TC= 22.00220000</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>2</td>
<td>857.0 6.900 22.0</td>
<td>CC= 857.0 CEB= 6.900 TD= 70.000000 TC= 22.00220000</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>3</td>
<td>682.0 7.600 21.0</td>
<td>CC= 682.0 CEB= 7.600 TD= 70.000000 TC= 22.00220000</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>4</td>
<td>450.0 10.700 21.0</td>
<td>CC= 450.0 CEB= 10.700 TD= 113.000000 TC= 22.00220000</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>5</td>
<td>734.0 9.200 21.0</td>
<td>CC= 734.0 CEB= 9.200 TD= 112.000000 TC= 22.00220000</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>6</td>
<td>429.0 12.600 21.0</td>
<td>CC= 429.0 CEB= 12.600 TD= 170.000000 TC= 22.00220000</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>7</td>
<td>285.0 15.300 21.0</td>
<td>CC= 285.0 CEB= 15.300 TD= 232.000000 TC= 22.00220000</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

**CUM. FISS. YIELD OF POINT( 1) = 0.270792E-01**

**CUM. FISS. YIELD OF POINT( 2) = 0.270792E-01**

**CUM. FISS. YIELD OF POINT( 3) = 0.270792E-01**

**CUM. FISS. YIELD OF POINT( 4) = 0.270792E-01**

**CUM. FISS. YIELD OF POINT( 5) = 0.270792E-01**

**CUM. FISS. YIELD OF POINT( 6) = 0.270792E-01**

**CUM. FISS. YIELD OF POINT( 7) = 0.270792E-01**

**CUM. FISS. YIELD OF POINT( 8) = 0.270792E-01**

**CUM. FISS. YIELD OF POINT( 9) = 0.270792E-01**

**WEIGHTED CFY = 0.03703**

**THE ERROR BAR = 0.00151**

**THE PERCENT ERROR BAR = 4.00742%**

**CFY( 1)= 0.394722E-01 **

**CFY( 2)= 0.397722E-01 **

**CFY( 3)= 0.415675E-01 **

**CFY( 4)= 0.323094E-01 **

**CFY( 5)= 0.365215E-01 **

**CFY( 6)= 0.307905E-01 **

**CFY( 7)= 0.302727E-01 **

**CFY( 8)= 0.270933E-01 **

**CFY( 9)= 0.320364E-01 **

**STOP --**
RUN ASSS'S
INSERT: NUMBER OF POINTS
7

INSERT: T(1/2)(1), T(1/2)(2), T(1/2)(3), T(1/2)(4), T(1/2)(5), T(1/2)(6), T(1/2)(7), T(1/2)(8), FY1, CFY2
0.41 ± 0.000004, 0.00000042 ± 0.00000042
0.4100000 4.4000000 170.0000000 0.00000000 0.0000000 0.00000002
INSERT: IRR. TIME, ENERGY, BRANCHING RATIO, ERR. BAR, WEIGHT, DECAY RATE
10.9220 ± 0.2012, 12.0000000 ± 0.00000000 ± 12
0.12000000 ± 0.00000000 ± 12 0.12000000 ± 0.00000000 ± 12
THE EFFICIENCY (0.22000000 ± 0.00000000 ± 0.0000000)

INSERT: COUNT NO., COUNT ERR. BAR, DECAY TIME, COUNT TIME
856.0 ± 40.3, 25.0 ± 20.0
CC = 856.0 CEB = 40.3 CEB = 20.0 CEB = 20000000 TC = 20000000
CUM. FISS. YIELD OF POINT( 1) = 0.63 ± 0.00000000 ± 0.0000000

INSERT: COUNT NO., COUNT ERR. BAR, DECAY TIME, COUNT TIME
752.0 ± 40.9, 20.0 ± 20.0
CC = 752.0 CEB = 40.9 CEB = 20.0 CEB = 20000000 TC = 20000000
CUM. FISS. YIELD OF POINT( 2) = 0.62 ± 0.00000000 ± 0.0000000

INSERT: COUNT NO., COUNT ERR. BAR, DECAY TIME, COUNT TIME
664.0 ± 41.1, 21.0 ± 21.0
CC = 664.0 CEB = 41.1 CEB = 21.0 CEB = 20000000 TC = 20000000
CUM. FISS. YIELD OF POINT( 3) = 0.62 ± 0.00000000 ± 0.0000000

INSERT: COUNT NO., COUNT ERR. BAR, DECAY TIME, COUNT TIME
613.0 ± 41.1, 21.0 ± 21.0
CC = 613.0 CEB = 41.1 CEB = 21.0 CEB = 20000000 TC = 20000000
CUM. FISS. YIELD OF POINT( 4) = 0.62 ± 0.00000000 ± 0.0000000

INSERT: COUNT NO., COUNT ERR. BAR, DECAY TIME, COUNT TIME
1195.0 ± 39.6, 43.0 ± 43.0
CC = 1195.0 CEB = 39.6 CEB = 43.0 CEB = 4300000000 TC = 4300000000
CUM. FISS. YIELD OF POINT( 5) = 0.65 ± 0.00000000 ± 0.0000000

INSERT: COUNT NO., COUNT ERR. BAR, DECAY TIME, COUNT TIME
1000.0 ± 40.2, 42.0 ± 42.0
CC = 1000.0 CEB = 40.2 CEB = 42.0 CEB = 4200000000 TC = 4200000000
CUM. FISS. YIELD OF POINT( 6) = 0.65 ± 0.00000000 ± 0.0000000

INSERT: COUNT NO., COUNT ERR. BAR, DECAY TIME, COUNT TIME
857.0 ± 39.2, 232.0 ± 42.0
CC = 857.0 CEB = 39.2 CEB = 232.0 CEB = 23200000000 TC = 23200000000
CUM. FISS. YIELD OF POINT( 7) = 0.65 ± 0.00000000 ± 0.0000000

WEIGHTED CFY = 0.064003
THE ERROR BAR = 0.00000000 ± 0.00000000 ± 0.00000000
THE PERCENT ERROR BAR = 16.08395%

CFY3( 1) = 0.632050E-01 ± 42.27305%
CFY3( 2) = 0.627943E-01 ± 42.29505%
CFY3( 3) = 0.630057E-01 ± 43.05005%
CFY3( 4) = 0.627710E-01 ± 43.05005%
CFY3( 5) = 0.654146E-01 ± 41.62505%
CFY3( 6) = 0.654033E-01 ± 43.71505%
CFY3( 7) = 0.652970E-01 ± 41.24505%
STOP --
RUN ASS

INSERT: NUMBER OF POINTS
3
INSERT:(1/2)(1),T(1/2)(2),T(1/2)(3),FY1,CFY2
0.0000001,2.3,39.7,0.0000001,0.0012302
0.0000001,2.3000000,39.7000000,0.0000001,0.0012302
INSERT:IRR. TIME, ENERGY, BRANCHING RATIO, S.E.RR. BAR, WEIGHT, FISSION RATE
10.9,218.6,0.5,6.0,0.0032,8.0538E-12
1.09000E+02,0.213600E+03,0.500000E+00,0.000000E+01,0.200000E+02,0.000000E+12
THE EFFICIENCY (0.21860E+03KEV)=0.16430E-01
EFFICIENCY ERROR BAR = 0.45400E+01

INSERT: COUNT NO., COUNT ERR. BAR, DECAY TIME, COUNT TIME
2677.0,44.7,67.0,23.0
CC = 2677.0 CEB = 44.700 TD = 67.0000000 TC = 23.0000000
CUM. FISSION YIELD OF POINT (1) = 0.517453E-01

INSERT: COUNT NO., COUNT ERR. BAR, DECAY TIME, COUNT TIME
1857.0,55.7,90.0,22.0
CC = 1857.0 CEB = 55.700 TD = 90.0000000 TC = 22.0000000
CUM. FISSION YIELD OF POINT (2) = 0.552363E-01

INSERT: COUNT NO., COUNT ERR. BAR, DECAY TIME, COUNT TIME
1300.0,71.1,112.0,21.0
CC = 1300.0 CEB = 71.100 TD = 112.0000000 TC = 21.0000000
CUM. FISSION YIELD OF POINT (3) = 0.535429E-01

WEIGHTED CFY = 0.05467
THE ERROR BAR = 0.01730
THE PERCENT ERROR BAR = 31.640834%

CFY3(1) = 0.519433E-01 +/- 45.329922%
CFY3(2) = 0.558363E-01 +/- 56.205007%
CFY3(3) = 0.596420E-01 +/- 71.477000%
STOP --
APPENDIX - B

(Growth and Decay of Fission Product)

If we consider a chain of three radioactive products for present data evaluation, since most of the precursor nuclides have very short life-times and had decayed completely by initiation of the first gamma-ray count. The figure below represents the decay process for present case:

![Diagram of radioactive decay process]

Fig.12: Successive radioactive decay process.

Let us take up the general case for the decay of a radioactive species, denoted by subscript 1, to produce another radioactive species, denoted by subscript 2, to produce another radioactive species, denoted by subscript 3. Where $\lambda_1$, $\lambda_2$, $\lambda_3$, IFY(1), IFY(2) and IFY(3) are the decay constants and the independent fission yield for the radioactive species 1, 2 and 3 respectively.

Then the behaviour of the three radioactive species during irradiation will be:
Then the behaviour of the three radioactive species during irradiation will be

\[ \lambda_1 N_1(t) = \text{IFY}(1) \times (1 - e^{-\lambda_1 t}) \]  

(1)

\[ \lambda_2 N_2(t) = [\text{IFY}(1) + \text{IFY}(2)](1 - e^{-\lambda_2 t}) + \frac{\text{IFY}(1) 2}{\lambda_1 - \lambda_2} (e^{-\lambda_1 t} - e^{-\lambda_2 t}) \]

\[ = \text{IFY}(2) \times (1 - e^{-\lambda_2 t}) + \frac{\text{IFY}(1) 2}{\lambda_1 - \lambda_2} (e^{-\lambda_1 t} - e^{-\lambda_2 t}) \]

(2)

\[ \lambda_3 N_3(t) = \text{IFY}(3) \times (1 - e^{-\lambda_3 t}) + \frac{\text{IFY}(2) \lambda_3}{\lambda_2 - \lambda_3} (e^{-\lambda_2 t} - e^{-\lambda_3 t}) + \]

\[ \frac{\text{IFY}(1) \lambda_2 \lambda_3}{(\lambda_2 - \lambda_1)(\lambda_3 - \lambda_2)(\lambda_3 - \lambda_1)} [(\lambda_3 - \lambda_1)(e^{-\lambda_1 t} - e^{-\lambda_3 t}) - \]

\[ (\lambda_3 - \lambda_2)(e^{-\lambda_2 t} - e^{-\lambda_3 t})] \]

(3)

At the end of irradiation, if \( t_i \) is the irradiation period:

then:

\[ A_o (1) = \lambda_1 N_1(t_i) \]  

(4)

\[ A_o (2) = \lambda_2 N_2(t_i) \]  

(5)

\[ A_o (3) = \lambda_3 N_3(t_i) \]  

(6)
\[ A_o(1) = IFY(1)(1 - e^{-\lambda_1 t}) \]  
(7)

\[ A_o(2) = CFY(2)(1 - e^{-\lambda_2 t}) + \frac{IFY(1)}{\lambda_1 - \lambda_2} \left( e^{-\lambda_1 t} - e^{-\lambda_2 t} \right) \]  
(8)

\[ A_o(3) = CFY(3)(1 - e^{-\lambda_3 t}) + \frac{CFY(2)\lambda_3}{\lambda_2 - \lambda_3} \left( e^{-\lambda_2 t} - e^{-\lambda_3 t} \right) + \]

\[ \frac{IFY(1)\lambda_2\lambda_3}{(\lambda_2 - \lambda_1)(\lambda_3 - \lambda_2)(\lambda_3 - \lambda_1)} \left( \lambda_3 - \lambda_1 \right) \left( e^{-\lambda_2 t} - e^{-\lambda_3 t} \right) \]

\[ -\lambda_1 t \left( e^{-\lambda_1 t} - e^{-\lambda_3 t} \right) \]

\[ (e^{-\lambda_2 t} - e^{-\lambda_3 t}) \]

(9)

After the end of irradiation, if \( t_0 = 0 \) = the time at the end of irradiation, then at any decay time \( t_d \) (from \( t_0 \)):

\[ A_3(t_d) = A_o(3) e^{-\lambda_3 t_d} + \left( \frac{A_o(1)\lambda_2\lambda_3}{(\lambda_2 - \lambda_1)(\lambda_3 - \lambda_1)} \right) e^{-\lambda_1 t_d} + \]

\[ \left( \frac{A_o(2)}{(\lambda_2 - \lambda_1)} \right) \frac{\lambda_3}{\lambda_3 - \lambda_2} e^{-\lambda_2 t_d} + \]

\[ \left( \frac{A_o(1)\lambda_1\lambda_2}{(\lambda_1 - \lambda_3)(\lambda_2 - \lambda_1)} - \frac{\lambda_3\lambda_2 A_o(1)}{(\lambda_2 - \lambda_3)(\lambda_2 - \lambda_1)} + \frac{A_o(2)\lambda_3}{(\lambda_2 - \lambda_3)} \right) e^{-\lambda_3 t_d} \]

(10)
A simple approximation can be made:

If a) $t_i \gg T_{1/2}(1)$ and $T_{1/2}(2)$

b) $t_d \gg T_{1/2}(1)$ and $T_{1/2}(2)$

and c) $\lambda_1, \lambda_2 >> \lambda_3$

then $A_\alpha(1) = \text{IFY}(1)$

$A_\alpha(2) = \text{CFY}(2)$

$A_\alpha(3) = \text{CFY}(3)(1-e^{-\lambda_3 t_i}) + \ldots \text{CFY}(3)(1-e^{-\lambda_3 t})$  

$A_3(t_d) = \text{CFY}(3)(1-e^{-\lambda_3 t_i})e^{-\lambda_3 t_d}$

After decay time $t_d$ we start collecting data during time period $t_c$:

$C \ (\text{counts}) = \varepsilon(E_\gamma) \cdot I_\gamma(E_\gamma) \int_{t_d}^{t_c+t_d} A_3(t) \cdot d_t \cdot N \cdot \phi_{th} \cdot \sigma_{th}$  

(15)
Substitute (14) into (15)

\[ C \text{ (counts)} = \varepsilon I_y . CFY(3) . N . \phi_{th} . \sigma_{th} \cdot (1 - e^{-\lambda_1 t}) e^{-\lambda_2 t} \cdot e^{-\lambda_3 t_c} \cdot e^{-\lambda d} \cdot \frac{1 - e^{-\lambda d}}{\lambda} \]

\[ = \varepsilon I_y . (CFY) . N . \phi_{th} . \sigma_{th} \cdot t \cdot e^{-\lambda_2 t} \cdot e^{-\lambda_3 t_c} \]

Equation (16) is applicable where assumptions a, b and c stands. But for present work a computer code was required, so, the solution must be general where assumptions a, b and c don't stand.

For the later solution substitute (7), (8) and (9) into (10) directly and rearranging:

\[ A_2(t_d) = CFY(3) (1 - e^{-\lambda_3 t}) e^{-\lambda_3 t} + \frac{CFY(2)}{2 \text{IFY}(1)} (e^{-\lambda_2 t} - e^{-\lambda_3 t}) - \frac{(e^{-\lambda_2 t} - e^{-\lambda_3 t})}{(\lambda_2 - \lambda_1)} \]

\[- \frac{(-\lambda_1 t - \lambda_3 t)}{(\lambda_2 - \lambda_1)(\lambda_3 - \lambda_1)} + \frac{(-\lambda_1 t)}{(\lambda_1 - \lambda_3)(\lambda_2 - \lambda_1)} \]

\[- \frac{(1 - e^{-\lambda_1 t})}{(\lambda_2 - \lambda_1)} + \frac{CFY(2)}{2 \text{IFY}(1)} (1 - e^{-\lambda_1 t}) + \frac{(-\lambda_1 t - \lambda_2 t)}{(\lambda_1 - \lambda_1)} \]
Then apply $A_3(td)$ from eq. (17) to equation (15) and integrate, we get

$$C(\text{counts}) = \varepsilon \cdot I_Y \cdot N \cdot \phi_{th} \cdot \sigma_{th} \cdot \text{CFY}(3)(1 - e^{-\lambda_3 t})$$

$$= \frac{e^{-\lambda_3 t}}{\lambda_3} (1 - e^{-\lambda_3 t}) + \frac{\text{CFY}(2)}{\lambda_2 \text{IFY}(1)} (e^{-\lambda_2 t} - e^{-\lambda_3 t})$$

$$- \frac{(e^{-\lambda_2 t} - e^{-\lambda_3 t})}{(\lambda_2 - \lambda_1)} - \frac{\lambda_1 t}{(\lambda_2 - \lambda_1)(\lambda_3 - \lambda_1)}$$

$$+ \frac{(\lambda_2 - \lambda_3)(1 - e^{-\lambda_1 t})}{(\lambda_1 - \lambda_3)(\lambda_2 - \lambda_1)} - \frac{(1 - e^{-\lambda_1 t})}{(\lambda_2 - \lambda_1)}$$

$$+ \frac{\text{CFY}(2)}{\lambda_2 \text{IFY}(1)} (1 - e^{-\lambda_2 t}) + \frac{e^{-\lambda_1 t} - e^{-\lambda_2 t}}{\lambda_1 - \lambda_2}$$
From equation (18) the required CFY(3) was extracted and found
where other variables were known figures or experimentally
measured numbers.
APPENDIX - C

Case of 218 Kev data (Doublet)

1) Two components:

190.8 sec 89 Kr and 39.7 sec 139 Xe mixed together

2) They can be decomposed to linear correlation

\[ A(t) = C(t) K = A_o e^{-\lambda t} \]

\[ C(t) = \frac{A_o}{K} e^{-\lambda t} \]

\[ \log(C(t)) = \log \frac{A_o}{K} + \log e^{-\lambda t} \]

or \( y = K_1 + (-\lambda t) \log e \)

\[ = K_1 - \frac{\lambda}{2.301} t \]

\[ = K_1 - \frac{0.693}{2.301 \frac{t}{T_{1/2}}} \]

\[ = K_1 - \frac{0.301}{T_{1/2}} X \]

OR \( Y = mX + b \)

where, \( m = - \frac{0.301}{T_{1/2}} \)

3) Solve b (intercept), \( m = - 0.301/190.8 = - 0.001578 \text{ S}^{-1} \)

<table>
<thead>
<tr>
<th>Count Rate C</th>
<th>(Y = \log C)</th>
<th>(X)</th>
<th>(b = Y - mx)</th>
<th>(W_i = 1/\Delta b^2)</th>
</tr>
</thead>
<tbody>
<tr>
<td>15.7±6%</td>
<td>1.196±6%</td>
<td>307</td>
<td>1.680±6%</td>
<td>(b_1)</td>
</tr>
<tr>
<td>10.6±9%</td>
<td>1.025±9%</td>
<td>389</td>
<td>1.639±9%</td>
<td>(b_2)</td>
</tr>
<tr>
<td>9.40±9%</td>
<td>0.973±9%</td>
<td>471</td>
<td>1.716±9%</td>
<td>(b_3)</td>
</tr>
<tr>
<td>6.12±13%</td>
<td>0.787±13%</td>
<td>553</td>
<td>1.660±13%</td>
<td>(b_4)</td>
</tr>
</tbody>
</table>

\(\Sigma W = 207.8\)
\[ b \equiv \Sigma \text{biwi/\Sigma wi} = 348.3/207.8 = 1.676 \]

\[ \Delta b \equiv [\Sigma \text{wi}]^{-1} = 0.069 \]

\[ b \pm \Delta b = 1.676 \pm 4.1\% \text{ or } \log C_\circ = 47.4 \pm 1.9 \text{ cps} \]

(4) Solve count rate of \( ^{139}\text{Xe} \)

<table>
<thead>
<tr>
<th>Total C(cps)</th>
<th>C(cps) of ( ^{89}\text{Kr} )</th>
<th>C(cps) of ( ^{139}\text{Xe} )</th>
<th>( X )</th>
</tr>
</thead>
<tbody>
<tr>
<td>153.6(5.0)</td>
<td>37.2(15)</td>
<td>116.4(52)</td>
<td>67</td>
</tr>
<tr>
<td>118.6(45)</td>
<td>34.2(14)</td>
<td>84.4(47)</td>
<td>90</td>
</tr>
<tr>
<td>93.5(42)</td>
<td>31.6(13)</td>
<td>61.9(44)</td>
<td>112</td>
</tr>
<tr>
<td>64.0(38)</td>
<td>29.2(12)</td>
<td>34.8(40)</td>
<td>113</td>
</tr>
<tr>
<td>56.4(29)</td>
<td>27.8(11)</td>
<td>29.6(31)</td>
<td>147</td>
</tr>
<tr>
<td>31.6(26)</td>
<td>23.8(10)</td>
<td>7.8(28)</td>
<td>190</td>
</tr>
<tr>
<td>27.8(23)</td>
<td>20.4(8)</td>
<td>7.4(24)</td>
<td>232</td>
</tr>
</tbody>
</table>
Figure 13 The Decay of 39.7 sec $^{139}$Xe & 191 sec $^{89}$Kr.
APPENDIX - D

Figures of the Decay of the Measured Isotopes
Figure 14 The Decay of $^{90}_{\text{Kr}}$ for $E^*=121.5$ KeV & $T_{1/2} = 32.3$ s
Figure 15 The Decay of $^{136m}$I for $E_x=381.4$ KeV & $T_1=45$ sec.
Figure 16 The Decay of $^{137}$Xe for $E_\beta=455.5$ KeV & $T_1=229$ s.
Figure 17  The Decay of $^{140}$Cs for $E_{0}=602.4$ KeV & $T_{1/2}=66$ sec
Figure 18  The Decay of $^{144}$La for $E_x = 397.4$ KeV & $T_1/2 = 40$ se
Figure 19  The Decay of $^{148}$Ce for $E^*=269.52$ KeV & $T_1=56$ sec.
Figure 19 The Decay of $^{148}$Ce for $E=269.52$ keV & $T_1=56$ sec
REFERENCES


5. Operation and Maintenance Manual (1 MW Open Pool Reactor) for National Tsing Hua University, General Electric (1960)


