SPACE-TIME REACTOR KINETICS STUDIES WITH THE UNIVERSITY OF FLORIDA SPERT ASSEMBLY

By
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PREFACE

The text of this dissertation is divided into two related but essentially independent parts:

Part 1. The University of Florida SPERT Assembly
   - Design and Calibration -

Part 2. Space-Time Reactor Kinetics Studies with the University of Florida SPERT Assembly

In Part 1, the design, operational safety and nuclear calibration of the large-in-one-space dimension, highly multiplicative University of Florida SPERT Assembly (UFSA) are described. The presentation of this material is pertinent for a complete understanding of the physical characteristics of the system to be studied in Part 2 and also because the system in itself is interesting from the nuclear engineering point of view. The data acquisition system used for the experimentation is presented in Chapter IV of this section.

In Part 2, the linear reactor kinetics studies performed with the UFSA are presented. The space-time dependent distribution of the neutrons in the assembly, following the introduction of a burst of neutrons at one end of the core, are studied in the time and in the frequency domain.

The study of the spatially dependent time profile of the neutron flux required a large number of figures containing the calculational and experimental results at different positions in the core. Some typical figures are included in the main text but most of the recorded (and calculated) time profiles are included as appendices to the main text.
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LIST OF SYMBOLS

$L^2$ .......... TRANSVERSE BUCKLING

$c_i$ .......... DELAYED NEUTRON PRECURSOR CONCENTRATION OF THE $i^{th}$ GROUP

$D_i$ .......... DIFFUSION COEFFICIENT OF THE $i^{th}$ GROUP

$f$ .......... FREQUENCY (cps)

$k, k_{\text{eff}}$ .... EFFECTIVE MULTIPLICATION CONSTANT

$\tau$ .......... NEUTRON LIFETIME

$M$ .......... NEUTRON MULTIPLICATION

$v_i$ .......... VELOCITY OF THE $i^{th}$ GROUP

$z$ .......... AXIAL COORDINATE

$\alpha$ .......... DECAY CONSTANT (IN THE TIME DOMAIN)

$\alpha$ .......... DAMPING COEFFICIENT (IN THE FREQUENCY DOMAIN)

$\beta$ .......... EFFECTIVE DELAYED NEUTRON FRACTION

$\xi$ .......... PHASE SHIFT PER UNIT LENGTH

$\rho$ .......... REACTIVITY

$\rho$ .......... COMPLEX INVERSE RELAXATION LENGTH

$\Sigma$ .......... MACROSCOPIC CROSS SECTION

$\phi$ .......... NEUTRON FLUX
Abstract of Dissertation Presented to the Graduate Council
in Partial Fulfillment of the Requirements for
the Degree of Doctor of Philosophy

SPACE-TIME REACTOR KINETICS STUDIES WITH
THE UNIVERSITY OF FLORIDA SPERT ASSEMBLY

By

Nils J. Diaz

March 1969

Chairman: Dr. M. J. Ohanian
Major Department: Nuclear Engineering Sciences

A large-in-one-space dimension, side reflected, highly multiplicative \( k_{\text{eff}} \sim 0.99 \) subcritical assembly was designed and calibrated. The sole purpose of the facility is the experimental investigation of the dynamic behavior of large reactor cores and to provide a test for space-time kinetics models presently in use. With this facility the linear aspects of neutron physics phenomena can be investigated in the absence of inherent feedback effects. This work was conducted under a subcontract with the Nuclear Safety Research Branch, Atomic Energy Division, Phillips Petroleum Company, under a prime contract with the United States Atomic Energy Commission.

The University of Florida SPERT Assembly (UFSA) is a light-water moderated subcritical facility fueled by 4.81\% enriched \( \text{UO}_2 \) pellets encased in stainless steel tubes of 0.465 inch outside diameter (SPERT F-1 Fuel). The fuel arrays are contained in a rectangular tank, 8 feet
long, 39 inches high, and of variable width. In this study, the core was 6.5 inches wide and 30 inches high. The effective multiplication constant of the assembly was determined to be $0.990 \pm 0.003$. The assembly is equipped with nuclear instrumentation capable of automatic scram action.

For the kinetics studies, a fast data acquisition system was developed to handle accurately the very high, time-changing count rate encountered in the measurements. It essentially consists of a transformer-coupled pulse amplifier to produce a fast logic signal at the input of a multichannel analyzer from the input signal originating in a long, thin He$^3$ counter. The instrumentation adequately handled count rates up to $3 \times 10^6$ counts/sec at the peak of the pulses. A high degree of reproducibility and fidelity in following the pulse profiles was obtained with this instrumentation.

The space-time kinetics studies were performed by analyzing the propagation of a fast neutron burst introduced at one end of the assembly, in the absence of inherent feedback effects. The experimental results are compared with the results obtained from the two-group, space-time dependent, one-dimensional diffusion theory scheme known as the WIGLE program. A stringent test of the model is provided by a combined analysis in the time and the frequency domain.

The WIGLE calculational scheme accurately predicts the delay times and the attenuation of the pulses when a first-flight spatial distribution is assumed for the fast source. At large distances from the source WIGLE underpredicts (~ 8% in the FWHM) the spreading of the pulse. A marked sensitivity to small changes in the transverse buckling was found for the model, as well as the experiment.
A one-to-one comparison of the predicted and measured values in the frequency domain was provided by performing identical numerical Fourier transformations of the WIGLE time profiles and the measured pulse shapes. The analysis in the frequency domain confirmed the results obtained in the time domain, although discrepancies past 100 cps are found in the ultrasensitive $\rho^2$ plane. The agreement in the $\rho$ plane, the system's dispersion law, is good up to 200 cps and reasonable up to 800 cps. Both theory and experiment showed a smooth behavior throughout the frequency range investigated, in both the $\rho$ and the $\rho^2$ plane.

Spatial effects in large cores are clearly demonstrated in this work. The determination of the range of applicability of the one-dimensional scheme requires extending the study to cases in which two-dimensional effects will be noticeable and the important feedback effects can be considered.
PART 1

THE UNIVERSITY OF FLORIDA SPERT ASSEMBLY
- DESIGN AND CALIBRATION -
CHAPTER I

INTRODUCTION

The development and construction of large power reactors focused the attention of industry and of the United States Atomic Energy Commission on the necessity of having reliable reactor dynamics analysis methods to accurately describe the spatial and temporal behavior of the neutron flux in these systems. The point-model reactor kinetics calculations seem to have been adequate for the gross evaluation of the time-dependent neutron flux during the occurrence of a transient but the model can be in large error when the physical size of the system and the magnitude of the perturbation necessitates that spatial effects in the redistribution of the neutron flux be considered. Preliminary calculations done by Johnson and Garner using a one-dimensional space-time kinetics model [1] showed that the space-time dependent scheme predicts a "destructive zone" much larger than that predicted by point-model kinetics.

The necessity of experimentally determining the validity of the various space-time kinetic analysis methods was brought out by Johnson and Garner [1], and recognized by the USAEC in establishing the Large Core Dynamics Experimental Program. The primary responsibility for this program has been vested in the Nuclear Safety Research Branch, Atomic Energy Division, Phillips Petroleum Company as major contractor for USAEC.
The Large Core Dynamics Experimental Program is to be performed in three phases:

Phase I. Pulsed Source Experiments in Subcritical, Multiplying Media, Large in One-Dimension

The first phase of the experimental program is to be conducted in a close-to-critical subcritical assembly, 8 feet long, 3 feet deep and with widths changing from 6.5 inches to 16 inches according to the core configuration and whether bare or side reflected cores are studied.

The experimental information obtained from studying the pulse propagation phenomena in this assembly is to be used to test the validity of current space-time kinetic models in the absence of inherent feedback effects.

Phase II. Kinetic Behavior for Control-Rod-Induced Power Excursions in Large, One-Dimensional Cores

A reactor large in one-dimension, 16 feet long, three feet deep and with varying widths to accommodate different metal/water ratios will be used to investigate the one-dimensional kinetic behavior of large cores subjected to a large perturbation. Both non-feedback (low power experiments) and self-shutdown measurements will be conducted.

Phase III. Kinetic Behavior for Control-Rod-Induced Power Excursions in Large, Two-Dimensional Cores

The same type of measurements performed for the one-dimensional core will be conducted in a two-dimensional core.

The measurements should provide the necessary information to establish the validity ranges for one-dimensional models, the basis for the development of a two-dimensional scheme and as a bridge to the complex, three-dimensional problem.

Phase II and Phase III of the research program will be performed
at the SPERT IV facility at the National Reactor Testing Site, Idaho.

The appropriate existing experimental equipment, as well as the extensive kinetics studies conducted by the Nuclear Engineering Sciences Department at the University of Florida, was conducive to the granting of a subcontract by the Phillips Petroleum Company so that the basic, linear kinetics studies of Phase I could be performed at the University of Florida.

The research to be performed as Phase I of the Large Core Dynamics Experimental Program can be succinctly defined as the experimental and analytical determination of the dynamic behavior of the neutron flux in slightly subcritical water moderated assemblies of SPERT F-1 fuel rods.

The facility in which the required measurements for Phase I, Large Core Dynamics Experimental Program, are to be conducted necessitated a thorough design and safety analysis. The assemblies are to be close to critical and the core has a large U235 inventory. The nuclear capabilities of such systems were the object of a detailed study to determine their operational characteristics under normal and accident conditions.

The flexible mechanical design, the safety instrumentation, the nuclear evaluation, as well as the experimental calibration of the first configuration under study constitutes Part 1 of this dissertation.

The reactor kinetics studies performed in the first of the configurations to be studied are dealt with in Part 2 of this manuscript.
CHAPTER II

DESCRIPTION OF THE FACILITY

General Features

The University of Florida Spert Assembly is a light water-moderated subcritical facility fueled by 4.81% enriched UO$_2$ pellets encased in stainless steel tubes of 0.4655" outside diameter. The fuel arrays are contained in a rectangular tank, 8 feet long, 39 inches high, and of variable widths. The system is designed so that both bare and reflected cores can be studied. Only one reflected core will be dealt with in Part 2 of this manuscript; information on three reflected cores is included in this chapter. The assembly width and fuel spacing may be varied in order to:

a) have a $k_{\text{eff}}$ not to exceed 0.99 in all cases to be considered.

b) accommodate non-moderator/moderator ratios of 0.5, 1.0, and 1.5, respectively.

Shown in Table I are the $k_{\text{eff}}$'s as a function of the moderator height, the fuel spacings, core widths, and total number of fuel elements for the different reflected cases to be considered. The calculational procedures used in the determination of the nuclear parameters and the $k_{\text{eff}}$ values for the three reflected configurations of the assembly are described in Appendix A. Only the sides of the assembly will be reflected. Fig. 1 shows an overall view of the facility.
FIG. 1 OVERALL VIEW OF THE FACILITY
TABLE I

\( k_{\text{eff}} \) vs. MODERATOR LEVEL OF UFSA REFLECTED CORES

Calculated Using the AIM6 Code
Core Length = 243.8 cm (96")
Reflector Width = 30.48 cm (12")
Active Fuel Height = 91.4 cm (36")

<table>
<thead>
<tr>
<th>Metal to Water Ratio</th>
<th>0.5</th>
<th>1.0</th>
<th>1.5</th>
</tr>
</thead>
<tbody>
<tr>
<td>Lattice Pitch (in)</td>
<td>0.7152</td>
<td>0.584</td>
<td>0.5332</td>
</tr>
<tr>
<td>Core Width (cm)</td>
<td>16.35</td>
<td>19.28</td>
<td>25.73</td>
</tr>
<tr>
<td>No. of Fuel Rods</td>
<td>1206</td>
<td>2132</td>
<td>3420</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Moderator Level (cm)</th>
<th>Effective Multiplication Constant</th>
</tr>
</thead>
<tbody>
<tr>
<td>20</td>
<td>.7695</td>
</tr>
<tr>
<td>25</td>
<td>.8288</td>
</tr>
<tr>
<td>30</td>
<td>.8715</td>
</tr>
<tr>
<td>35</td>
<td>.9022</td>
</tr>
<tr>
<td>40</td>
<td>.9249</td>
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<td>45</td>
<td>.9420</td>
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<tr>
<td>50</td>
<td>.9551</td>
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<tr>
<td>55</td>
<td>.9655</td>
</tr>
<tr>
<td>60</td>
<td>.9738</td>
</tr>
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<td>65</td>
<td>.9805</td>
</tr>
<tr>
<td>70</td>
<td>.9870</td>
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<tr>
<td>75</td>
<td>.9906</td>
</tr>
<tr>
<td>80</td>
<td>.9944</td>
</tr>
<tr>
<td>85</td>
<td>.9977</td>
</tr>
<tr>
<td>91.4</td>
<td>1.0012</td>
</tr>
</tbody>
</table>

No. of Fuel Rods = 1206 for all entries.
The assemblies are highly multiplicative; this is important for the extrapolation of the results of the study to critical systems. The system's subcriticality is attractive because of the inherent safety of such systems and of the absence of inherent feedback effects.

In order to provide as "clean" a core as possible, a unique control system which has been successfully used on the UFAPA [2] will be employed. In this system the reactivity is controlled by adjustment of the water height in the assembly. The water height is controlled by the position of two "V"-notched weirs located in a water "box" hydraulically coupled with the assembly through flexible lines. The quantity of water discharging through a "V"-notched weir varies as $H^{5/2}$ ($H$ is the distance between the apex of the weir and the water level) thus providing precise control of the moderator height. The hydraulic coupling assures that under normal operating conditions (with continuous flow) there will be the same water level in the core and the reflector tanks.

The UFSA subcritical assembly is located in an isolated and shielded room in the Nuclear Research Field Building, approximately three miles from the University campus.

The Nuclear Research Field Building consists of four bays, two of them having shielded rooms for experiments with subcritical and moderating assemblies. The shielded walls consist of stacked concrete block eight feet high and thirty-eight inches wide covered with plywood to assure that the blocks remain in place. The ceiling of this single-floor building is approximately 15 feet above the floor and consists of excelsior-filled cement bonded board. Neutron reflection from this ceiling over the walls does not constitute a hazard to personnel operating the accelerator-type neutron source. The access door from the control room
is interlocked with the neutron generator and the subcritical assembly scram system, as is the door on the only other entrance to the shielded room from the fuel storage area. Across the front face of the assembly, a screened wire cage with a lockable door controls access to the core.

While not in use in the assembly the fuel is stored in a room adjacent to the facility, built entirely for this purpose.

A more complete description of the facility and its characteristics can be found in the Design and Hazards of the UFSA and its addenda [3, 4, 5].

The system has been licensed under Atomic Energy Commission SPECIAL NUCLEAR MATERIAL LICENSE SNM 1050, March 1968. The license allows for the possession of 5400 fuel rods with a total U235 inventory of 190 kgs.

**Fuel Characteristics**

The UFSA is fueled with Spert F-1 type fuel elements provided by the Phillips Petroleum Company.

The fuel characteristics are:

Fuel Composition: \( \text{UO}_2 \) in pellet form

\( \text{U}_{235} \) enrichment: \( 4.81 \pm 0.15\% \)

Active Fuel Length: 36" \( \pm 0.062" \)

Active Fuel Diameter: .42" \( \pm 0.005" \)

Fuel Tube Material: stainless steel

Fuel Tube Length: 41.625"

Fuel Tube o. d.: .4655" \( \pm 0.0025" \)
Mechanical Design

The entire assembly can be divided into three components: the supporting platform and dump tank; the basic core tank and fuel rod support structure; and the combination core side walls and side reflector tanks.

The supporting platform is composed of 5 inch steel I-beams, raised 5 feet from the floor level by six steel columns 3 1/2 inches in diameter. The column footings rest within a 6x8x2 foot steel rank which serves as a reservoir for the continuous water flow system and as a dump tank. Under normal operating conditions, this represents a minimum distance of about 4 feet between the bottom of the assembly and the water surface in the reservoir. This distance is sufficiently large so that the bottom of the assembly is considered to be unreflected under all conditions.

All the core and reflector hardware is made of type 5456-H321 aluminum. The bottom of the basic tank is made of a 24x96x3/4 inch plate bolted to the steel I-beam platform. The underside of the plate is covered with a .030 inch thick Cadmium sheet. The lower fuel rod support assembly rests on the plate. The end walls of the basic tank are made of a 24 x 39 3/4 inch plate and are supported by two 2 1/2 x 2 1/2 x 1/4 inch steel angle braces welded to the I-beam platform. Two four-inch aluminum channels span the eight foot dimension of the tank, confining the upper fuel rod support system and detector mounts.

The side walls of the basic tank serve also as reflector tanks when the reflected cores are under study. These tanks have dimensions of 12 x 96 x 37 inches. The arrangement allows one to vary the width of the core with a sole structural support.
The end walls are permanently covered with Cadmium on the outside surface while the side walls have movable Cadmium covers to define the boundaries for the bare and reflected cases. To optimize the number of neutrons inserted into the assembly by the neutron generator, the accelerator target penetrates about 4 inches into the core. A water-tight port is provided for this purpose. The port can be removed and a blind flange inserted in its place. Several fuel rods must be taken out, the number depending on how deep the target goes into the assembly and on the lattice pitch.

The core section of the assembly consists of an interchangeable fuel rod spacing system made of 3/4 x 1/2 x 1/8 inch channels, 5/8 x 1/4 inch bars and aluminum shims mounted on the base plate of the tank. The bars have milled slots to accommodate the .25 inch end tip of the fuel rod and to set the pitch along the core width. The shims are placed between the channel bar units to set the pitch along the core length (96 inches) (see Fig. 2). The top fuel rod spacing system consists of an aluminum grating. The mesh is determined by the lattice pitch under study. The grating is made of aluminum bars and spacers, as shown in Fig. 3. Thus, fuel rod removal along the length of the core is possible to locate the detector for the experimental measurements.

The one-half inch long rod tip is fully surrounded by aluminum, with practically no reflecting characteristic, but there is a 7/8 inch length of rod between the end of the active fuel and the tip which is surrounded by water. This bottom reflector is unavoidable and will be considered in the calculations.
FIG. 2A  BOTTOM FUEL ROD SPACING SYSTEM
FIG. 2B BOTTOM FUEL ROD SPACING SYSTEM
FIG. 3  TOP FUEL ROD SPACING SYSTEM
Moderator Flow Control System

The moderator flow control system of the UFSA can be better described by the water flow schematic shown in Fig. 4. Besides the normal fill and drain functions for the moderator, it serves as an accurate reactivity control using adjustable moderator height by continuous flow.

The characteristic components of this system are described below.

A. Storage Tank: A 6 x 8 x 2 foot steel tank located directly below the assembly will serve as the reservoir for the circulating light-water moderator and as a dump tank. Normal water heights while operating will be between 6 and 12 inches. The tank also serves as a footing for the assembly supports. This arrangement makes a very convenient and compact facility.

B. Core and Reflector Tanks: As seen from the flow diagram, water is pumped from the reservoir to the core and reflector tanks through a manifold at one end of the assembly and flows from the other end of the core and reflectors tanks to the weir "box". From the weir "box", water flows over the weirs back to the storage tank through a flexible line.

The core section is equipped with two normally open solenoid activated dump valves, 3 inches in diameter, located at each end of the core. These valves provide the reactor with a fast shutdown safety system. The reflector tanks have their own 1 1/2 inch normally open solenoid valves actuated by the same safety system.

Since the quantity of water discharging through a V-notched weir varies as $H^{5/2}$ where $H$ is the height of the water level above the apex of the V-notch, the water level and hence, the reactivity, can be controlled in a precise manner simply by varying the height of the weirs and the rate of flow of water into the tank. This is accomplished by
FIG. 4  REACTIVITY-CONTROL FLOW SYSTEM
an automatically operated pneumatic control valve. A plot of the flow rate versus the height of the water level above the apex of the weirs is shown in Fig. 5.

The weir plate is rigidly mounted on a "box" or small tank (see Fig. 6). The weir "box" is connected to a drive mechanism composed of the following: guide post, slide block, and drive screw. The guide post is a 2 inch diameter pipe attached to the support column of the crane which is used for removal of the reflector tanks. The "box" is mounted on the guide block which slides along the vertical post and provides a rigid support for the system and is driven up and down by means of the drive screw which is fixed at the top of the guide post support and passes through the guide block. The upper limit of the position of the weir "box" is controlled by mechanical stops whose position is determined as part of the initial start-up procedure for each configuration to be considered.

The final adjustment of the position of the weir "box" is such that when the water reaches the moderator level in the assembly corresponding to $k_{eff} \leq 0.99$ for a given configuration it will be flowing about 2.0 inches above the apex of the V-notch weirs. At this design level the flow rate is ~ 7 gallons per minute with the $k_{eff}$ values as given in Table I for the full fuel loading. After the operating height of the weir box has been determined for $k_{eff} < 0.99$ in the initial start-up, stops are inserted to prevent raising the weirs above this height (if the height is less than the active fuel height). It should also be pointed out that the orifice in the line limits the pump capacity to a flow rate which is just sufficient to bring the weirs to full flow. A further increase in flow rate would cause discharge over the entire
FIG. 5 FLOW RATE vs. HEIGHT OF WATER LEVEL ABOVE WEIR APEX
FIG. 6 WEIR "BOX"
perimeter of the weir "box" into the drain line effectively preventing any further increase of the moderator level in the assembly.

The measurement of the water height in the core is accomplished by fixing a reference mark on the slide block at the same level as the bottom of the weir within the "box". An accurate scale is provided to read off the distance between the bottom of the core and the apex of the weirs. Continuous indication of the moderator level in the core is provided on the console by means of a recorder calibrated between the bottom of the active fuel and the maximum moderator height and by a manometer, connected directly to the core, for precise measurement of the water level. These two measuring systems insure reproducibility of the moderator height for the experiments.

The water is pumped out of the storage tank by a constant speed centrifugal pump which has a "no load" capacity of 20 gal/min. The control valve is designed to restrict the flow to the maximum design value of 12 gal/min. A deionization system is provided to keep the water as pure as possible at all times. The pneumatic flow control system consists of two differential pressure cells, transmitters, control valve, and recorder-controller. The strip type chart recorder-controller records both flow rate and moderator level in the core. The control valve is of the air-to-open type which will close in the case of air supply loss, stopping the flow into the assembly. The flow diagram for the air system is shown in Fig. 7. A pressure differential from the pressure transmitters applied to the recorder-controller allows both manual and automatic control of the flow rate through the valve operated by the controller.
FIG. 7 AIR SYSTEM SCHEMATIC
Instrumentation and Interlock System

The instrumentation and interlock system of the UFSA has been discussed extensively in the reports submitted to the Atomic Energy Commission [3, 4, 5] in conjunction with the license application. More recently, Mr. L. B. Myers submitted a detail technical report on the subject [6]. A brief descriptive explanation is given below.

A block diagram of the safety system logic flow in use at the UFSA subcritical assembly for routine monitoring is shown in Fig. 8. There are five principal channels of instrumentation:

a. Start-up channel using a He$^3$ proportional counter, scaler, and rate meter. The counter is located at the bottom of the core, close to the geometrical center of the assembly.

b. Log power and period instrument No. 1 channel using a compensated ion chamber (operated uncompensated) as a signal to a Log N amplifier. The chamber is located along the longitudinal axis of the assembly, on the bottom of the core some two feet from the neutron generator end.

c. Period instrument No. 2 channel using a compensated ion chamber (operated uncompensated) as a signal to a log N amplifier. The chamber is located along the longitudinal axis of the assembly, on the bottom of the core, some six feet from the neutron generator end.

d. Linear neutron flux No. 1 channel using an uncompensated ion chamber as a signal to a micromicroammeter. The chamber is mounted on the top core support frame, close to the geometrical center of the core.

e. Linear neutron flux No. 2 channel using a compensated ion chamber (operated uncompensated) to feed a signal to a micromicroammeter. The chamber is mounted on the top core support frame on the opposite
FIG. 8 UFSA SAFETY SYSTEM LOGIC FLOW DIAGRAM
side from the linear channel No. 1.

Items a. through d. are part of the safety amplifier while item f. is used to display the neutron flux on a console front panel meter.

The safety amplifier monitors the seven continuously varying input signals and provides a trip signal if any of the input signals fall outside of acceptable limits. The safety amplifier provides means of adjusting these limits over a wide range.

The duality of the scram action (see Fig. 8) is a prominent feature of the safety system. It can be said that no single failure will invalidate both automatic scram channels. Furthermore, it has been determined that no single failure can invalidate both the manual and automatic scrams. The method of measurement and the function of each instrumentation and safety channel are shown in Table II (parts A and B).

A series of safety interlocks prevent water from flowing into or remaining in the assembly unless a proper sequence of events are followed and certain conditions are satisfied. The conditions are:

a. The moderator temperature must be $\geq 60^\circ F$. This is established by the desire to obtain the experimental data near room temperature conditions. The insertion of water at $32^\circ F$ will introduce a maximum $k$ of $.00342$ (based on the calculated negative temperature coefficient of reactivity) above the design $k_{eff}$ value with no hazards created.

  b. The four instrumentation channels must have their high voltage on.

  c. The core width must be smaller than 26 cm.

  d. The door to the assembly room must be locked.

  e. The start-up channel must count more than 2 counts/sec.

  f. The neutron flux, subcritical assembly power level and period
### TABLE II

**UFSA INSTRUMENTATION AND CONTROL**

**A. NUCLEAR**

<table>
<thead>
<tr>
<th>Measured Parameter</th>
<th>Method of Measurement</th>
<th>Application</th>
</tr>
</thead>
<tbody>
<tr>
<td>a. Low level neutron flux</td>
<td>$^3$He detector – pulse discriminator; at neutron generator end of core</td>
<td>Insure source is present before adding reactivity. Scram on low count rate</td>
</tr>
<tr>
<td>b. Linear neutron flux</td>
<td>CIC$^a$ – ammeter; on side of core</td>
<td>Indicate power level scram on power</td>
</tr>
<tr>
<td>c. Log neutron flux$^b$</td>
<td>CIC$^a$ – log N and period amp; under core near center line</td>
<td>Indicate power level scram on high power; log N recorder</td>
</tr>
<tr>
<td>d. Linear neutron flux</td>
<td>UIC – ammeter; on side of core</td>
<td>Indicate reactor power; linear N recorder</td>
</tr>
<tr>
<td>e. Reactor period 1$^b$</td>
<td>CIC$^a$ – log N and period amp</td>
<td>Indicate reactor period; scram on short period</td>
</tr>
<tr>
<td>f. Gamma flux</td>
<td>Ion chamber – area monitor; on front of reactor cage</td>
<td>Criticality monitor for storage room. Area monitor for reactor room. Activate evacuation alarm</td>
</tr>
<tr>
<td>g. Detector power supply voltage</td>
<td>Unijunction transistor oscillator and relay. Monitor detector voltage for b, c, d and e</td>
<td>Scram reactor on low detector voltage</td>
</tr>
<tr>
<td>h. Reactor period 2</td>
<td>CIC$^a$ – log N and period amp; under core near center line</td>
<td>Indicate reactor period; scram on short period</td>
</tr>
</tbody>
</table>

$^a$ To be operated in the uncompensated mode

$^b$ Common detector and instrument
### TABLE II (Continued)

#### B. NON-NUCLEAR

<table>
<thead>
<tr>
<th>Measured Parameter</th>
<th>Method of Measurement</th>
<th>Application</th>
</tr>
</thead>
<tbody>
<tr>
<td>a. Reactor water temperature</td>
<td>Fenwall temperature switch in inlet line</td>
<td>Scram reactor on low reactor inlet water temp. (60°F)</td>
</tr>
<tr>
<td>b. Reactor water level</td>
<td>Barksdale pressure switch mounted on weir&quot;box&quot;with sensing line connected to core</td>
<td>Scram reactor if water level in core exceeds top of weir height</td>
</tr>
<tr>
<td>c. Reactor door and personnel</td>
<td>Limit switches on doors and push buttons inside reactor room</td>
<td>Scram system and shut down neutron gun if reactor doors are opened or interior switches are activated</td>
</tr>
<tr>
<td>d. Reactor core width</td>
<td>Limit switches on reflector tanks</td>
<td>Prohibit filling reflector tanks when distance between tanks exceeds widest reflected core width</td>
</tr>
<tr>
<td>e. Reactor water level</td>
<td>Barnstead pressure switch senses level in weir box</td>
<td>Stops pump when water reaches 11cm below weir apex</td>
</tr>
<tr>
<td>f. Reflector tank water level (low)</td>
<td>Float switches in reflector tanks</td>
<td>Indicates water is filling reflector tanks</td>
</tr>
<tr>
<td>g. Flow control valve shut</td>
<td>Limit switch on valve</td>
<td>Requires closing valve before starting pump</td>
</tr>
<tr>
<td>h. Reactor flow</td>
<td>Differential pressure cell and pneumatic control</td>
<td>Control flow rate. Indicate and record flow rate</td>
</tr>
<tr>
<td>i. Reactor water level</td>
<td>D/P cell and pneumatic system</td>
<td>Indicate and record water level</td>
</tr>
</tbody>
</table>
must be as specified under Operating Limits in this report.

g. After a normal start-up, the water height in the core must be within 0.5" of the level set by the position of the weirs.

**Fuel Storage**

The large amount of fuel needed for the experimental program required a detailed criticality analysis of the fuel storage area. Criticality considerations of the fuel storage arrangements follow the Atomic Energy Commission regulations regarding the subject. Three different criteria were used to calculate the effective multiplication of the fuel storage area to assure that the array will remain subcritical under the worst circumstances. The methods and the corresponding conditions are outlined below.

The fuel storage array consists of three slabs of air-spaced fuel pins, separated by a minimum distance of 54 inch face to center. The fuel is stored in steel baskets containing 308 pins per basket. Two sets of 1/4 inch thick plastic plates, located at the bottom and top of each basket, drilled to properly position the fuel rods. The characteristics of the fuel slab are:

- Slab Width = 3.73 in = 9.46 cm (corresponds to 7 fuel pins in transverse direction)
- Slab length = 14 ft
- Height = 3 ft (active fuel height)

a. **Multiregion-multigroup calculation**

The effective multiplication factor of the fuel storage array consisting of three slabs (3.73 inch wide, 14 ft long and 3 ft wide) which are 54 inch apart (face to face) was computed for the case of flooding
the storage area to the level of the active fuel height. A 2-foot reflector on both sides was used to represent an infinite reflector. No reflector was considered on the ends, but the contribution of this to the system would be small. The calculation was done using four groups and seven regions and followed the method outlined in Appendix A of this thesis. The following configuration, which is symmetric about the indicated center line, was assumed:

![Diagram](image)

Under these water-moderated and reflected conditions, a $k_{\text{eff}} = 0.79$ was obtained.

b. **Solid angle criterion for slabs**

To determine the interaction between the fuel storage slabs in the proposed 3-slab array, the solid angle criterion established in 10 CFR, Part 70, 70.52, paragraph (b) was used. This establishes the maximum
total solid angle subtended by any unit in the array to be 6 steradians if the effective multiplication factor for the individual slabs is less than 0.3, as is the case here.

From 10 CFR 70.52 (d) (2) (i) the minimum required separation distance, i.e., center of one slab to face of adjacent slab, is obtained from:

\[
3 \text{ steradians} = \frac{\text{cross sectional area}}{(\text{separation distance})^2}
\]

This gives, separation distance \( = \left(\frac{14 \times 3}{3}\right)^{1/2} = 3.74 \text{ ft.} \)

We propose to establish a minimum distance between faces of adjacent slabs of 4.5 ft. This gives a total solid angle for the center slab of 3.88 steradian, which is well below the established criteria.

c. Comparison with Clark's criteria

For 5% enriched, 0.4 inch diameter uranium oxide rods with a 190.13 gm/liter U-235 concentration (compared to 4.81% enriched, 0.42 inch diameter uranium oxide rods with a 200 gm/liter U-235 concentration in our case) the following data is obtained from pp. 39 and 59, respectively, of DP-1014:

<table>
<thead>
<tr>
<th>Width of Slab (cm)</th>
<th>Buckling (cm(^{-2}))</th>
</tr>
</thead>
<tbody>
<tr>
<td>Critical</td>
<td>Safe</td>
</tr>
<tr>
<td>11.2</td>
<td>10.4</td>
</tr>
<tr>
<td>0.014945</td>
<td>0.015912</td>
</tr>
</tbody>
</table>

It should be noted that the slab widths quoted on page 39 of DP-1014 are for an infinite water-moderated and reflected slab. From the buckling values given, the critical width of the infinite water-moderated unreflected slab is 25.7 cm; the corresponding safe width is 24.9 cm. When twice the reflector savings for the latter case as given on page 59 of DP-1014 is subtracted, a safe width for the infinite, water-
moderated reflected slab of 10.44 cm is obtained consistent with the 10.4 cm value.

Thus the slab width of 9.46 cm proposed by us compares favorably from the safety viewpoint with the safe width for an infinite, water-moderated and reflected slab and is considerably narrower than the safe width for an infinite, water-moderated and unreflected slab. Within the present context it should also be pointed out that as indicated on page 54 of the Design and Hazards Report [3], no flooding of the storage area seems possible from natural causes.

**Neutron Sources**

Two types of neutron sources were used throughout this work.

1) Two Pu-Be sources mounted in an aluminum cylinder which can be driven remotely through a plastic pipe from a shielded box located in one corner of the facility room to underneath the center of the core. Neon lights provide indication at the console of the position of the sources. These sources, which have a combined yield of $-3.2 \times 10^6$ n/sec are used for start-ups and for the inverse multiplication measurements.

2) A Texas Nuclear Neutron Generator which is used in continuous mode for static measurements and in the pulsing mode for the pulse propagation measurements. The generator is of the Cockcroft-Walton type, TNC Model 150-1H with continuously variable high voltage from 0-150 kv and has been modified to obtain larger currents by removing the einzel lenses and installing a new 22 electrode accelerator tube and gap lense. Pre- and post-acceleration beam deflection produces sharp, low-residual pulses.
The accelerator was used with a 4-5 curie tritium target. The position of the target can be changed to keep the source centered on the target-end of the assembly for any given moderator level.
CHAPTER III

OPERATIONAL SAFETY

Introduction

The University of Florida SPERT Assembly, due to its large size, enriched uranium-oxide fuel and nuclear potentialities required a thorough study of its capabilities, operational characteristics, initial loading procedures and of the behavior of the assembly under accident conditions. The study was part of the requirements established by the Division of Material Licensing of the USAEC prior to the granting of an operating license.

Legally, a subcritical assembly has to comply with regulations under 10 CFR Part 70 "Licensing of Special Nuclear Materials" since no self-sustaining nuclear reaction is envisioned. In the case of the UFSA, however, the Commission felt that certain technical sections of 10 CFR Part 50, which deals with nuclear reactor licensing, should apply and serve as a guide for the design and the safety analysis.

The basic philosophies employed in the design of the system were:

a) The UFSA facility has been designed to remain subcritical under normal operating conditions.

b) The safety instrumentation (see Part 1, Chapter I) has been designed such that a single failure will not invalidate both the manual and automatic scram and will not cause subsequent failures.

c) The design basis accidents were postulated on a single failure
criterion.

d) Operating limits have been set to delineate the normal operating ranges of the assembly.

e) Initial loading procedures have been established to determine the safe operating multiplication factor of each configuration.

A series of administrative controls are necessarily applied to all segments of the experimental program and strictly enforced.

**Initial Loading**

A series of calculations were done to determine which of the two following schemes should be employed for the initial loading of UFSA:

I) Step loading of the fuel from the center out, accompanied by step increases in water level with the usual inverse multiplication determination.

II) Loading all the fuel into the dry tank and proceeding with a careful evaluation of the multiplication as a function of water level.

Since the UFSA core is very loosely coupled as far as the lumped reactivity parameter is concerned, the second method was selected due to the fact that a better determination of the multiplication was possible from a basic moderator height-zero loading inverse counts determination. The slope of the $k_{eff}$ vs. water height curve has a slope substantially smoother than the $k_{eff}$ vs. per cent fuel loading (full water height) curve.

The moderator level control system in operation at the facility provides a extremely reliable and safe mode of adjusting the water level without safety compromises.

The following regulations were followed for the initial fuel
loading, and will be followed for subsequent cores:

1) After the fuel has been loaded, prior to each new incremental change in the water level, the water is drained completely and the weir (and water level scram) adjusted to prevent a level increase beyond the desired value.

2) The first three measurements of the inverse multiplication are obtained at water heights of approximately 20 cm, 25 cm, and 30 cm above the bottom of the active fuel. As shown in Table I, the maximum $k_{\text{eff}}$ calculated for a water height of 30 cm is 0.87. Subsequent filling increments are not to exceed the least of the following:

   a. An increase in water height of 10 cm.
   b. An increase in water height which, by extrapolation of the inverse multiplication curve, would increase the $k_{\text{eff}}$ by one-half of the amount required to make the assembly delayed critical.
   c. An increase in water height which would, by extrapolation of the inverse multiplication curve, result in a $k_{\text{eff}}$ of 0.990. For values of $k_{\text{eff}}$ above 0.95, the $k_{\text{eff}}$ of the assembly are also determined by pulsed source techniques.

3) At each filling step, the measured $k_{\text{eff}}$ of the assembly is compared with the calculated value. If significant deviations of the experimental values from the calculated $k_{\text{eff}}$ vs. water height curves occur, the experiments are to be discontinued, and a detailed analysis of the results obtained performed. If it is determined that the dimensions of the assembly should be changed in order to achieve the desired $k_{\text{eff}}$ at maximum water height, the University of Florida will apply for and obtain written approval from the Atomic Energy Division of the Phillips Petroleum Company before such changes are made.
Operating Limits

A description of the operating limits of the subcritical assembly, including the basis for such limits are listed below.

Effective Multiplication Factor

Specification: the maximum allowable $k_{\text{eff}}$ will be 0.985±0.005. The absolute value of $k_{\text{eff}}$ as well as the slope of the $k_{\text{eff}}$ vs. water height will be carefully measured so as not to exceed the limiting value.

Basis: the upper limit of $k_{\text{eff}} = 0.985±0.005$ is established by: the accuracy with which $k_{\text{eff}}$ can be measured, the reported [7] differences between calculations and experiments in similar cores and the value of the multiplication factor required to make a meaningful study of the dynamic properties of large cores. Comparison [7] between 29 calculations and the corresponding critical experiments (on cores similar to the UFSA) established that an overestimate of $k_{\text{eff}}$ is generally made; the standard deviation for these cases was +0.00175 and the maximum underestimate of the multiplication was for a case yielding $k_{\text{eff}} = 0.9966$, a 0.34% deviation.

The absolute value of $k_{\text{eff}}$ will be determined for water levels yielding a $k_{\text{eff}} > .95$ by pulsed techniques independently of the inverse multiplication measurements. The method to be used is the Garelis-Russell[8] method which, when appropriate corrections are made for the reflector, has been shown to give good results. In this method both $a = \frac{1-k_{\text{eff}}(1-\beta)}{\lambda}$ and $\frac{k_{\text{eff}} \beta}{\lambda}$ are determined; then $k_{\text{eff}}$ may be obtained by independently obtaining $\beta$ or $\lambda$. Since of the two parameters $\beta$ changes the most slowly with $k_{\text{eff}}$, it is valid to use a value based on theoretical calculations.
Reactivity Addition Rate

Specification: The reactivity addition rate is controlled by the water flow rate into the subcritical assembly. The maximum flow is fixed to be 12 gpm. At this maximum flow rate, the rates of addition of water, and consequently or reactivity, computed between water heights of 30 and 45 cm. are:

<table>
<thead>
<tr>
<th>Lattice Pitch</th>
<th>Rate of increase of water level</th>
<th>Δk/cm</th>
<th>Δk/sec</th>
<th>$/sec</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>0.5332&quot;</td>
<td>0.0435</td>
<td>0.0093</td>
<td>0.057</td>
</tr>
<tr>
<td></td>
<td>0.584&quot;</td>
<td>0.0439</td>
<td>0.00865</td>
<td>0.054</td>
</tr>
<tr>
<td></td>
<td>0.7152&quot;</td>
<td>0.0431</td>
<td>0.008</td>
<td>0.049</td>
</tr>
</tbody>
</table>

It should be noted that these reactivity addition rates are a large overestimate compared to the calculated rates at $k_{eff} = 0.98$.

Basis: The maximum rate of addition of reactivity was established by the calculated values of $k_{eff}$ vs. water height and the maximum flow rate. The values specified above constitute an upper limit in the region of interest and are considered to be safe under circumstances.

The flow rate is a function of the capacity of the pump, the orifice and the pneumatic control valve and cannot exceed 12 gpm.

Reactivity Removal Rate

Specification: A conservative value for the reactivity removal rate is taken from the slope of the $k_{eff}$ vs. water height curves near the maximum designed $k_{eff}$ values. Since no difference is detected for the scram times of the three configurations, only one rate of removal will be specified, corresponding to the smallest slope.
Drain rate, including the system

reaction time ....................... 2.65 cm/sec

Rate of removal of reactivity ........ 0.00037 Δk/cm

.................. 0.00098 Δk/sec

.................. 0.140 $/sec

it should be noted that the value of Δk/cm used to specify the reactivity removal rate is ~ 20 times smaller than the corresponding value specified for the reactivity insertion rate.

**Basis:** the rate of drainage from the assembly established the reactivity removal rate. Consistent with the approach taken when specifying the reactivity, the time to drain 45 cm of water from the assembly was measured to establish a lower limit on the rate of decrease of water height. Even under these extreme assumptions, i.e., using the maximum slope of the $k_{\text{eff}}$ vs. water height curve for the insertion rate, the small calculated slope around $k_{\text{eff}} = .99$ for the removal rate and the reduced pressure head, the reactivity removal rate is three times larger than the insertion rate.

**Subcritical Assembly Power Level and Power Level Scram**

**Specification:**

Power Level .......................... 0.5 watt

Power Level Scram ..................... 1.0 watt

**Basis:** with the maximum source strength available and with $k_{\text{eff}} = 0.985$ the maximum average power was calculated to be .130 watts by taking into account the spatial dependence of the flux and a steady source at one end.

Assuming that a reactivity accident occurs, based on the maximum reactivity addition rate specified above, the design basis accident
predicts that from a $k_{\text{eff}}$ of 0.993 it would take approximately 15 seconds to double the power level. Even if the power level indicator were not to scram the system until the power level reached 10 kw, assuming a one second delay time to actuate the scram, the power would increase to only 69 kw by the time the scram actually begins. Based on the reactivity removal rate described above, the power level would then decrease rapidly to a very low value.

**Period Scram**

**Specification:**

Period scram ....................... 15 sec

**Basis:** A positive period will be obtained in the subcritical assembly for any addition of reactivity beyond a given steady state condition. If the maximum reactivity insertion rate of 0.05 $/$sec is considered, the initial positive period is about 50 sec and decreases monotonically with time. The period channel has been determined to respond reliably to periods $\approx 50$ sec. Originally, the period scram was set at 50 sec but repeated scrams caused by the start-up of the neutron generator forced the reduction of the scram period to 15 sec for operational reasons.

**Average Neutron Flux and Neutron Flux Scram**

**Specification:**

Ave neutron flux for

most reactive core ................... $1.5 \times 10^5$ n/cm$^2$ - sec

Neutron flux scram .................. $3.0 \times 10^5$ n/cm$^2$ - sec

**Basis:** the above are based on the average power calculated for the assembly. Doubling of the flux will occur when 0.75 $/$ worth of reactivity is added to the system from any design subcriticality level.

Under these conditions, even at the maximum $k_{\text{eff}}$ status, the facility
will still be subcritical and the instrumentation will have sufficient
time to scram the system before accidental criticality occurs.

**Design Basis Accident Analysis**

Two types of accidents have been postulated to occur in conjunction
with the UFSA experimental program. The design basis accident analysis
will deal with the following two cases: a) a dropped fuel rod accident
in which a fuel rod is broken releasing the fission products accumulated
during previous operation of the assembly; b) an accidental criticality
resulting in a power excursion. It will be shown below that the potential consequences of either accident are well within the radiation dose
limits specified by Title 10 CFR Part 20.

**Dropped Fuel Rod Accident**

For the purposes of this analysis, it is assumed that a dropped
fuel rod results in a broken pin releasing the accumulated fission pro-
ducts. The following bases have been established to calculate the
radiation dose from such an accident:

a. It is stipulated that the assembly has been operated continu-
ously for 200 hr at an average power level of 0.065 watt. It should be
noted that this is a conservative value since the assembly will not be
operated on a continuous basis; the 200 hours of operating time for each
configuration is essentially spread over a period of several weeks.

b. The assembly contains 1200 rods. Since the actual loadings for
the three reflected cases are 1206, 2132 and 3430 rods respectively, this
will result in conservative results.

c. The peak to space-averaged flux ratio during operation is a
maximum of 120, with the rod at peak flux contributing 1/10 of the
total power of the assembly.

d. The fuel rod subjected to the specified highest specific power is broken open and the fuel is completely fragmented, releasing 100% of the gaseous fission products.

e. The iodine (in elementary form) diffuses uniformly throughout the 18' x 32' x 15' room to calculate the on-site internal dose. The off-site dose was calculated using the very conservative Pasqual's principle of atmospheric diffusion.

The fission products inventory was calculated by means of the RSAC code [9]. The results indicated that the iodine isotopes were the only significant contributor to the inhalation dose. Using the active worker breathing rate of $3.47 \times 10^{-4} \text{ m}^3/\text{sec}$ specified in 10 CFR 20, a person remaining in the room would accumulate a thyroid dose of 0.023 mr for each minute he remains in the room, after the hottest rod breaks open. The total dose to a person that inhales all of the iodine contained in the rod would be 0.2 r. The established radiation safeguards at the University of Florida require the personnel to abandon the area immediately and notify the Radiation Safety Officer. The maximum time required to evacuate the assembly and the fuel storage area is 10 sec with 30 sec needed to evacuate the entire building; these times have been measured during practice evacuation of the building.

The off-site total inhalation dose, calculated assuming the iodine is released in one puff with zero wind velocity, cloud inverted condition, were typically:
The direct radiation dose from all the fission products of the irradiated fuel elements is calculated to be 1.3 mr/hr.

**Accidental Criticality and Subsequent Power Excursion**

Accidental criticality and a subsequent power excursion could occur only by the uncontrolled addition of water to the assembly and gross error in the calculations and/or procedures. The occurrence of such an accident is highly improbably and would require:

a. Setting a core width corresponding to the 1.5 metal/water ratio and proceed to install the fuel spacer system for the 0.5 metal/water ratio, load the fuel under these conditions and disregard small water height increments and $1/M$ measurements. To do this, several administrative rules would have to be wilfully ignored.

and/or

b. Improper setting of the following: the weir height, the water height scram system for any configuration and a gross error in the calculations causing criticality at about half the design core height (91.44 cm). It should be noted that the calculational method used to predict that $k_{\text{eff}}$ values has been tested successfully against the results of 29 different critical assemblies [7].

and/or

c. An obstructed 2 inch line from the core to the weir drain system and a simultaneous failure of the water height scram together with the referred to error in the calculations.
d. Violation by the facility operator of the administrative procedures requiring a visual check of the assembly before start-up and continuous attention to the control console instrumentation to determine the status of the assembly at all times.

The consequences of such an accident were determined by assuming the following:

1. The assembly is initially at a steady state power level of 10 watts (normal average power is .065 watt).

2. Water flows into the system at the maximum rate of 12 gpm.

3. The reactivity addition rate is 0.05 $/sec$. This rate is larger than the calculated rate (.04 $/sec$) for the case discussed above and more than that calculated to occur near critical for properly loaded assembly.

4. The power scram is set, through calibration or other error, at 10 kw. Corresponding error settings occur for the neutron flux and period scram.

5. A scram occurs 1 sec after a power level of 10 kw is reached. This time has been determined as the elapsed time from the initiation of a scram signal to the opening of the dump valves.

6. No feedback effects are considered. This is a good approximation to our case due to the low power levels involved and again is a conservative assumption.

The calculations were made using the IREKIN code, described in reference [10]. IREKIN numerically solves the point model kinetics equations.

Starting with the assembly one dollar subcritical ($k_{eff} = .993$), and proceeding with the described excursion, the accident would yield
a peak power of 69 kw, the total energy release is 0.5 Mwsec. The power vs. time behavior of the assembly for the postulated power excursion is shown in Fig. 9.

The combined neutron and gamma dose to the operator is 1.5 rem assuming that: the energy release is instantaneous, all neutrons have an energy of 1 MeV and there is no attenuation in the assembly. The proper RBE factors were taken into consideration.

It is concluded that, even if such an improbable accident would occur, the hazards to personnel and the general population are not significant.
At $t=0$: $k_{\text{eff}} = 0.993$, Power = 0
Ramp Insertion of $0.05\$/sec
Scram 1 sec after 10 kw
Scram Rate = $0.14\$/sec
Total Energy Release = 0.5Mwsec
CHAPTER IV

THE DATA ACQUISITION SYSTEM

Introduction

In the delicate and laborious task of performing nuclear experiments the most common source of difficulties and errors lies in the acquisition of the data. Modern nuclear instrumentation with its excellent time-energy resolution has enhanced detection sensitivity to the extent that deviations previously masked by the poorer resolution of the equipment are easily distinguishable. The major problem now lies in the degree of reproducibility of the results. Although in general the instrumentation is very reliable the enhanced sensitivity demands continual standardization for the sake of reproducibility.

The "brain" of the data acquisition system for pulsed neutron measurements is the multichannel analyzer (MCA) which is now available with a large number of data channels and narrower channel widths for increased time resolution. In general, the mode of data acquisition for a pulsing experiment differs from that of many other types of nuclear experiments. In particular, neutron interactions with a suitable detector are fed into the analyzer while it is time-sweeping. Ideally every neutron interaction should be counted regardless of the amplitude of the collected pulse. This implies that the linear signal originated at the neutron detector should be converted to a logic signal so that its probability of being recorded is independent of its amplitude.
Logic signals have fixed amplitude-shape characteristic and convey information by their presence, absence of time relationship with respect to a reference signal. Only gamma and noise discrimination are necessary; this process will inherently eliminate the counting of weak neutron interactions.

The process of registering events whose density per unit time varies considerably requires an electronic system with extremely high time resolution and a broad frequency response. Typically the counting rates at the peak of the neutron burst in pulsed neutron measurements vary from $10^5$ to $>10^6$ event/sec. This high, time-varying count rate caused drastic losses in conventional instrumentation due to pulse pile-up and circuit induced distortion when the dynamic range of the circuit is exceeded causing a net amplitude shift in excess of that tolerable by the system. The alternative modus operandi is to use low count rates so that resolution losses are minimized; this, however, increases the probability of systematic errors due to much longer running times.

The count rates encountered during the initial experiments with a He$^3$ counter close to the neutron generator target in the SPERT assembly exceeded $10^6$ event/sec at the peak. Saturation effects were observed in modular instrumentation incorporating the latest F.E.T. preamplifier and double-delay line pulse shaping amplifier at about $10^5$ event/sec. The fast pulse train caused a drop in the base voltage of the Field Effect Transistor which prevented further counting until a significant reduction in the count rate allowed the voltage to recover. Changing to a fast scintillation prototype preamplifier improved the situation somewhat. However, if the potential count rate in the UFSA assembly
were efficiently handled with few losses the actual running time of a particular experiment could be reduced to a few minutes. This possibility forced the development of the ultra-fast instrumentation channels used in this work.

The electronic counting system\(^1\) used throughout the experimentation used a transformer-coupled pulse amplifier to produce a fast logic signal to the input of the MCA from a slower input signal originating in a He\(^3\) counter. The system has superior counting and stability characteristics. The resolution time of the system is practically nil compared to that of the MCA. In general, data acquisition times were reduced to a few minutes; in that time it was possible to obtain, at most of the detector locations, the following counting statistics:

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Peak Channel Counts:</td>
<td>217</td>
</tr>
<tr>
<td>Counting Time:</td>
<td>2 - 10 minutes</td>
</tr>
<tr>
<td>No. of data channels:</td>
<td>1024</td>
</tr>
<tr>
<td>Channel Width:</td>
<td>20 micro sec</td>
</tr>
<tr>
<td>Pulse repetition rate:</td>
<td>30 hz</td>
</tr>
<tr>
<td>Noise Level:</td>
<td>40 event/min</td>
</tr>
<tr>
<td>Background Level:</td>
<td>140 event/min</td>
</tr>
</tbody>
</table>

The Neutron Detector

The neutron detectors used throughout the experimental program were proportional counters filled with He\(^3\). The counters are of the Texlium variety made by Texas Nuclear Corporation and were specially built to conform to the UFSA core grid. Detectors of this type have been

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1. The assistance of Mr. Joel B. Ayers of ORTEC, Inc., who suggested this electronic instrumentation is gratefully acknowledged.
preferred over the BF$_3$ variety at the University of Florida because of the larger neutron absorption cross section and operational reliability. He$^3$ undergoes the following reaction

$$\text{He}^3 + n \rightarrow \text{H}^3 + p + 0.764 \text{ Mev.}$$

The cross section for this reaction is $5327 \pm 10$ barn at $v_o = 2200$ m/sec compared to $3840 \pm 11$ barn for B$^{10}$. He$^3$ follows a $1/v$ law in the energy range from 0 to 200 kev. The pulse heights yielded by a He$^3$ filled counter are proportional to the energy of the neutrons plus 764 kev. The reaction has been used for neutron spectroscopy from the 100 kev to 2 Mev energy range. Gamma discrimination can be easily accomplished by the use of a biased integral discriminator.

The detectors were long and thin and each took the place of a fuel element in the core. The active length of the counter is slightly less than the active length of the fuel. Two one atmosphere (predominantly thermal detection) and one ten atmosphere (more responsive to higher energy neutrons) detectors were used in the experimental program. A sketch of the physical characteristics of the counters is shown in Fig. 10. The thin, long cylindrical shape enhances the time characteristics of the counter. Although normally a 1 atmosphere He$^3$ detector operates with a bias voltage of 1000 volt, the minimum input pulse voltage requirement of the pulse transformer was such that the operating voltage of the counters had to be raised to 1200 volt. At this voltage the slope of the counts vs. high voltage curve was about 8% per 100 volt; therefore very stable, low ripple high voltage supplies were used to insure reproducibility of the detector response.
FIG. 10  PHYSICAL CHARACTERISTICS OF THE $\text{He}^3$ NEUTRON COUNTERS
The Electronic Instrumentation

A block diagram of the instrumentation used in the pulse propagation experiments is shown in Figs. 11 and 12. Two independent data acquisition systems with a common start-stop clock are necessary to carry out the measurements: (1) a system connected to a movable detector that obtains the time profile of the propagating pulse at a given position and (2) the all important normalizing detector, fixed at one position. Two 1 atmosphere He³ detectors with the characteristics described above were used for these purposes. The system is essentially composed of signal transmitting devices and data registering and handling units.

The movable detector data acquisition system (MDDAS) consists of:

1. Detector - High Voltage Power Supply

The high voltage power supply was an ultrastable FLUKE 405 B with superior stability and negligible ripple. Manufacturer's specifications state the stability at .005% per hour and the ripple at less than 1 mv RMS.

2. ORTEC Model 260 Time-Pickoff unit, with 3000 volt isolation

The time-pickoff units are normally used to detect the time of arrival of a detected particle, usually with subnanosecond precision. The use of this characteristic and the electronic arrangement shown in Figs. 11 and 12 permitted the counting of neutron events with excellent time resolution. To our knowledge this is the first time that the time-pickoff units have been used for this application.

Briefly, the system operates as follows: the primary of a toroidal transformer, having a bandpass for very high frequencies only, is inserted between the detector and the bias power supply. Fast
FIG. 11  MOVABLE DETECTOR DATA ACQUISITION SYSTEM
FIG. 12 NORMALIZING DETECTOR DATA ACQUISITION SYSTEM
components of the detector signal will actuate a wide band transistor amplifier and tunnel diode discriminator from the secondary of the transformer. A line driver buffer is also provided. The power and control bias are provided by the ORTEC 403 Time Pickoff Control.

3. Modified ORTEC 403 Time Pickoff-Control

The 403 Time-Pickoff Control provides control and fan-out buffering for time derivation units such as the 260 Time Pickoff Unit. The fan-out buffer accepts the fast negative logic signal from the 260 unit and provides either fast negative or slower positive logic output signals.

The 403 Time-Pickoff Control had to be modified to make its output signal compatible with the input requirements of the 212 Pulsed Neutron Logic Unit used with the MCA. The 212 plug-in unit requires pulses with rise times longer than 50 nsec while the positive output from the 403 control unit has a rise time ~10 nsec. Furthermore, the positive logic signal from the 403 control has a 0.5 microsec pulse width which is too wide for the time resolution required. Therefore two modifications were made: a capacitor\(^2\) (C12) was changed from 270 pf to 68 pf reducing the pulse width to 0.12 microsec and an inductor coil (L1) was removed from the system changing the rise time to 50 nsec.

The modifications made the system compatible with the input requirements of the MCA and avoided overdriving the analyzer.

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2. Refer to ORTEC, Inc. "Instruction Manual for 403 Time-Pickoff Control."
4. Technical Measurement Corp. 1024 Multichannel Analyzer (MCA) and Data Handling Units.

The time analysis of the pulse and the corresponding data output are obtained from the following coupled instrumentation:

- Pulsed Neutron Logic Unit, TMC Model 212
- Digital Computer Unit, TMC Model CN1024
- Data Output Unit, TMC Model 220C
- Digital Recorder, Hewlett-Packard Model 561B
- Binary Tape Perforator, Tally Model 420

A trigger from an external pulse generator starts the sweep of the analyzer as controlled by the pulsed neutron logic unit and initiates the burst at the accelerator. The 212 has variable channel widths from 10 to 2560 microsec. The full 1024 channels were used in all the measurements. The storage capacity of the CN1024 is \(2^{17}\) counts. Both printed paper tape and perforated binary tape were obtained as output.

5. ORTEC 430 Scaler

For some of the flux traverses and the \(1/M\) measurements, the integral counts were recorded in a 10 Mhz scaler.

The normalizing detector data acquisition system (NDDAS) consists of the same components as the movable detection system except that a 256 multichannel analyzer, TMC Model CN110 was used for data handling and the integral counts were accumulated in an ORTEC 429 Scaler, modified for 7 Mhz counting rate.

The Resolution Time Correction

The resolution time of the pulsed neutron data acquisition system was the parameter under consideration while searching for a well-matched
and fast electronic instrumentation set-up. Some of the findings were surprising and may explain some of the discrepancies found in previous pulse propagation and neutron wave experiments which used an accelerator type neutron source. In these experiments it has been customary to increase the neutron yield of the generator as the movable detector is positioned farther away from the source to keep running times within tolerable limits. The count rate at the movable detector is then maintained at a level which has been determined to be acceptable; however the normalizing detector - usually fixed in a position close to the source (normally at the "thermalizing tank" if one is used) - may then be exposed to count rates beyond the capabilities of the detector and the associated instrumentation. Since the movable detector signal was always fed into an analyzer, saturation effects could be easily detected; the signal from the normalizing counter, however, was always being fed into a scaler where saturation effects may go unnoticed since a time profile is not available.

During most of the present work, two MCA's were available and the phenomenon described above was observed and count rates throughout the experimental program limited to those permissible by the time characteristics of the counting system.

The experimental program being carried out with the UFSA subcritical calls for the one to one comparison between theoretically calculated and experimentally determined time profiles of the neutron flux as a function of space. The shape of the pulse, rather than conventional integral parameters extracted from it, is therefore the significant and required information. For this reason the influence of the resolution time correction on the shape of the pulse came under close
The resolution time of several, submicrosecond preamplifier-linear amplifier combinations was shown to be a function of the mode of operation and of the count rate. For this type of instrumentation, there is a significant difference between the resolution time as determined by a steady-state technique (two source method) and a dynamic method (maximum count rate method [11]), the latter method yielding a much larger resolution time.

For the instrumentation finally chosen to carry out the pulsed experiments in this work no significant difference was found between the results of the two methods. Furthermore the pulse transformer-amplifier combination behaves like an ideal non-paralyzing system. The resolution time of the overall data acquisition system was found to be primarily determined by the multichannel analyzer.

As a matter of illustration, when the system depicted below was used the resolution time changed from -0.24 microsec as determined under low count rate steady-state conditions to -8 microsec as determined under high count rate, pulsing mode conditions.
Shown in Fig. 13 are two time profiles obtained at the same position in the UFSA core for the same count rate. The conventional preamplifier-linear amplifier system shown above was used to obtain one of the profiles and the instrumentation selected for this work was used to obtain the other. Both were corrected for resolution time losses with the best available value for the resolution time. The distortion observed in the pulse shape obtained with the conventional preamplifier-linear amplifier system is non-linear due to the count-rate dependent resolution time. It should be noted that the count rate near the peak of the pulse was less than $10^5$ count sec. The conventional system response "flattens out" when it reaches complete saturation - in this case above $10^5$ count sec. Saturation effects are not observed in the pulse transformer system until the count rate exceeds $3.3 \times 10^6$ count sec. An erroneous resolution time correction, or one which used a resolution time that does not characterize the system throughout the range of count rates will change the shape of the neutron pulse and consequently affect any analysis done on the pulse shape obtained.

As pointed out by Bierman, Garlid and Clark [11], in a pulsing experiment it is necessary to determine whether the counting system has the characteristics of a purely non-paralyzing system or those of a mixture of paralyzing and non-paralyzing systems. Using the method described by Bierman and co-workers, it was established that the data acquisition system being used in the present work is, as close as can be determined, non-paralyzing. The resolving time of the system is essentially determined by the width of the input pulse to the analyzer and the MCA characteristics.

It should be noted that for a wide range of count rates, depending
FIG. 13  TIME PROFILES OF NEUTRON BURST RECORDED BY CONVENTIONAL ELECTRONIC INSTRUMENTATION AND BY THE TIME-PICKOFF SYSTEM
on the magnitude of the resolving time, it is not necessary to prop-
erly identify the counting system since no significant difference in
the corrected counts are observed when using the paralyzing or non-
paralyzing correction. This is shown in Fig. 14.

The resolution time correction for the UFSA data acquisition
system is then given by:

$$N_T = \frac{N_0}{1 - \frac{N_{0,\text{true}}}{C W \times T_{r,s}}}$$

where

- $N_T$ = true number of events in a given channel
- $N_0$ = observed number of events in a given channel
- $T_r$ = resolution time of the system
- $C W$ = channel width
- $T_{r,s}$ = number of sweeps of the analyzer

and, for a non-paralyzing system

- $T_r$ = reciprocal of the maximum observable count rate

Using this method the following resolution or resolving time fac-
tors were determined:

A) MDDAS including 1024 Multichannel Analyzer

$$T_r = 0.31 \pm 0.015 \text{ microsec}$$

B) MDDAS excluding Multichannel Analyzer

$$T_r = 0.20 \pm 0.01 \text{ microsec}$$

C) NDDAS including 256 Multichannel Analyzer

$$T_r = 0.56 \pm 0.02 \text{ microsec}$$

D) NDDAS excluding Multichannel Analyzer

$$T_r = 0.20 \pm 0.01 \text{ microsec}$$
FIG. 14  THE PARALIZING, NON-PARALIZING SYSTEM RESOLUTION TIME CORRECTION AS A FUNCTION OF COUNT RATE
The Normalization Technique

The analysis of a pulse propagation experiment will yield information on the velocity of propagation, the attenuation and pulse shape as a function of position of a propagating disturbance. To determine the attenuation or relative amplitude of the pulse at different positions in an assembly, the data must be normalized to a fixed reference so that variations in the source strength, data acquisition time, etc., can be properly accounted for. The usual technique requires accumulating integral counts in a scaler for each measurement and reducing all measurements to the fixed reference afterwards. As was mentioned previously, the resolution time correction can have a significant effect on the normalizing factor since widely differing count rates are employed. It is extremely hard, if not impossible, to obtain a proper resolution time correction based on integral counts determined from a time-varying count rate.

Two methods of normalization and their relative merits are discussed below.

The Integral Count Method is the normally employed method of normalization. The total counts accumulated with the NDDAS for all runs is referred to a predetermined one, with the normalizing detector in a fixed position while the movable detector changes position.

The Analyzer Method, which is essentially the same as the above except that the counts arising from the normalizing detector are stored as a function of time in a multichannel analyzer. The average of a series of ratios obtained by dividing the corrected channel counts by the corresponding channel counts of a predetermined measurement gives the normalizing factor.
The Analyzer Method is intrinsically more accurate than the Integral Count Method because it permits an "exact" correction for resolution time losses by the use of the expression previously given applied to the recorded time profile. The correction for the integral counts is, on the other hand, inaccurate since the count rate is continuously changing and no base exists for a resolution time correction.

It was found, however, that as long as the count rate near the peak of the pulse is kept well within the capabilities of the NDDAS no significant difference is observed in the results of the two methods. This is due to the fact that ratios are being taken in both cases; this tends to minimize whatever differences there might be. Thus, it is concluded that with proper care the Integral Method is adequate whenever an analyzer is not available for normalization purposes.

It should be noted that an "effective" resolution time can be used to improve the results of the Integral Method. This "effective" resolution time can be found by forcing the normalizing ratio obtained from two runs by the Integral Method to match the normalizing ratio obtained from the Analyzer Method for the same two runs by adjusting the resolution time correction applied to the integral counts.

Comments

Certain inconsistencies in the results of the first few experiments prompted a careful inspection of the multichannel analyzer modus operandi. For the sake of completeness the significant findings are listed below.

A) Operation with the 10 microsec channel width (selected by the settings of the 212 plug-in-unit) proves to be unreliable due to instabilities in the clock and gating circuits. Channel widths of 20
microsec or longer are stable.

B) The address current setting of the CN1024 is extremely critical, markedly so for high count rates.

C) An optimum pulse into the analyzer should have a rise time of 50 nanosec, a total width of .1 microsec and an amplitude of 3-5 volt.

D) Above noise level, the discriminator setting of the 212 logic unit becomes irrelevant when a constant pulse height is used as input.

E) Reproducibility tests were performed on the analyzer with the neutron generator in continuous mode.

The statistical analysis of the channel counts gave:

- 72% were less than 1 from the mean
- 25% were between 1 and 2 from the mean
- 3% were between 2 and 3 from the mean

The system is statistically well-behaved.
CHAPTER V

NUCLEAR CALIBRATION OF THE UFSA SUBCRITICAL

Introduction

The nuclear calibration of the University of Florida SPERT Assembly was performed prior to the space-time kinetics studies. The calibration involved conventional inverse multiplication measurements, absolute $k_{eff}$ determination by pulsing techniques and comparison with multigroup-multiregion diffusion theory calculations. The Garelis-Russell technique was employed to determine $k\beta/\lambda$ and this result used to calculate $k_{eff}$ by coupling it with the experimentally determined decay constant and the theoretically calculated effective delayed neutron fraction. During this phase of the experimentation, "spatial effects" were noted in both $\alpha$ and $k\beta/\lambda$. These effects and other interesting kinetic phenomena involving these basic reactor parameters were considered worthy of further study and were investigated during the main part of the research. They are discussed in Part 2, Chapter V. In this section the results pertinent to the necessary calibration of the system are given.

Theoretical Notes

The Inverse Multiplication Method

Under ideal conditions, usually met only in small-fast assemblies, the reactivity can be represented by
\[
\frac{1 - k}{k} = \frac{1}{M - 1} \text{ or } 1 - k = 1/M
\]

where \( M \) is the net neutron multiplication in the assembly with a centrally located source. In practice, the multiplication is obtained from the ratio of multiplied to unmultiplied counts with a centrally located source. The unmultiplied counts are obtained with the fissile material removed and all other conditions undisturbed. In water-moderated cores it is difficult to match neutron spectra for multiplied and unmultiplied counts and deviations from the ideal \( M \) are to be expected. If possible, a search for detector locations should be conducted so as to obtain curves that follow the expected behavior of \( 1/M \).

Even if \( k \) can not be directly inferred from the \( 1/M \) determination, the curve of reciprocal count rate vs. the parameter that controls reactivity (fuel loading or moderator height or \% control rod withdrawal) is a useful guide for safely approaching criticality if a well-behaved curve can be obtained.

The inverse multiplication curve can be obtained as a function of moderator height by first obtaining a series of unmultiplied counts at various water levels and the multiplied counts as the water level is raised with the assembly originally air-spaced. Sensitivity to geometrical configuration (source-detector-water level) requires an empirical determination of "well-behaved" detector positions.

**Reactivity Measurements by the Pulsing Technique**

The pulsed-neutron technique has been used successfully for several years to measure reactivity. The transient neutron density following a burst of neutrons is used to determine the reactivity of the system by either the Simmons and King method [12], Sjostrand's area ratio method
[13], Gozani's extrapolated area-ratio analysis [14] or the Garelis-Russel technique [8]. In all these techniques it is essential that a fundamental spatial distribution of the neutrons be established for a correct determination of the decay constant and, therefore, the reactivity of the system.

The Simmons and King method established that a value for the reactivity can be obtained directly if a prompt fundamental decay constant can be measured at delayed critical. The value of $\alpha$ at delayed critical determines $\beta/\lambda$ and if these parameters are assumed constant over the reactivity range of interest a value of $\alpha$ can be obtained. The technique has given good results up to ~ $20$ subcritical in small multiplicative systems. The method strongly depends on being able to establish the prompt fundamental decay mode; it suffers from the inconvenient necessity of a delayed critical measurement and the assumed constancy of $\beta/\lambda$ throughout the ranges of reactivity.

The Sjostrand method improves the Simmons and King method in that the delayed critical measurement is no longer necessary but the results are shadowed by the strong influence of higher spatial harmonics. The method is based on the premise that the impulse response curve of the system is dominated by the prompt fundamental mode.

Gozani's treatment is a significant improvement over Sjostrand's method. Gozani proposed the extraction of the fundamental mode of prompt neutron decay from the impulse response curve and the extrapolation of this curve to zero time. The reactivity in dollars can be found by integrating under this curve; the method is independent of the presence of higher prompt spatial modes.

The Garelis-Russell technique, similar to Gozani's extrapolated
area-ratio method, is of practical value because of its intrinsic elimination of the effect of prompt higher harmonics. This method, which was used in the present work, was postulated originally for a repetitively pulsed (with a delta function source in time), bare, monoenergetic reactor but has proven to be of broader application. Garelis and Russell postulate, that for the conditions specified above:

\[
\frac{1}{R} \int N_p \exp((k\beta/t)t) dt = \frac{1}{R} \int N_p dt + N_d/R
\]

where

- \( N_p \) = prompt contribution to the neutron density
- \( N_d \) = delayed contribution to the neutron density
- \( R \) = pulsing rate

The following conditions should be satisfied for the correct application of the method.

a) \( R >> \lambda \), where \( \lambda \) is the decay constant of the shortest lived precursor group.

b) \( R >> \alpha \), where \( \alpha \) is the prompt fundamental decay constant.

c) The system must be pulsed a sufficient number of times so that \( \exp (m a_{sn}/R) << \exp (-a_{sn}/R) \), where \( m + 1 \) is the total number of pulses and the \( a_{sn} \) are the roots of the inhour equation.

d) The prompt root dominates the decay.

The Garelis-Russell treatment permits the determination of \( \rho (\$) \) when all the above conditions are satisfied, by the relation:

\[
\rho (\$) = \frac{\alpha}{k\beta/\lambda} - 1
\]

An absolute value of \( \rho \) is obtained by the use of a calculated effective delayed neutron fraction. Garelis has discussed the use of the method in reflected systems; the technique seems to be of practical value in
Becker and Quisenberry were able to compute a correction [16] for the observed spatial dependence of the reactivity in two-region systems by recognizing the differences in the spatial distributions of prompt and delayed neutrons. Their excellent comparative study of the above techniques emphasized the need for their recommended spatial correction unless the neutron detector is properly positioned to minimize this correction.

The study of Garlid and Bierman [17] correctly points out that in very large systems "an asymptotic spatial distribution cannot be established before the pulse has decayed away, since the asymptotic mode is one that is uniform everywhere in space." They proceed to apply a combination of first flight, age, and time-dependent diffusion theory to the study of pulsed measurements in large aqueous media; their conclusion is that their measured apparent decay constant is a good approximation to the asymptotic value and that pulsed measurements in very large multiplying systems may also give good results.

**Inverse Multiplication Measurements**

The safe approach to the design value of $k_{\text{eff}} \leq 0.99$ was undertaken with the conventional 1/M measurements until $k_{\text{eff}} = 0.95$ and then by both the 1/M and the pulsing technique.

To establish detector positions free from geometrical effects (source-detector-water level), six different locations were used until a water level of 60 cm ($k \approx 0.98$) was reached and four locations afterwards. Two of the detector locations, the closest to the neutron source, failed to describe the multiplication of the system. Shown in Fig. 15 are the
FIG. 15 DETECTOR POSITIONING SCHEME
locations employed for these measurements and for the absolute $k_{\text{eff}}$ determination by pulsing. Two 1 cu, centrally located Pu-Be sources were used for these measurements.

The measurements were performed in the following sequence:

a) Unmultiplied counts vs. water level were obtained from a water level of 20 cm to 91.4 cm above the bottom of the active fuel in steps of 5 cm.

b) The fuel was loaded in the assembly in the presence of two centrally located Pu-Be sources, with proper monitoring.

c) Multiplied counts vs. water level were obtained from a water level of 20 cm in 5 cm steps until an effective multiplication factor $\leq 0.99$ was reached. This procedure follows the criteria established for Initial Loading of the assembly, Part 2, Chapter III of this work; the 5 cm steps were more conservative increments than those specified for the Initial Loading of the assembly.

The results of these measurements are shown in Figs. 16 (A, B). All the curves were well behaved in the sense that none "nose-dived." Position 2 and 4 failed to properly describe the multiplication of the assemblies because of their nearness to the source. Position 1 seems to overestimate the multiplication, mainly because the unmultiplied count rate was extremely low at this position thus an apparently high ratio of multiplied/unmultiplied counts was obtained. A detector more sensitive to high energy neutrons was used in locations 1 and 6.

It should be noted that when the inverse multiplication is plotted vs. $1/H^2$ the predictions become quite linear much earlier than when plotted versus $H$. Better predictions are therefore made with the $1/H^2$ curves but the approach can become less conservative by underestimating
FIG. 16A INVERSE MULTIPLICATION vs. MODERATOR LEVEL
FIG. 16B INVERSE MULTIPLICATION vs. SQUARED INVERSE HEIGHT
the multiplication.

Shown in Table III are the values estimated for $k_{\text{eff}}$ for the four locations that seemed to represent the system best. Position 3 gives a lower limit and Position 1 an upper bound. No attempt was made to establish the error associated with measured $k_{\text{eff}}$, but it is believed that the 0.99 value obtained at the last water level is within $\pm 0.005$ of the true value.

**Absolute Determination of $k_{\text{eff}}$**

After an estimated value of $k_{\text{eff}} > 0.95$ was obtained from the $1/M$ measurements, an independent determination of $k$ was required by the operating license at every new increment in the moderator level (as determined by the criteria established in Part 1, Chapter 3). The technique chosen for this determination was the Garelis-Russell method of measuring $k\beta/\lambda$ and a simultaneous determination of the prompt fundamental decay constant.

Different detector locations were used to determine the influence of the source and of higher order harmonics contamination. Strong "spatial effects" were observed in both $\alpha$ and $k\beta/\lambda$. This phenomena will be discussed in detail in Part 2, Chapter V because of its importance. A seemingly true fundamental decay constant and "spaced-converged" $k\beta/\lambda$ were obtained at large distances from the source and were used to determine the reactivity of the system.

A Fortran IV, IBM 360 computer program named UNIPUL was coded to perform a unified analysis of the pulsed neutron data (see Appendix B). The program calculates the decay constant using Peierl's statistical analysis [18] and $k\beta/\lambda$ using the Garelis-Russell approach after the data
TABLE III

SUMMARY OF 1/M AND PULSED MEASUREMENTS

UFSA R1 Core
0.5 M/W Ratio
16.35 cm wide reflected core

<table>
<thead>
<tr>
<th>Height (cm)</th>
<th>Pos. 1</th>
<th>Pos. 3</th>
<th>Pos. 5</th>
<th>Pos. 6</th>
<th>INVERSE MULTIPLICATION $k_{eff}$</th>
<th>PULSED EXP $k_{eff}^b$</th>
<th>PREDICTED $k_{eff}$</th>
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<tr>
<td>20</td>
<td>.415</td>
<td>.481</td>
<td>.425</td>
<td>.581</td>
<td></td>
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<td>.7675</td>
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<tr>
<td>25</td>
<td>.744</td>
<td>.629</td>
<td>.705</td>
<td>.771</td>
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<tr>
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<td>.969</td>
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<td>.96</td>
<td>.9853</td>
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<td></td>
<td></td>
<td>1.00117</td>
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<td></td>
</tr>
</tbody>
</table>

\(a\) Above bottom of active fuel
\(b\) Averaged from 3 detector positions
has been resolution time corrected and background subtracted. A "pure" delayed neutron background is statistically calculated and used to determine $k\beta/\lambda$. The data can be normalized to a reference detector position for the analysis of the pulse propagation measurements in the time and in the frequency domain. A Fourier analysis of the pulse can also be performed if required.

Shown in Fig. 17 (A, B) are the experimentally determined $\alpha$, $k\beta/\lambda$ and $k$ as a function of moderator height obtained by averaging results from three chosen detector locations in the "asymptotic" region. The results are summarized, together with the $1/M$ measurements and theoretically calculated values in Table III. The excellent agreement between the experimental and theoretical results should be considered somewhat fortuitous. The calculations were done following the method outlined in Appendix A. Some later calculations [19] done by the Phillips Petroleum Co., showed more disagreement, especially at low water levels. The last calculations tried to account for the fact that there is a fissionable reflector above each experimental moderator height. This fact was disregarded in the calculational results shown in this work. The agreement at the 75 cm water level is good for all calculational methods.

Conclusions

The University of Florida SPERT Assembly has been operated for several months with very few operational problems. The system has proven to be extremely reliable and the instrumentation has performed adequately. The calibration of the system has established that meaningful values for the reactivity can be obtained when applying the
FIG. 17A  DECAY CONSTANT vs. MODERATOR LEVEL
**FIG. 17B** \( k_8/\bar{\kappa} \) AND \( k \) vs. MODERATOR LEVEL
Garelis-Russell technique to a reflected slab assembly when proper care is taken. Agreement between calculated and experimentally determined values of the reactivity is termed excellent but since only one case has been studied judgement on the overall applicability of the technique to this type of reactor configurations should be reserved until the other assemblies to be studied in the UFSA facility are duly analyzed. It should be pointed out that multiple detector positions are necessary to establish when an asymptotic decay constant is obtained, and that the value of $k_\text{B}/\lambda$ is affected by the input pulse width. As mentioned previously, a more detailed analysis of the pulsed neutron reactivity measurements is conducted later in this thesis.
PART 2

SPACE-TIME REACTOR KINETICS STUDIES WITH THE UNIVERSITY OF FLORIDA SPERT ASSEMBLY
CHAPTER I

INTRODUCTION

Statement of the Problem

The dynamic behavior of large reactor cores is recognized as one of the areas of reactor analysis in which practical, reliable calculational methods are needed. Although a significant amount of theoretical work has been done in this area, the experimentation has been restricted to the fundamental work done by Miley and co-workers at the University of Illinois [20, 21, 22] and the dispersion law studies in a heavy water-moderated, natural uranium assembly performed at the University of Florida [23]. The work at Illinois was hampered by the very broad input pulse obtained at the thermal column of the TRIGA reactor, by the neutronic size of the assembly and by the low value of the effective multiplication ($k_{\text{eff}} \leq 0.92$). The work at Florida, using comparatively narrow pulses from a neutron generator (and thermalizing tank), was also restricted by the neutronic size of the system and by the $k_{\text{eff}}$. No clean determination of "complete" spatial effects in multiplicative media has been reported.

The Reactor Safety Division of the United States Atomic Energy Commission established the Large Core Dynamics Experimental Program, as proposed by the Atomic Energy Division of the Phillips Petroleum Company, to study the behavior of large reactor cores. The objective
is the experimental verification of present space-time kinetics models and to guide the development of new calculational techniques.

Phase I of the Large Core Dynamics Experimental Program is being conducted at the University of Florida SPERT Assembly, under Subcontracts C281 and C635 with the Phillips Petroleum Co. This part of the program will provide the most fundamental neutron physics type of information as the redistribution characteristics of the neutron flux in a close-to-critical assembly are studied in the absence of inherent feedback effects. The adequacy of basic parts of the analytical models can be tested accurately by concentrating on the neutron physics problem.

In this work, the propagation of a narrow, non-asymptotic neutron burst is studied in the time and in the frequency domain. The experimental results are compared with results obtained from the two-group, space-time dependent, one-dimensional diffusion theory calculational scheme known as the WIGLE program [24]. A stringent test of the model is provided by a combined analysis in the time and frequency domain. Clean, unambiguous experimental information will be shown demonstrating the presence of spatial effects in large cores.

**Description of the Study**

Pulse propagation phenomena in a large-in-one-space dimension side reflected core were studied by introducing a fast neutron burst through one face of the assembly. A Cockcroft-Walton (TNC) type generator was employed to produce square neutron pulses of selected widths using the (d,t) neutron reaction.

The assembly which was studied is described in detail in Part 1
of this work. The subcritical assembly consists of a square array of 4.8\% enriched UO$_2$ SPERT F-1 fuel elements moderated by light water and with essentially an infinite light water reflector on the sides. The core is 243 cm long, 16.35 cm wide, has an effective height of 76 cm with 30.5 cm wide side reflectors. The effective multiplication constant of the assembly has been determined to be 0.990$\pm$0.003.

A fast, reliable set of electronic instrumentation was developed and used to record the very high, time-varying count rate as a function of position in the core of the assembly. A technique was developed to subtract the contribution of epicadmium neutrons from the recorded time profile of the neutron flux.

The study of the pulse propagation phenomena was conducted in a two-fold manner: in the time and in the frequency domain. The study in the time domain consisted of several sections, focusing the attention on basic aspects of the space and time dependence of propagating disturbances. The following aspects were investigated:

a) The propagation of 0.5 and 1.0 msec wide input pulse introduced at one end of the assembly. This investigation constituted the main part of the research; it involved a detailed comparison of the experimental results with the predictions of the WIGLE calculational scheme. The sensitivity of the one-dimensional model to small variations in the transverse leakage was also investigated.

b) Static flux traverses were conducted to determine the steady-state neutron distribution in the asymptotic region. Dynamic flux traverses were performed to determine whether any propagation occurs in the transverse direction.

c) The effect that the input pulse width has on the propagation
phenomena was studied.

The space-time data was also analyzed in the frequency domain by numerically obtaining the amplitude and phase of the zeroth Fourier moment of the pulse. The experimentally determined dispersion law of the system is compared to that predicted by the WIGLE scheme. The results of the WIGLE calculations performed for the analysis in the time domain were Fourier analyzed so that a one-to-one comparison could be performed.

Space dependent effects on reactivity measurements were studied by analyzing the pulsed data using the Garelis-Russell model.

Nomenclature Used in the Description of Pulse Propagation Phenomena

A series of directly measurable or inferred parameters are used to describe the pulse propagation phenomena, in the time and in the frequency domain. The most important of these parameters are defined below. The pulse shapes and the conventional full-width-at-half-maximum also form part of the analysis.

Delay Time

The delay time, $t_d$, is defined as the time displacement of the peak of the pulse from a reference position to the position under consideration. In this work all delay times are referred to zero time; this corresponds physically to the initiation of the pulse at the neutron generator.

Propagation Time

The propagation time of a narrow pulse in the assembly is defined as the delay time between two extreme positions in the assembly.
Asymptotic Velocity of Propagation

The asymptotic velocity of propagation, $v_p$, is defined as the inverse of the slope of the delay time vs. distance curve in the asymptotic region.

The Asymptotic Dynamic Inverse Relaxation Length

The asymptotic dynamic inverse relaxation length, $k_d$, is defined as the inverse of the distance required for the amplitude of the pulses to attenuate by a factor $e$.

The Damping Coefficient of the Neutron Wave

The damping coefficient of the neutron wave, $\alpha$, is a measure of the exponential attenuation of the amplitude of the wave for a given frequency. The damping coefficient is determined from the amplitudes of the zeroth Fourier moment by numerical transformation for each frequency of interest.

The Phase-Shift per Unit Length

The phase-shift per unit length of path, $\xi$, is determined from the phase angles of the zeroth Fourier moment by numerical transformation for each frequency of interest.

The damping coefficient and the phase shift per unit length of path are the real and the imaginary components of the complex inverse relaxation length, $\rho$, respectively.

It should be noted that in this work pulse and wave propagation from the "conventional" viewpoint, i.e., propagation of a residual disturbance rather than the propagation of a pulse or wave front, is being studied.
CHAPTER II

THEORETICAL NOTES

Introduction

The main concern of this work has been the experimental investigation of space-time kinetics effects on a large core subjected to a perturbation; the basic aspects of the neutron physics phenomena were studied. The analysis of the experiment has been conducted in both the time and the frequency domain. The theoretical model used for comparison with experiment, in the time and the frequency domain, was the two-group, space-time dependent diffusion equations numerically solved by the WIGLE code [24]. A one-to-one comparison of theory and experiment was performed in the time domain. For the corresponding study in the frequency domain both the WIGLE and experimental time profiles as a function of position were analyzed into its wave components by a numerical Fourier transformation; this method was suggested by Moore [25] and confirmed by Booth [26]. In this work only the zeroth Fourier moment is analyzed; the amplitude and phase angle of the zeroth Fourier moment correspond to those measured by the conventional neutron wave experiment.

Review of the Literature

A considerable amount of theoretical effort has been devoted to the space-time kinetics problem in nuclear reactors. The more important
methods presently in use are:

a) Direct solution of the multigroup, space-time diffusion equations, referred to as the "exact" method. This method constitutes a benchmark for the other techniques. The most prominent of these calculational schemes are:

1) WIGLE, a two-group, one-dimensional space-time diffusion theory computer program [24, 27, 28].

2) TWIGLE, a two-dimensional version of WIGLE [29].

3) FREAK, a fast reactor multigroup kinetics code [30].

4) A four-group, one-dimensional space-time diffusion theory computer program [31] developed by the Phillips Petroleum Co.

b) Flux Synthesis methods, which used the idea that the flux shapes which occur during a transient can be bracketed by a set of shape functions [32, 33, 34].

c) Conventional Modal methods, exemplified by the Foderaro-Garabedian technique [35] in which the flux is expanded in terms of the eigenfunctions of the wave equation (Helmholtz modes).

d) The Quasistatic approach of Ott [36, 37], a more sophisticated "factorizing" technique (the flux is factorized into an amplitude and a shape function). Use is made of the fact that the time dependence of the shape function is less important than the time dependence of the amplitude function.

e) The Adiabatic approximation [38], a "factorizing" technique in which, besides ignoring the time dependence of the shape function, no distinction is made between the prompt and delayed neutron sources.

Other related studies, on a somewhat different context, were performed by Kystra and Uhrig [39] and by Kavipurapu [40]. They studied the
space dependent reactor transfer function, within the framework of a combination of time-dependent Fermi-Age and diffusion theories.

A parallel field of study has been the neutron wave technique [41, 42, 43]. A major improvement in the neutron wave field was achieved when Moore proved [25] that the pulsed technique is equivalent to the neutron wave method when the pulsed data is analyzed in the Fourier transform plane. The work by Booth [26] confirmed this simplifying and time-saving approach to the powerful but time-consuming neutron wave measurements.

The study of multiplying media by neutron wave (or pulse) propagation has not been developed as extensively as in non-multiplying media. Brehm made a very elegant analysis of the problem, showing the excitation of slowing down modes whose relaxation lengths he was able to compute [44]. Dunlap and Perez studied the dispersion law of a heavy water-moderated, natural uranium assembly [45].

To date, no complete analysis of a pulse propagation experiment, i.e., in the time and in the frequency domain is available in the literature.

The WIGLE Calculational Scheme

The version of the WIGLE code used for the calculations performed in this work is an IBM 360 version of the WIGLE-40 program described by Radd [27]. The following description is extracted from Ref. 27.

WIGLE is a one-dimensional, two-energy-group, time-dependent diffusion program. It calculates the space-time behavior of the neutron flux in a reactor during a transient. The calculation is restricted to one-dimensional slab geometry with zero gradient or zero flux.
boundary conditions. Up to six delayed neutron groups can be considered in the calculations. The feedback subroutine may be used to introduce arbitrary changes in the reactor parameters, either for inherent feedback or for other time-dependent variations.

The basic equations solved by the WIGLE program are as follows:

\[ \nabla^0 D_1 \nabla \phi_1 - \Sigma_1 \phi_1 + \chi_1 (\nu \Sigma f_1 \phi_1 + \nu \Sigma f_2 \phi_2) (1 - \gamma_1 \delta) \]
\[ + \sum_{i=1}^{I} (C_{1i} + C_{2i}) + S_1 = \frac{\partial \phi_1}{\partial t} \]  
(1)

\[ \nabla^0 D_2 \nabla \phi_2 - \Sigma_2 \phi_2 + \chi_2 (\nu \Sigma f_1 \phi_1 + \nu \Sigma f_2 \phi_2) (1 - \gamma_2 \delta) \]
\[ + \Sigma_{f1} \phi_1 + \sum_{i=1}^{I} (1 - \delta) C_{2i} \lambda_1 + S_2 = \frac{\partial \phi_2}{\partial t} \]  
(2)

\[ - \lambda_i C_{1i} + \gamma_1 \beta_1 \nu \Sigma f_1 \phi_1 = \frac{\partial C_{1i}}{\partial t} \]  
(i = 1, 2, ..., I)  
(3a)

\[ - \lambda_i C_{2i} + \gamma_2 \beta_1 \nu \Sigma f_2 \phi_2 = \frac{\partial C_{2i}}{\partial t} \]  
(3b)

The number (I) of delayed neutron groups may be 0, 1, or 6. \( \delta \) can be 1 or 0. When \( \delta = 1 \) and \( \gamma_1 = \gamma_2 \), these equations are the conventional, two-group, time-dependent diffusion equations.

The time-dependent equations, and those equations necessary to represent the inherent feedback effects if they are considered, are reduced to finite (time) difference equations and numerically solved. WIGLE can handle up to 60 regions, 251 mesh points and 999 time steps.
**Neutron Wave Analysis**

The neutron wave technique has proven to be a very powerful method of determining the deficiencies a model has in predicting propagation phenomena or in establishing the "quality" of the results from a corresponding experiment.

The study of the space-time data in the frequency domain "looks" at the overall behavior of the asymptotic spatial region of the assembly with parameters that include information on all spatial points for each frequency of interest. The comparison of the predicted and measured real and imaginary components of the complex inverse relaxation length, $\delta$, is therefore a more comprehensive evaluation of the discrepancies than the point-wise comparison in the time domain. The analysis in the $\delta^2$ plane is an especially stringent test of the model and the experimental data.

In this work, the comparisons in the frequency domain were made by numerically determining the zeroth Fourier moment of the data measured in the time domain. This method was suggested by Moore.¹

Moore expressed Fourier moments of the space-time data as

$$\psi_n(r, f) = \int_{-\infty}^{\infty} dt \, t^n e^{-2\pi i ft} \psi(r, t)$$

where $\psi(r, t)$ is the neutron pulse at space point $r$ as a function of time, $t$, and $f$ is the frequency in cycles per second.

The fundamental space and energy mode propagating through the medium is given by:

---

¹ The author acknowledges Dr. M. N. Moore for several enlightening discussions on the subject of neutron waves propagation.
\[ \psi_o(r, f) = A_{10} (f) Q_{23} (r_2, r_3, B^2) \]
\[ \exp \{ [-\alpha_1 (f, B^2) + i \xi_1 (f, B^2)] z \} \]

where:

- \( A_{10} \) specifies the source condition.
- \( Q_{23} (r_2, r_3, B^2) \) specifies the transverse space dependence.
- \( \alpha_1 \) and \( \xi_1 \) are respectively the real and imaginary components of the complex inverse relaxation length.

The Fourier transform \( \psi_o(\hat{r}, f) \) of the unit impulse response \( \psi(\hat{r}, t) \) is the transfer function of the system. Therefore the amplitude and phase curves obtained from a numerical Fourier transformation of the pulse propagation data corresponds to those measured directly by the wave experiment.

It should be mentioned that, although most common nomenclature refers to the pulse propagation phenomena and/or the neutron wave propagation as "thermal" neutron propagation because of the "close-to-thermal" source employed and the characteristics of the media studied, there is no such "pure" phenomena in a multiplying medium. The disturbance propagates whether it started as a "fast" or "thermal" pulse and quickly loses memory of its origin (specially in water moderated media). The only difference between inserting a thermal or a fast burst of neutrons lies in the spatial distribution of source neutrons. The thermal source will appear essentially localized at the point of insertion while the fast source neutrons will penetrate deep into the assembly.
CHAPTER III

DESCRIPTION OF THE MEASUREMENTS

Introduction

In most pulse propagation and/or neutron wave measurements the experimental technique employed to obtain the thermal neutron pulse shape is the so-called cadmium difference method. In this technique the contribution of the epicadmium source neutrons is subtracted by making measurements with and without a cadmium shutter between the source and the system. No attempt is usually made to subtract the epicadmium neutron contribution due to fissions if the medium being studied is multiplicative.

But, most neutron detectors interact with epicadmium neutrons with varying degrees of efficiency; therefore to isolate the thermal neutron pulse shapes a different experimental technique was used in this work. Thus, it was possible to make a one-to-one comparison of the experimentally obtained thermal neutron pulse shapes with the corresponding results from the analytical scheme.

The subcritical configuration and electronic instrumentation used has been detailed in Part 1, Chapter IV. Unless specified, the measurements were made at a moderator height of 75 cm, corresponding to a $k_{\text{eff}} = 0.99 \pm 0.003$.

The system was operated under continuous water flow conditions. No heating of the water was observed by either the energy transferred
by the pump or the low operating average power level of the system (0.13 watt). The large volume of circulating water aided in maintaining the temperature constant. In one 14-hour period of operation the temperature change was about 1°C.

The Epicadmium Subtraction Method

The use of a thermalizing tank and the cadmium difference method is widespread for pulse propagation measurements. In the particular case of the UFSA assembly this technique is not pertinent, since the primary reason for the measurement was to test the analytical model under stringent, viz. non-asymptotic, conditions; furthermore, with the WIGLE model it is possible to account for the spatial and energy dependence of the fast source rather well within a two-energy group, slab geometry scheme. Therefore, no thermalizing tank was used and no cadmium plate was inserted between the source and the assembly.

The \(^3\text{He}\) detectors used for the measurements respond to epicadmium neutrons, although with decreasing efficiency of detection as the neutron energy increases. Therefore, to obtain a true basis of comparison between the model and experiment, the contribution of the epicadmium neutrons was subtracted from the time profile recorded by the detector.

The thermal and epicadmium neutron time profiles measured in the assembly do not differ significantly in shape in the asymptotic region but show significant differences in the time scale, the relative attenuation and the shapes near the source. Although the detectors used are about ten times more efficient for detecting thermal neutrons, the epicadmium flux is much larger than the thermal flux throughout the system and, in particular, close to the source. Therefore, the
following technique was used in most of the measurements:

a) Measurements were taken with the bare movable detector at each designated position and a normalization factor obtained from the fixed detector.

b) Measurements were taken with the same movable detector covered with a 0.018 in thick cadmium sleeve (the Cd cut-off energy for this thickness is -.53 ev) at the same positions and a normalization factor obtained from the fixed detector.

c) The time profile from each measurement was resolution time corrected, background subtracted and normalized with reference to the fixed detector (refer to Part 1, Chapter IV).

d) A point by point subtraction of the epicadmium time profile from the total time profile yielded the thermal neutron flux pulse shape at each spatial point.

It should be mentioned that, to be rigorous, a correction should be applied to the epicadmium flux at each space point to take into account the perturbation of the thermal flux introduced by the cadmium sleeve; this was not attempted in this study. It should not be significant in the highly absorbing medium being considered here.

As a corollary, it was expected that the epicadmium flux time profile could be compared with the theoretically predicted fast flux. The main problem encountered with this comparison lies in the fact that the energy-dependent efficiency of the detector affects the characteristics of the pulse. The efficiency of detection for the thermal group (0-.5 ev) is fairly constant and the comparison with the calculated thermal flux is on solid grounds; the same argument is not valid
however for the fast group. Although the results for the epicadmium flux obtained in these measurements will be presented, they should be looked on with the reservation that the energy-dependent neutron counting statistics invalidate the one-to-one comparison with calculational results. The epicadmium flux was obtained primarily for the necessary correction of the time profiles. The analysis of the thermal neutron group is duly emphasized.

Shown in Fig. 18 is a typical plot of the total and epicadmium time profiles recorded in the experiment and the thermal pulse shape obtained by subtraction. The magnitude of the correction is space-dependent, since no constant thermal/fast ratio (as detected by the counter) was observed. The correction varied from 3-10% of the total flux values.

The Geometrical Arrangement

Shown in Fig. 19 is a plan view of the source-subcritical assembly arrangement used throughout this work. The detector positions used for the measurements are clearly indicated. The numbers shown correspond to the positions of fuel pins with respect to the end of the assembly facing the neutron generator. There were 7 grid-holes between detector positions, giving a distance of 12.72 cm between data points. Throughout the rest of this report the positions of the data points will be referred to as PXX, where XX is the grid-hole number at which the data was recorded, i.e., P41 refers to position number 41 starting from the front face of the assembly. A total of 134 rows of fuel pins were in the assembly. Data was taken at nineteen spatial points for the main part of the research. The target assembly of the neutron generator
FIG. 18 THE TOTAL, EPICADMIUM AND THERMAL FLUX 117.44 CM FROM THE SOURCE
Dimensions in cm

FIG. 19 UFSA SOURCE-SUBCRITICAL ASSEMBLY GEOMETRICAL ARRANGEMENT
- PLAN VIEW -
penetrated 10 cm into the core. The purpose of this arrangement was to increase the solid angle subtended by the target in the assembly to augment the number of neutrons going into the system and to reduce the number of unscattered fast neutrons dispersing into the room and creating a shielding problem.

Synopsis of the Measurements

Flux Traverses

A series of flux traverses were done on the assembly to determine the flux shapes across the width and the height of the assembly. The traverses were done dynamically and statically.

Static traverses were performed with Indium foils at a distance of 66.6 cm from the neutron source. Measurements were done on the width and height of the assembly with the neutron generator in the continuous mode for a period of 6 hours to achieve the saturation activity of the foils. The foils were 5 mil thick, 7/16" in diameter, and 99.97+% Indium.

Dynamic traverses were performed across the width of the core at 41, 54, 130 cm from the neutron source with the generator in the pulsing mode to determine whether any propagation occurs in the transverse direction. The long He²² detectors were used for the measurements, which employed pulse widths of 0.5 and 1.0 msec as input.

Clean Core Pulse Propagation Measurements

A whole series of measurements using narrow, square, fast bursts of neutrons as a disturbance were carried out to determine the most important characteristics of the pulse propagation phenomena. Most of the measurements utilized the entire length of the assembly. Certain
aspects of pulse propagation phenomena were studied using a few or a single detector location in the clean core; these measurements will be discussed separately from the main part of the research as complementary experiments.

Pulse propagation measurements across the entire length of the core were conducted with input pulse widths of 0.5 and 1.0 msec. The time profile of the thermal neutron flux was obtained by the cadmium subtraction method at nineteen locations in the core. The first detector position was 3 cm from the neutron target and measurements were taken every 12.72 cm thereafter. The data thus obtained was then analyzed in the time domain.

With the exception of the measurements at the last four detector locations, $2^{16}$ or $2^{17}$ counts were accumulated in the peak channel of the analyzer and two measurements conducted at each position. The full 1024 available channels of the analyzer were used. The channel width was 20 microsec with an intrinsic 10 microsec storage dead time giving an effective time between channels of 30 microsec. Running times varied from 3 minutes close to the source to two hours at the position farthest from the source.

**Propagation of a Narrow Pulse**

To test the analytical model under more severe conditions, a narrow, 100 microsec wide input pulse was introduced into the assembly and the pulse profile recorded at a few detector locations. The experiment is identical to the one described above except that fewer detector locations were employed.

**Propagation of a Wide Pulse**

To illustrate the propagation characteristics of a pulse violating
the Johnson criterion [25] a wide, 10 msec square pulse was introduced into the assembly. Three selected positions were used for these measurements.

**Pulse Propagation vs. Pulse Width**

During the experimentation described above, the observation was made that narrow input pulses will yield the same pulse shape in the asymptotic region, with the peaks displaced in time. To gain an insight into this predicted occurrence, measurements were taken at one detector location (P83) with input pulse widths of 0.1, 0.5, 1.0, 2.0, 3.0, 4.0, 5.0, 6.0, 7.0, 8.0, 9.0 and 10.0 msec.

**Effect of Room Return at Peripheral Detector Positions**

The effect of neutrons reflected from the concrete walls on the measurements was investigated by recording pulse shapes at peripheral detector positions. The measurements were done with and without a shield of borated paraffin in the region in which the detector was located.
CHAPTER IV

EXPERIMENTAL AND THEORETICAL RESULTS
IN THE TIME DOMAIN

The Analytical Model

The analytical results were obtained using an IBM 360 version of the WIGLE code\(^1\) supplied by the Phillips Petroleum Company, and made operative on the IBM 360-50 at the University of Florida. Certain modifications were made to WIGLE to obtain punched card output of the time profile at selected spatial points. In this manner, the pulse shapes at the detector positions corresponding to the experimental measurements were readily available for direct comparison with experiment.

The WIGLE scheme, being a one-dimensional model, requires that the transverse leakage be taken into account by the proper adjustment of the parameters. The two-group nuclear parameters for WIGLE were obtained by the procedure outlined in Appendix A. The eigenvalue, three-dimensional flux shapes and the bucklings were obtained from the CORA [46] computer program. The core width was set at 16.35 cm, the reflector width at 30.5 cm and the reactor height at 76 cm. The experimental measurements were made with 75 cm of water above the bottom of the active fuel but there was a small water reflector at the bottom; thus the

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1. The author is indebted to Mrs. M. E. Radd, Nuclear Safety Research Branch, Phillips Petroleum Company, for her valuable cooperation in familiarizing the author in the use of the WIGLE code.
"effective" core height is very close to 76 cm. The transverse leakage was taken into account in the WIGLE scheme by changing the absorption cross section of the fast and the thermal group with the proper $D_{1}B_{1}^{2}$ obtained from the above calculations.

The aluminum port in which the target section of the accelerator is located created a large void in the source section of the core. Shown in Fig. 20 is a plan and front view of this arrangement. To mock-up the physical situation as closely as possible the input to the WIGLE code consisted of parameters for two different types of regions: one for the normal core and one for the region depicted in Fig. 20. The parameters calculated for the UFSA core were then volume-weighted together with two group parameters for aluminum and light water as a first approximation to the parameters in the source region (Region 1 in the calculations).

In Fig. 21 the type of material, number of mesh points and the mesh spacing used in the calculations is shown. Table IV shows the value of the parameters for Region 1 and for Regions 2, 3, 4, 5 respectively. Before the final runs were made the importance of the mesh spacing was investigated by running two identical cases with 166 and 99 mesh points respectively. No difference was found between results of these two cases.

The choice of time increments was more critical. When time steps of 5 $\mu$sec were used in the vicinity of the time where the value of the fast source was changed from $10^{4}$ to 0 severe oscillations developed at the first space point. To avoid this problem, time steps of 0.2 to 0.5 $\mu$sec were employed in this time region and the magnitude of the source was reduced in two steps; first to one-half the original value and then-
FIG. 20 PLAN AND FRONT VIEW OF THE CORE REGION ENCLOSING THE NEUTRON SOURCE
<table>
<thead>
<tr>
<th>REGION NO.</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
</tr>
</thead>
<tbody>
<tr>
<td>MATERIAL NO.</td>
<td>2</td>
<td>1</td>
<td>1</td>
<td>1</td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td>NO. OF MESH POINTS</td>
<td>12</td>
<td>9</td>
<td>4</td>
<td>7</td>
<td>64</td>
<td>1</td>
</tr>
<tr>
<td>MESH SIZE (cm)</td>
<td>.9916</td>
<td>1.0</td>
<td>1.179</td>
<td>1.8166</td>
<td>3.17905</td>
<td>2.777</td>
</tr>
<tr>
<td>MESH POINT NO.</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>DETECTOR POSITION NO.</td>
<td>13</td>
<td>25</td>
<td>32</td>
<td>36</td>
<td>40</td>
<td>44</td>
</tr>
<tr>
<td></td>
<td>6</td>
<td>13</td>
<td>20</td>
<td>27</td>
<td>34</td>
<td>41</td>
</tr>
</tbody>
</table>

**FIG. 21** ONE-DIMENSIONAL ARRANGEMENT OF THE UFSA CORE USED IN THE WIGLE CALCULATIONAL SCHEME
TABLE IV

INPUT PARAMETERS FOR THE WIGLE CALCULATIONAL SCHEME

UFSA R1 Core
0.5 M/W Ratio
16.35 cm wide reflected core

<table>
<thead>
<tr>
<th>Region</th>
<th>CORE</th>
<th>Regions 2,3,4,5,6</th>
<th>ALUMINUM</th>
<th>WATER</th>
</tr>
</thead>
<tbody>
<tr>
<td>D₁(cm)</td>
<td>1.5211</td>
<td>1.06379</td>
<td>1.9338</td>
<td>1.1370</td>
</tr>
<tr>
<td>σₐ₁(cm⁻¹)</td>
<td>.055588</td>
<td>.054350</td>
<td>.000228</td>
<td>.000589</td>
</tr>
<tr>
<td>σₙ₁(cm⁻¹)</td>
<td>.02116</td>
<td>.025224</td>
<td>.000159</td>
<td>.0483</td>
</tr>
<tr>
<td>νₑ₁(cm⁻¹)</td>
<td>.003247</td>
<td>.007337</td>
<td>0.0</td>
<td>0.0</td>
</tr>
<tr>
<td>D₂(cm⁻¹)</td>
<td>.2895</td>
<td>.20809</td>
<td>3.6078</td>
<td>.14938</td>
</tr>
<tr>
<td>σₐ₂(cm⁻¹)</td>
<td>.4234</td>
<td>.12217</td>
<td>.01048</td>
<td>.019242</td>
</tr>
<tr>
<td>νₑ₂(cm⁻¹)</td>
<td>.10027</td>
<td>.22662</td>
<td>0.0</td>
<td>0.0</td>
</tr>
<tr>
<td><a href="cm%E2%81%BB%C2%B9sec">1/ν₁</a></td>
<td>.5647E-7</td>
<td>.3324E-7</td>
<td></td>
<td></td>
</tr>
<tr>
<td><a href="cm%E2%81%BB%C2%B9sec">1/ν₂</a></td>
<td>.3324E-7</td>
<td>.3324E-5</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Note: Thermal group: 0-0.53 ev
Fast group: 0.53 ev - 10.0 MeV
to zero with 2-5 \( \mu \text{sec} \) (10 time steps) between the two steps. It was also observed that a long time after the input pulse was cut-off oscillations developed at spatial points close to the source when time increments larger than 50 \( \mu \text{sec} \) were used.

The time increments used for the 0.5 and 1.0 msec input pulse problems are shown in Table V. Stable operation of the code was obtained with these time steps. Running times were of the order of 20 minutes in the UF IBM 360-50.

The main problem in establishing a fair comparison between the calculational and experimental results lies in an adequate physical description of the spatial distribution of the external fast source. For example, when calculations were done with a spatially localized, 1 cm wide source placed at the physical location of the neutron target (Fig. 20) the calculated pulse shapes were found to agree reasonably well with experiment but the peaks of the pulses were delayed by 50 to 500 microsec with respect to the experimental values. Furthermore the calculations predicted a much sharper attenuation of the peak of the pulse near the source, although predicting an asymptotic relaxation length in agreement with experimental results. Therefore, a careful study was undertaken to obtain the "best" description of the spatial distribution of the source.

The best approximation to the source that could logically be assumed based on physical grounds was that of a spatial distribution given by a first-flight kernel of the form \( e^{-\Sigma_f Z} \) where \( \Sigma_f \) is the removal cross-section for source neutrons. The distribution used is shown in Fig. 22. The choice of a plane source kernel is reasonable – especially at points a few cm removed from the source – due to the physical
### TABLE V

**TIME STEPS USED FOR THE WIGLE CALCULATIONS**

UFSA R1 Core
0.5 M/W Ratio
16.35 cm wide reflected core

<table>
<thead>
<tr>
<th>Time Step (Microsec)</th>
<th>2</th>
<th>4</th>
<th>1</th>
<th>5</th>
<th>1</th>
<th>2</th>
<th>5</th>
<th>10</th>
<th>25</th>
<th>50</th>
</tr>
</thead>
<tbody>
<tr>
<td>To Time Step No.</td>
<td>40</td>
<td>140</td>
<td>160</td>
<td>170</td>
<td>205</td>
<td>235</td>
<td>315</td>
<td>415</td>
<td>655</td>
<td>795</td>
</tr>
</tbody>
</table>

**Input Pulse Width = 0.5 msec**

<table>
<thead>
<tr>
<th>Time Step (Microsec)</th>
<th>3</th>
<th>5</th>
<th>1</th>
<th>.5</th>
<th>1</th>
<th>2</th>
<th>5</th>
<th>10</th>
<th>25</th>
<th>50</th>
</tr>
</thead>
<tbody>
<tr>
<td>To Time Step No.</td>
<td>30</td>
<td>210</td>
<td>220</td>
<td>230</td>
<td>265</td>
<td>295</td>
<td>375</td>
<td>475</td>
<td>695</td>
<td>834</td>
</tr>
</tbody>
</table>

**Input Pulse Width = 1.0 msec**
FIG. 22 SPATIAL DISTRIBUTION OF SOURCE NEUTRONS INCORPORATED INTO THE WIGLE SCHEME
configuration of the source and the assembly. The planar nature of the source was confirmed experimentally by obtaining pulse propagation data across the width of the core close to and far from the source.

The removal cross-section for the uncollided source neutrons in region 1 was chosen to be the removal cross-section of neutrons from the fast group in that region as obtained from the two-group scheme (ref. Table IV). For regions 2,3,4 and 5 the corresponding removal cross-section was taken to be the removal cross-section of neutrons from the fast group in the four-group scheme.

It should be noted that the results are not sensitive to the value of the removal cross-section in region 1. The above choice of the removal cross-section for the remainder of the assembly is consistent with the overall calculational scheme used and with the value determined empirically for 14 Mev neutrons in light water.

This choice of a first-flight source gave calculational results in surprisingly good agreement with experiment, particularly for the delay times and the spatial attenuation. The predicted pulse shapes are somewhat narrower than the pulse shapes obtained from experiment. No significant differences between the results of the localized sources case mentioned above and a point kernel source distribution were observed. The point kernel source distribution underpredicts the penetration of the first collision fast neutrons into the assembly.

At this point a few comments on the energy distribution of the source neutrons are pertinent. A pure tritium target bombarded with a beam of monoenergetic deuteron ions will produce essentially monoenergetic neutrons, since the relationship between the energy of the ions, the neutron energy and the angle of emission is unique. However,
the (d,t) reaction rapidly becomes contaminated with the (d,d) reaction due to the accumulation of the deuteron beam on target. After ~ 600 microamp-hr per unit area of accumulated beam on target the original neutron yield from the (d,t) reaction has dropped to half of its original value while (d,d) neutrons account for ~ 1% of the neutron beam. After ~ 2000 microamp-hr per unit area of accumulated beam on target, the number of (d,d) neutrons has increased to 2%; this represents a detectable contamination, and may have some effect on the removal cross-section of source neutrons. In the present case, however, due to the coarse energy mesh used in the calculation scheme the effect is not considered to be important.

Typical results obtained with the WIGLE code are shown in Fig. 23 (A, B, C) for a 0.5 msec input pulse. The general characteristics of the pulse propagation phenomena are clearly displayed. The calculated spatial distribution of the neutrons as a function of time is shown for times of 1,2,3,5,7 and 9 msec in Fig. 24 (A, B). The spatial dispersion increases with time while the pulse attenuates severely in time and space.

The WIGLE results are discussed simultaneously with the results of the clean core measurements.

**Flux Traverses**

Static and dynamic flux traverses were obtained in the clean core to establish the following:

a) The asymptotic steady-state flux shapes in the transverse direction.

b) The planar propagation of the neutron pulse. This is extremely
FIG. 23A PULSE SHAPES PREDICTED BY WIGLE AT DIFFERENT POSITIONS IN THE UFSA R1 CORE
FIG. 23B  PULSE SHAPES PREDICTED BY WIGLE AT DIFFERENT POSITIONS IN THE UFSA R1 CORE

NORMALIZED TO
WIG-TERM-48
PULSE WIDTH = 0.5 MSEC
FIG. 23C PULSE SHAPES PREDICTED BY WICLE AT DIFFERENT POSITIONS IN THE UFSA R1 CORE

NORMALIZED TO WIC-THERM-76
PULSE WIDTH = 0.5 MSEC
FIG. 24A THE CALCULATED SPATIAL DISTRIBUTION OF THE THERMAL FLUX AT DIFFERENT TIMES AFTER THE PULSE
FIG. 24B  THE CALCULATED SPATIAL DISTRIBUTION OF THE THERMAL FLUX AT DIFFERENT TIMES AFTER THE PULSE.
important since the one-dimensional WIGLE calculations cannot account for transverse propagation of a disturbance.

Static Flux Traverses

Shown in Figs. 25 and 26 are the mapped steady-state fluxes across the height and the width of the core respectively, at a distance of 66.6 cm from the neutron source. This location was chosen because the delay time and attenuation of the pulses showed it to be at the beginning of the "asymptotic region", viz. the region in which both the velocity of propagation and the relaxation length have reached their asymptotic values. Both flux maps were done with Indium foils and appropriate corrections were applied to account for self-shielding and interference (shadowing) effects between foils.

The fluxes calculated from four-group diffusion theory using the AIM-6 code are displayed together with the experimental data. A cosine fit was made on the vertical flux and is shown as a solid line in Fig. 25; the fit obtained was excellent. The analysis of the results in this direction gives an extrapolation distance between 5 and 7 cm.

The results were not as consistent across the width of the core. A sophisticated aluminum foil holder was built to permit the location of the foils between two rows of fuel; the foils were inserted in slots parallel to the axis of the assembly. Due to the mechanical difficulty of positioning the foils in the assembly, the physical impossibility of having them perfectly horizontal and the arduous, significant correction due to the short distance between foils, only one irradiation was performed. When all this is taken into account the results look reasonable. It was found that the bowing of the fuel rods affected these measurements significantly. This was determined by positioning the
FIG. 25 THE ASYMPTOTIC STEADY-STATE VERTICAL FLUX
FIG. 26  THE ASYMPTOTIC STEADY-STATE HORIZONTAL FLUX
long detector in one position and rotating the fuel elements that surround it. The narrow core (6.5 in wide) is obviously very susceptible to this effect and to the exact centering of the neutron source whose location could not be determined to better than 0.5 cm accuracy.

Dynamic Flux Traverses

Dynamic flux traverses across the width of the core and reflector were carried out at distances of 41, 54, and 130 cm from the source. As mentioned above, these measurements are important since the theoretical model is one-dimensional and planar propagation is conceptually desirable.

Shown in Table VI are the "peaking times" obtained for several positions across the width of the core and side reflectors. The experiment was performed with both 0.5 and 1.0 msec input pulses. Only the results for the 0.5 msec input pulse are shown. The results for the 1 msec case display the same behavior. All the peaks occurred well within the experimental accuracy. Comparison of pulse shapes at different positions across the core and reflector revealed no differences in the basic shapes. Near the outer wall of the reflector a secondary pulse, which peaked at a time corresponding to the input pulse width, was observed. It was confirmed experimentally that this was due to neutron reflections from the walls of the facility room.

From the pulse propagation measurements, reactivity data was also obtained. These results are analyzed in Part 2, Chapter VI.

Clean Core Pulse Propagation Measurements

The main portion of the research propagation measurements in the clean, cold, side- reflected assembly. These most detailed experiments
TABLE VI

DELAY TIMES MEASURED ACROSS THE WIDTH OF THE CORE
- 0.5 MSEC INPUT PULSE -

UFSA R1 core
0.5 M/W ratio
16.35 cm wide reflected core

<table>
<thead>
<tr>
<th>Position No.</th>
<th>Distance to Core Center (cm)</th>
<th>Delay Times (msec)</th>
<th>Distance to the source</th>
</tr>
</thead>
<tbody>
<tr>
<td>A</td>
<td>7.27</td>
<td>.860</td>
<td>2.42</td>
</tr>
<tr>
<td>B</td>
<td>5.45</td>
<td>.830</td>
<td>2.51</td>
</tr>
<tr>
<td>C</td>
<td>3.63</td>
<td>.830</td>
<td>2.48</td>
</tr>
<tr>
<td>D</td>
<td>1.82</td>
<td>.830</td>
<td>2.45</td>
</tr>
<tr>
<td>E</td>
<td>0.0</td>
<td>.830</td>
<td>2.42</td>
</tr>
<tr>
<td>F</td>
<td>3.63</td>
<td>.830</td>
<td>2.54</td>
</tr>
<tr>
<td>G</td>
<td>5.45</td>
<td>.830</td>
<td>2.42</td>
</tr>
<tr>
<td>H</td>
<td>7.27</td>
<td>.860</td>
<td>2.51</td>
</tr>
<tr>
<td>I</td>
<td>10.9</td>
<td>.860</td>
<td>2.42</td>
</tr>
<tr>
<td>RE1</td>
<td>12.5</td>
<td>.890</td>
<td>2.45</td>
</tr>
<tr>
<td>RE2</td>
<td>14.3</td>
<td>.920</td>
<td>2.48</td>
</tr>
<tr>
<td>RE3</td>
<td>16.1</td>
<td>.950</td>
<td>—</td>
</tr>
<tr>
<td>RE4</td>
<td>17.9</td>
<td>.920</td>
<td>2.57</td>
</tr>
<tr>
<td>RE5</td>
<td>28.9</td>
<td>.920</td>
<td>—</td>
</tr>
<tr>
<td>RE6</td>
<td>31.8</td>
<td>—</td>
<td>2.45</td>
</tr>
<tr>
<td>RE7</td>
<td>34.3</td>
<td>.890</td>
<td>—</td>
</tr>
</tbody>
</table>
were conducted to investigate the propagation, throughout the core, of 0.5 and 1.0 msec input pulses. As mentioned in the previous section, particular aspects of the propagation phenomena investigated are dealt with separately.

The experimental technique used has already been described. Briefly, the time profile of the normalized thermal neutron flux is obtained at selected positions by the integral count method of normalization and the epicadmium subtraction technique. Nineteen detector positions were used for these measurements. For each pulse width two runs were made at each position, except at the far end of the core where data acquisition times became significantly longer. Most of the experimental errors assigned to the data were obtained by analyzing the results of these two measurements.

Following the nomenclature developed by Doshi and Miley [20] at the University of Illinois, the delay times \( t_d \), the dynamic inverse relaxation length \( \kappa_d \), the asymptotic velocity of propagation \( v_p \), the full-width at half-maximum (FWHM) of the propagating pulse and the pulse shapes will be analyzed to describe the propagation characteristics of a neutron burst in the very close to critical assembly. The experimental results are presented together with the calculated values.

It should be noted that practically all the figures showing the pulse shapes were plotted using a Calcomp plotter which is part of the IBM 1800 Computer facility in the Nuclear Engineering Sciences Department at the University of Florida. The data is plotted in a continuous manner and the points are superimposed later for convenience. Typically the experimental pulse shapes were plotted from 333 time steps; the corresponding calculational results have 231-245 time steps.
Shown in Fig. 27 (A, B, C) are a set of the experimentally determined thermal neutron time profiles for the 0.5 msec input pulse case. A representative set of the experimentally determined spatial distribution of neutrons in the assembly for increasing times after the source pulse is shown in Fig. 28 (A, B). The space and time dependent redistribution of the thermal flux as the disturbance propagates through the assembly is clearly displayed.

Tables VII and VIII show the experimentally determined and theoretically calculated delay times with reference to zero time (this time corresponds to the time at which the neutron generator initiates the burst and the multichannel analyzer starts its sweep), the FWHM and the counts at the peak of the pulse normalized to position 83 which is located in the asymptotic region. In Figs. 29 through 32 the delay times and the spatial attenuation for both pulse widths is shown.

Very good agreement is obtained between the theoretical and experimental results, throughout the length of the assembly, including the region near the source and the region farthest removed from the source where end effects are significant. This apparent agreement is encouraging and vouches for the calculational scheme and source description used.

The Propagation Time and the Asymptotic Velocity of Propagation

The propagation time, which is defined as the total delay time between peaks at the two extreme positions of the assembly, is $3.65 \pm 0.1$ msec.

The asymptotic velocity of propagation is defined as the inverse of the slope of the curve of the delay time vs. distance in the
FIG. 27A EXPERIMENTAL PULSE SHAPES AT DIFFERENT POSITIONS IN THE UFSA R1 CORE
FIG. 27B EXPERIMENTAL PULSE SHAPES AT DIFFERENT POSITIONS IN THE UFSA R1 CORE

NORMALIZED TO THERM-5-49
PULSE WIDTH = 0.5 MSEC
FIG. 27C  EXPERIMENTAL PULSE SHAPES AT DIFFERENT POSITIONS IN THE UFSA R1 CORE

NORMALIZED TO
THERM- S- 76
PULSE WIDTH = 0.5 MSECS
FIG. 28A  EXPERIMENTALLY DETERMINED SPATIAL DISTRIBUTION OF THE NEUTRONS AT DIFFERENT TIMES AFTER THE PULSE
UFSA R1 Clean Core
12.716 cm between points
0.5 msec input pulse

FIG. 28B EXPERIMENTALLY DETERMINED SPATIAL DISTRIBUTION OF THE NEUTRONS AT DIFFERENT TIMES AFTER THE PULSE
TABLE VII

CLEAN CORE PULSE PROPAGATION STUDIES
EXPERIMENTAL AND THEORETICAL RESULTS
- 0.5 MSEC INPUT PULSE WIDTH -

UFSA R1 Core
0.5 M/W Ration
16.35 cm wide reflected core

<table>
<thead>
<tr>
<th>POSITION NO</th>
<th>DISTANCE TO SOURCE (cm)</th>
<th>PEAK AT (msec)</th>
<th>FWHM (msec)</th>
<th>PEAK COUNTS</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Theory</td>
<td>Exp</td>
<td>Theory</td>
</tr>
<tr>
<td>6</td>
<td>3</td>
<td>0.500</td>
<td>0.50±.01</td>
<td>0.542</td>
</tr>
<tr>
<td>13</td>
<td>15.71</td>
<td>0.501</td>
<td>0.50±.01</td>
<td>0.615</td>
</tr>
<tr>
<td>20</td>
<td>28.43</td>
<td>0.504</td>
<td>0.52±.01</td>
<td>0.779</td>
</tr>
<tr>
<td>27</td>
<td>41.15</td>
<td>0.635</td>
<td>0.63±.01</td>
<td>1.098</td>
</tr>
<tr>
<td>34</td>
<td>53.86</td>
<td>0.840</td>
<td>0.82±.03</td>
<td>1.398</td>
</tr>
<tr>
<td>41</td>
<td>66.58</td>
<td>1.060</td>
<td>1.04±.03</td>
<td>1.766</td>
</tr>
<tr>
<td>48</td>
<td>79.30</td>
<td>1.300</td>
<td>1.28±.03</td>
<td>2.055</td>
</tr>
<tr>
<td>55</td>
<td>92.01</td>
<td>1.560</td>
<td>1.50±.04</td>
<td>2.355</td>
</tr>
<tr>
<td>62</td>
<td>104.73</td>
<td>1.840</td>
<td>1.82±.04</td>
<td>2.655</td>
</tr>
<tr>
<td>69</td>
<td>117.44</td>
<td>2.125</td>
<td>2.11±.04</td>
<td>2.925</td>
</tr>
<tr>
<td>76</td>
<td>130.16</td>
<td>2.400</td>
<td>2.40±.05</td>
<td>3.195</td>
</tr>
<tr>
<td>83</td>
<td>142.88</td>
<td>2.700</td>
<td>2.65±.06</td>
<td>3.465</td>
</tr>
<tr>
<td>90</td>
<td>155.60</td>
<td>3.000</td>
<td>2.96±.06</td>
<td>3.735</td>
</tr>
<tr>
<td>97</td>
<td>168.31</td>
<td>3.300</td>
<td>3.29±.06</td>
<td>3.900</td>
</tr>
<tr>
<td>104</td>
<td>181.03</td>
<td>3.570</td>
<td>3.50±.06</td>
<td>4.050</td>
</tr>
<tr>
<td>111</td>
<td>193.74</td>
<td>3.810</td>
<td>3.85±.08</td>
<td>4.125</td>
</tr>
<tr>
<td>118</td>
<td>206.46</td>
<td>4.000</td>
<td>4.00±.08</td>
<td>4.200</td>
</tr>
<tr>
<td>125</td>
<td>219.17</td>
<td>4.150</td>
<td>4.18±.08</td>
<td>4.200</td>
</tr>
<tr>
<td>132</td>
<td>231.88</td>
<td>4.210</td>
<td>4.21±.09</td>
<td>4.200</td>
</tr>
</tbody>
</table>

a Normalized to Position 83

b Errors assigned from the deviation from the mean of two measurements except for the last 5 space points. This error is usually larger than the counting statistics error. The error of the last 5 points was calculated from counting statistics.
# TABLE VIII

CLEAN CORE PULSE PROPAGATION STUDIES
EXPERIMENTAL AND THEORETICAL RESULTS
- 1.0 MSEC INPUT PULSE WIDTH -

UFSA R1 Core
0.5 M/W Ration
16.35 cm wide reflected core

<table>
<thead>
<tr>
<th>POSITION NO</th>
<th>DISTANCE TO SOURCE (cm)</th>
<th>PEAK AT (msec)</th>
<th>FWHM (msec)</th>
<th>PEAK COUNTS$^a$</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Theory</td>
<td>Exp</td>
<td>Theory</td>
</tr>
<tr>
<td>6</td>
<td>3.000</td>
<td>1.000</td>
<td>0.97±0.03</td>
<td>1.020</td>
</tr>
<tr>
<td>13</td>
<td>15.71</td>
<td>1.000</td>
<td>1.00±0.03</td>
<td>1.065</td>
</tr>
<tr>
<td>20</td>
<td>28.43</td>
<td>1.000</td>
<td>1.03±0.03</td>
<td>1.140</td>
</tr>
<tr>
<td>27</td>
<td>41.15</td>
<td>1.046</td>
<td>1.04±0.03</td>
<td>1.365</td>
</tr>
<tr>
<td>34</td>
<td>53.86</td>
<td>1.215</td>
<td>1.19±0.03</td>
<td>1.620</td>
</tr>
<tr>
<td>41</td>
<td>66.58</td>
<td>1.410</td>
<td>1.33±0.03</td>
<td>1.875</td>
</tr>
<tr>
<td>48</td>
<td>79.30</td>
<td>1.630</td>
<td>1.65±0.06</td>
<td>2.175</td>
</tr>
<tr>
<td>55</td>
<td>92.01</td>
<td>1.880</td>
<td>1.81±0.06</td>
<td>2.468</td>
</tr>
<tr>
<td>62</td>
<td>104.73</td>
<td>2.150</td>
<td>2.10±0.06</td>
<td>2.760</td>
</tr>
<tr>
<td>69</td>
<td>117.44</td>
<td>2.420</td>
<td>2.41±0.06</td>
<td>3.030</td>
</tr>
<tr>
<td>76</td>
<td>130.16</td>
<td>2.725</td>
<td>2.71±0.06</td>
<td>3.315</td>
</tr>
<tr>
<td>83</td>
<td>142.88</td>
<td>3.025</td>
<td>3.01±0.06</td>
<td>3.585</td>
</tr>
<tr>
<td>90</td>
<td>155.60</td>
<td>3.275</td>
<td>3.30±0.06</td>
<td>3.750</td>
</tr>
<tr>
<td>97</td>
<td>168.31</td>
<td>3.575</td>
<td>3.58±0.06</td>
<td>3.945</td>
</tr>
<tr>
<td>104</td>
<td>181.03</td>
<td>3.850</td>
<td>3.83±0.06</td>
<td>4.110</td>
</tr>
<tr>
<td>111</td>
<td>193.74</td>
<td>4.075</td>
<td>4.10±0.08</td>
<td>4.185</td>
</tr>
<tr>
<td>118</td>
<td>206.46</td>
<td>4.275</td>
<td>4.30±0.08</td>
<td>4.200</td>
</tr>
<tr>
<td>125</td>
<td>219.17</td>
<td>4.425</td>
<td>4.47±0.09</td>
<td>4.275</td>
</tr>
<tr>
<td>132</td>
<td>231.88</td>
<td>4.500</td>
<td>4.60±0.12</td>
<td>4.275</td>
</tr>
</tbody>
</table>

$^a$ Normalized to Position 83

$^b$ Errors assigned from the deviation from the mean of two measurements except for the last 5 space points. This error is usually larger than the counting statistics error. The error of the last 5 points was calculated from counting statistics.
FIG. 29 CALCULATED AND EXPERIMENTAL DELAY TIMES
- 0.5 MSEC INPUT PULSE -
FIG. 30  CALCULATED AND EXPERIMENTAL DELAY TIMES
- 1.0 MSEC INPUT PULSE -
ASYMPTOTIC INVERSE
RELAXATION LENGTH

Theory 0.03025
Exp. 0.0297 ± 0.0012

FIG. 31 AMPLITUDE ATTENUATION OF THE THERMAL NEUTRON FLUX
- 0.5 MSEC INPUT PULSE -
FIG. 32 AMPLITUDE ATTENUATION OF THE THERMAL NEUTRON FLUX
- 1.0 MSEC INPUT PULSE -
"asymptotic" region of the assembly. In this region a linear relationship between delay times and distance from the source exists. The asymptotic velocity of propagation for the clean UFSA core is given in Table IX.

<table>
<thead>
<tr>
<th>Input Pulse Width (msec)</th>
<th>Theory</th>
<th>v_p (m/sec)</th>
<th>Experiment</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.5</td>
<td>456</td>
<td>448±9</td>
<td></td>
</tr>
<tr>
<td>1.0</td>
<td>456</td>
<td>452±7</td>
<td></td>
</tr>
</tbody>
</table>

It should be mentioned that the asymptotic velocity of propagation is independent of the assumed distribution of the source. When a localized source in space was introduced into the WIGLE scheme the results yielded the same asymptotic velocity as the calculations with a spatially distributed source. This is to be expected since v_p is a characteristic of the system and has been defined in a region removed from source effects.

The Dynamic Inverse Relaxation Length

The dynamic inverse relaxation length, κ_d, is defined as the inverse of the distance required for the amplitude of the pulses to attenuate by a factor e. κ_d is determined by the relationship between the logarithm of the magnitude of the peak of the propagating pulses and distance in the "asymptotic" region where this relationship is linear.
The attenuation of the amplitude of the peaks for the UFSA assembly were shown in Figs. 31 and 32; the normalized peak values were tabulated in Tables VII and VIII. Again excellent agreement was obtained between theory and experiment, not only in the asymptotic region but throughout most of the assembly. The first two data points (P6 and P13) are considerably larger in magnitude than the values predicted by WIGLE. This is probably caused by an undercorrection for the epicadmium flux detected at these positions and by not accounting for the significant neutron streaming in the large void created by the aluminum port in which the target is located. Position 27 is probably the only "bad" data point and no explanation has been found for its behavior since four consecutive runs were made at that position and all four showed identical behavior. The WIGLE results reflect the influence of end effects more markedly than experiment. It should be noted that reflections from the concrete walls in the assembly room are significant at peripheral detector positions.

<table>
<thead>
<tr>
<th>Input Pulse Width (msec)</th>
<th>( \kappa_d ) (cm(^{-1}))</th>
<th>Theory</th>
<th>Experiment</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.5</td>
<td>.03025</td>
<td>.0297+ .0012</td>
<td></td>
</tr>
<tr>
<td>1.0</td>
<td>.03025</td>
<td>.0301+ .0010</td>
<td></td>
</tr>
</tbody>
</table>
**The Full Width at Half-Maximum (FWHM)**

The FWHM is defined as the time-width of the pulse at half the peak amplitude. The FWHM is one of the parameters conventionally used to compare pulse shapes and is therefore included in this study.

An analysis of the experimental and calculated values of the FWHM tabulated in Tables VII and VIII showed the following.

a) Close to the source, theory predicts slightly larger FWHM values than experiment, the difference decreasing with distance from the source. As discussed above the discrepancy near the source is due to difficulties in adequately describing the void region behind the source.

b) Between P27 and P34 the FWHM are practically identical.

c) Past P34 the experimentally determined FWHM becomes increasingly larger than the WIGLE prediction. The maximum difference is about 350 microsec, which corresponds to ≈ 8% of the experimental value. The agreement is still considered to be very good.

**The Flux Shapes**

A detailed comparison of the theoretical and experimental flux shapes obtained for the UFSA assembly was also made.

Plots of the time profiles of the thermal flux at each detector position are included in Appendix C for the 0.5 and the 1.0 msec wide source pulses. The peaks are normalized to unity.

The following information is supplied in each figure:

a) Identifying Run Number

b) Location of the Experimental Peak in Time

This parameter is determined directly by a computer routine which is part of the plotter program. The routine simply finds the time channel
in which the largest number of counts was recorded and in many instances is not the best value for the location of the peak. The values quoted in Tables VII and VIII represent refined values obtained by inspection of the experimental data.

c) The Input Pulse Width
d) The Experimental and Theoretical FWHM
e) The Location with Respect to the Source is given in the figure title.

A comparison between the theoretical and experimental pulse shapes reveals the following:

a) Shapes agree very closely in the spatial region dominated by the neutron source. Small discrepancies start to show up near P41.
b) Past P41 the theoretical pulses become consistently narrower than the experimental one. As discussed above, the other main characteristics describing the propagation of the pulses are well matched.

Discussion of the Clean Core Results

The fact that good agreement was found between the theoretical and experimental results deserves some elaboration since the agreement may seem fortuitous.

The calculational scheme used to obtain the WIGLE input parameters is well established. The parameters are therefore considered to be as reliable as the present state of the art permits. The main uncertainty as far as the calculational model is concerned is in the allocation of a transverse buckling.

To investigate the sensitivity of the two-group, one-dimensional model to changes in the transverse buckling (incorporated into the WIGLE scheme as a change in the absorption cross-section) a series of
calculations were performed in which the height of the assembly was varied from 70 to 85 cm. The respective core heights with their corresponding vertical bucklings, $k_{\text{eff}}$ (calculated using AIM-6) and "absorption" cross-sections used in the WIGLE code are listed in Table XI.

The results of the calculations showed a marked sensitivity to changes in the transverse buckling. The delay times and the peak counts vs. distance to the source are shown in Fig. 33 and 34 respectively. The experimental results for 76 cm are also shown. The data for the attenuation curves was normalized to the attenuation at P83 of the 76 cm WIGLE calculations. Pulse shapes for the different heights are shown together with the experimental results at 76 cm in Fig. 35 (A, B, C) for three representative positions in the assembly.

Some additional experimental measurements were also made at a given position for different core heights to supplement the calculational predictions. It should be pointed out that these measurements are of a preliminary nature, made over a short period of time simply to obtain confirmation of the theoretical trends. The results of these measurements are shown in Fig. 36.

It is not surprising that the results for the core showed as much sensitivity to changes in the transverse buckling since the width and height of the core are rather small in comparison to the length of the assembly. It is to be noted that the changes in the "absorption" cross-section due to changes in the transverse buckling are rather small and are reflected in the third significant figure in the fast group cross-section and the fourth significant figure in the thermal group cross-section. This emphasizes the importance of an accurate
TABLE XI

CHANGES IN THE NUCLEAR PARAMETERS
DUE TO CHANGES IN THE TRANSVERSE BUCKLING

UFSA R1 Core
0.5 M/W Ratio
16.35 cm wide reflected core

<table>
<thead>
<tr>
<th>Reactor Height (cm)</th>
<th>Estimated $k_{eff}$</th>
<th>Vertical Buckling (cm$^{-2}$)</th>
<th>$\Sigma_{a1}$</th>
<th>$\Sigma_{a2}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>70.0</td>
<td>.986</td>
<td>20.14 E-04</td>
<td>.054671</td>
<td>.12223</td>
</tr>
<tr>
<td>72.5</td>
<td>.988</td>
<td>18.78 E-04</td>
<td>.054526</td>
<td>.122207</td>
</tr>
<tr>
<td>76</td>
<td>.990</td>
<td>17.08 E-04</td>
<td>.054345</td>
<td>.12217</td>
</tr>
<tr>
<td>77.5</td>
<td>.992</td>
<td>16.43 E-04</td>
<td>.054277</td>
<td>.122158</td>
</tr>
<tr>
<td>80.0</td>
<td>.995</td>
<td>15.42 E-04</td>
<td>.054169</td>
<td>.122137</td>
</tr>
<tr>
<td>85.0</td>
<td>.998</td>
<td>13.66 E-04</td>
<td>.053981</td>
<td>.122096</td>
</tr>
</tbody>
</table>

WIGLE X-Section (cm$^{-1}$)
UFSA R1 Clean Core
12.716 cm between points
0.5 msec input pulse
Experiment at 76 cm

FIG. 33  DELAY TIMES OF THE THERMAL FLUX CALCULATED BY THE
WIGLE CALCULATIONAL SCHEME FOR DIFFERENT CORE HEIGHTS
FIG. 34  AMPLITUDE ATTENUATION OF THE THERMAL FLUX CALCULATED BY THE WIGLE CALCULATIONAL SCHEME FOR DIFFERENT CORE HEIGHTS

R1 Clean Core
12.716 cm between points
0.5 msec input pulse width
0 Experiment at 76 cm

Curves i.d. by core height
FIG. 35A THE SENSITIVITY OF THE ONE-DIMENSIONAL, TWO GROUP, SPACE-TIME KINETICS SCHEME TO CHANGES IN THE TRANSVERSE BUCKLING
FIG. 35B THE SENSITIVITY OF THE ONE-DIMENSIONAL, TWO GROUP, SPACE-TIME KINETICS SCHEME TO CHANGES IN THE TRANSVERSE BUCKLING
FIG. 35C THE SENSITIVITY OF THE ONE-DIMENSIONAL, TWO GROUP, SPACE-TIME KINETICS SCHEME TO CHANGES IN THE TRANSVERSE BUCKLING
FIG. 36  EXPERIMENTAL PULSE SHAPES AS A FUNCTION OF CORE HEIGHT
evaluation of the nuclear parameters.

The asymptotic propagation velocity and dynamic inverse relaxation length as a function of core height are shown in Table XII. As expected, the velocity of propagation and the attenuation decrease as criticality is approached.

The above analysis puts the good agreement between experiment and theory on a firmer footing. When the observed phenomenon is very sensitive to small variations in system characteristics the probability that fortuitous agreement may be obtained between theory and experiment diminishes with the degree of sensitivity.

Comments on the Fast Group Results

The following comments are pertinent regarding the fast group results as calculated by the two-group WIGLE model:

1. The curve of delay times vs. distance remains flatter than the corresponding curve for the thermal group near the source region; it then bends rapidly to yield the same asymptotic velocity of propagation as that of the thermal group.

2. The fast group flux peaks between positions P6 and P13; this has been confirmed experimentally. In contrast the thermal group flux peaks at P6. The asymptotic inverse relaxation length of the fast group is, as expected, identical to the one determined for the thermal group.

3. The fast group pulse profiles are slightly wider than those for the thermal group.

The time profiles from the cadmium-covered detector measurements, which were used to determine the thermal group pulse profiles by the epicadmium subtraction method – were compared with the calculated fast
TABLE XII

THE CALCULATED ASYMPTOTIC VELOCITY OF PROPAGATION AND DYNAMIC INVERSE RELAXATION LENGTH VS. CORE HEIGHT
- 0.5 MSEC INPUT PULSE WIDTH -

UFSA R1 Core
0.5 M/W Ratio
16.35 cm wide reflected core

<table>
<thead>
<tr>
<th>Core Height (cm)</th>
<th>K_d (cm⁻¹)</th>
<th>V_p (msec⁻¹)</th>
</tr>
</thead>
<tbody>
<tr>
<td>70</td>
<td>0.03565</td>
<td>571</td>
</tr>
<tr>
<td>72.5</td>
<td>0.03233</td>
<td>506</td>
</tr>
<tr>
<td>76</td>
<td>0.03027</td>
<td>456</td>
</tr>
<tr>
<td>77.5</td>
<td>0.02772</td>
<td>435</td>
</tr>
<tr>
<td>80</td>
<td>0.02515</td>
<td>397</td>
</tr>
<tr>
<td>85</td>
<td>0.02047</td>
<td>321</td>
</tr>
</tbody>
</table>
group pulse shapes. Several typical examples are shown in Appendix D. The following features were observed:

1. The calculated pulse shapes are delayed in time in comparison to the measured shapes throughout most of the length of the assembly.

2. The measured and calculated attenuation of the peaks is in poor agreement.

3. The measured and calculated pulse shapes are practically identical.

The validity of the comparison between theory and experiment in this instance is not clear, however. The main question is whether the experimental measurement corresponds to the fast group used in the calculations, since the sensitivity of the detector rapidly diminishes with increasing neutron energies. Therefore the conclusions based on the above comparison must be taken with some reservation, until a more detailed study can be conducted.

**Propagation of a Narrow Pulse**

A series of measurements were made in the clean core utilizing a 100 microsec wide input pulse. Only 4 positions along the longitudinal axis were studied. The purpose of these measurements was to obtain a qualitative idea of how well the WIGLE scheme can predict the space–time kinetic behavior of a system perturbed by a very narrow pulse. From the experimental viewpoint the narrow input pulse resulted in much poorer count rates, thus requiring longer data acquisition times.

Listed below are the calculated and measured FWHM and delay times for these measurements; the pulse shapes at the different positions are included in Appendix E.
TABLE XIII

DELAY TIMES AND FWHM FOR A NARROW INPUT PULSE

<table>
<thead>
<tr>
<th>DISTANCE TO SOURCE (cm)</th>
<th>PEAK AT (msec)</th>
<th>FWHM (msec)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Theory</td>
<td>Exp</td>
</tr>
<tr>
<td>53.9</td>
<td>0.60</td>
<td>0.580</td>
</tr>
<tr>
<td>79.3</td>
<td>1.08</td>
<td>1.06</td>
</tr>
<tr>
<td>130.2</td>
<td>2.27</td>
<td>2.26</td>
</tr>
<tr>
<td>142.9</td>
<td>2.50</td>
<td>2.53</td>
</tr>
</tbody>
</table>

Significant discrepancies between theoretical and experimental pulse shapes were noted in the region where source effects are still noticeable; in contrast, excellent agreement was found in this region for the 0.5 and 1.0 msec input pulse cases. Deep in the asymptotic region the measured pulse shapes are identical to those determined for the 0.5 and 1.0 msec input pulses.

The results are somewhat inconclusive, however, since near the source the measured pulse shapes appeared wider than the corresponding shapes for the 0.5 and 1.0 msec input pulsed, although they occur earlier in time. Unfortunately, the assembly was dismantled before these comparisons were completed. The observed discrepancies deserve further experimental study.

It should also be noted that noticeable oscillations were observed in the WIGLE results in the spatial region close to the source.

Propagation of a Wide Pulse

A series of measurements were also made in the clean core with a 10 msec wide input pulse. This pulse is much wider than the
characteristic propagation time (~3.7 msec) across the assembly of a narrow (0.5 msec) pulse. It should be recalled that the Johnson criterion [47] postulates that to be able to observe spatially dependent effects, the FWHM of the input pulse should be smaller than the characteristic propagation time across the assembly. For example, the experiments of Miley and co-workers at the University of Illinois [20, 21, 22] showed that for an initially asymptotic very wide input pulse no shape changes were observed as the pulse "propagated" across the assembly.

The pulse shapes resulting from these measurements at three positions in the assembly are shown in Appendix F. The delay times, the FWHM and area under the pulses, normalized to unity at the first position, are given below in Table XIV.

**TABLE XIV**

**DELAY TIMES AND FWHM FOR A WIDE INPUT PULSE**

<table>
<thead>
<tr>
<th>DISTANCE TO SOURCE (cm)</th>
<th>PEAK AT PEAK (msec)</th>
<th>FWHM (msec)</th>
<th>AREA UNDER THE CURVE</th>
</tr>
</thead>
<tbody>
<tr>
<td>41.2</td>
<td>10.07</td>
<td>10.1</td>
<td>1.0</td>
</tr>
<tr>
<td>79.3</td>
<td>10.17</td>
<td>10.7</td>
<td>1.15</td>
</tr>
<tr>
<td>142.9</td>
<td>10.22</td>
<td>10.5</td>
<td>0.92</td>
</tr>
</tbody>
</table>

From these results it seems that, although the delay times between the recorded peaks are quite small a certain "rearrangement" of the initially non-asymptotic pulse takes place. Close to the source the pulse is practically square, at a distance of 79 cm it has widened somewhat and at large distances from the source the pulse seems to be trying to achieve the normal "dumbbell" shape. Johnson's criterion appears to
be too restrictive, since some spatial redistribution of the input pulse does occur, although the effect is certainly much less pronounced than in the case of a narrow pulse. In the case of the Illinois experiments the input pulse was \(^\sim\) 30 msec wide and already had an asymptotic shape, with the characteristic propagation time being of the order of a few msec.

In the next section the propagation of pulses as a function of the input pulse width will be studied qualitatively. These measurements were made to complement the above results.

**Pulse Shape vs. Input Pulse Width**

The analysis of the measurements with input pulse widths of 0.1, 0.5 and 1.0 msec revealed that these propagating pulses achieved identical shapes, within the experimental accuracy, at distances larger than 130 cm from the source. This fact was believed to hold for input pulses much narrower than the characteristic propagation time in the assembly. A series of pulse shapes were measured as a function of input pulse width at a distance of 142.9 cm from the source; these measurements should provide information on the qualitative behavior of propagating pulses arising from widely differing input pulse widths.

The results of these measurements are shown in Appendix G; the peak times and the FWHM for each input pulse recorded at P83 are given in Table XV.

As previously observed, the peaks are increasingly displaced in time as the input pulse widens. This behavior has been well predicted by the two-group, space-time dependent diffusion theory model. It appears that for the 2 msec wide input pulse deviations from the
### TABLE XV

**PULSE SHAPES VS. INPUT PULSE WIDTH**

UFSA R1 Core  
0.5 M/W Ratio  
16.35 cm wide reflected core

<table>
<thead>
<tr>
<th>INPUT PULSE WIDTH (msec)</th>
<th>PEAK AT FWHM (msec)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>(msec)</td>
</tr>
<tr>
<td>.1</td>
<td>2.53</td>
</tr>
<tr>
<td>.5</td>
<td>2.65</td>
</tr>
<tr>
<td>1.0</td>
<td>3.04</td>
</tr>
<tr>
<td>2.0</td>
<td>3.79</td>
</tr>
<tr>
<td>3.0</td>
<td>4.39</td>
</tr>
<tr>
<td>4.0</td>
<td>4.72</td>
</tr>
<tr>
<td>5.0</td>
<td>6.02</td>
</tr>
<tr>
<td>6.0</td>
<td>6.97</td>
</tr>
<tr>
<td>7.0</td>
<td>7.67</td>
</tr>
<tr>
<td>8.0</td>
<td>8.62</td>
</tr>
<tr>
<td>9.0</td>
<td>9.67</td>
</tr>
<tr>
<td>10.0</td>
<td>10.22</td>
</tr>
</tbody>
</table>
"characteristic asymptotic" pulse shape already exists. Characteristic "dumbbell" shapes are observed however until the HWHM propagation time. For wider input pulses, the "propagating" pulses are unable to achieve the smooth shape observed for narrower pulses. Still, a certain spatial "re-arrangement" of the shape occurs, as was discussed above for the 10 msec input pulse.

Effect of Room Return at Peripheral Detector Positions

At positions farthest removed from the source, it was observed that a secondary peak was occurring at times corresponding to the input pulse width. This was also observed at peripheral positions in the reflector tanks. To prove that neutrons reflected from the concrete walls were the cause of this secondary peak, measurements were made at the outer edge of the reflector tanks with the neutron detector shielded by borated paraffin placed outside the reflector tank and with the detector unshielded.

The results of these measurements are shown in Fig. 37A. A pronounced peak at ~0.5 msec was found when the detector was not shielded from the walls and practically disappeared when about 6" of borated paraffin was placed between the reflector tank and the concrete wall. The input pulse width was 0.5 msec. Shown in Fig. 37B is the result of subtracting the "shielded" pulse from the "unshielded" one. A sharp 0.5 msec pulse is obtained. The "early" peaks are therefore blamed on fast neutrons reflected from the walls.

The penetration of these neutrons into the reflectors was analyzed by positioning a detector at increasing distances from the outer reflector tank wall. It was found that the effect was negligible at a
FIG. 37A

EFFECT OF ROOM RETURN AT PERIPHERAL DETECTOR POSITIONS

NORMALIZED TO
R1, REF REF 76
PULSE WIDTH = 0.5 MSEC
EXP. FWHM = 3.570 MSEC
R1REFREF76
PEAK AT 0.450 MSEC
PULSE WIDTH = 0.5 MSEC
EXP. FWHM = 0.570 MSEC

FIG. 37B EFFECT OF ROOM RETURN AT PERIPHERAL DETECTOR POSITIONS
distance of 5 cm from the outer wall. At the far end of the assembly the reflected pulse "propagated" into the assembly, attenuating and dispersing with distance to the end wall. This pulse quickly dissipated into the assembly with no significant effect on the measurements.
CHAPTER V

EXPERIMENTAL AND THEORETICAL
RESULTS IN THE FREQUENCY DOMAIN

Introduction

The results of the pulse propagation measurements and of the spacetime-dependent WIGLE calculations were analyzed in the sensitive Fourier transform plane. This neutron wave type analysis probes into the deficiencies of the model and/or the experimental data that the analysis in the time domain might fail to reveal. In this study the analysis is performed with the sole intention of testing the model and not to extract parameters.

The WIGLE predicted time profiles of the propagating disturbance as a function of space and the pulse propagation data were Fourier analyzed in an identical manner; the amplitude and phase angle of the zeroth Fourier moment thus obtained were least-squares fitted (log amplitude vs. z and phase angle vs. z for each frequency of interest) yielding the damping coefficient, $\alpha$, and the phase shift per unit length, $\xi$. The predicted and measured dispersive characteristics of the system are then compared in the $\rho$ and $\rho^2$ plane.

The neutron wave analysis was applied to both the 0.5 and 1.0 msec input pulse space-time measurements and calculations.
Method of Analysis

The following method of analysis was applied to the calculated and measured results of the UFSA R1 clean core space-time studies:

1. A conventional zeroth moment Fourier transformation was performed on the experimental data. The numerical integration employed Simpson's rule of integration. Equal time steps (30 microsec) were used throughout this analysis. The number of data points varied from 500 to 800, depending on where a stable neutron background was reached, as determined by an statistical analysis performed in the UNIPUL program. A computer code, MORE, was coded for this purpose.1

2. A conventional zeroth moment Fourier transformation was performed on the space-time results obtained from the WIGLE calculations. It should be recalled that the WIGLE calculation employs a series of time increments in different time intervals. To conform to this scheme Simpson's rule of integration with different time increments was used for each time interval. Trapezoidal integration was used to bridge the gap between the different time intervals. A computer program, MORWIG, was coded for this purpose.

3. The amplitudes and phases of the zeroth Fourier moment as a function of frequency, obtained from the MORE and MORWIG numerical transformation, were least-squares fitted to obtain the damping coefficient and the phase shift per unit length for the frequencies of interest.

In this manner, the theoretical and experimental results in the

1. The MORE and MORWIG programs were coded by Dr. M. J. Ohanian, University of Florida.
time domain were transformed to the frequency domain, with identical numerical and fitting procedures, setting a firm basis for a one-to-one comparison.

The following interesting numerical problems are worthy of note:

a) To determine to what extent the numerical transformation is affected by the accuracy of the operations of the IBM 360 computer, sample problems were run in single precision (4 byte words) and double precision (8 byte words) using the MORWIG program. No differences were found in the results up to the sixth significant figure, but the double precision computations consumed almost twice the amount of time as the single precision ones.

b) Initially, the results of every third WIGLE time calculation was punched on cards for the comparison in the time domain and for the input to the Fourier transformation code. Less than 250 time points, with time increments of up to 150 microsec, were available for each space point. The numerical transformation performed on these results was somewhat unsatisfactory since not all the amplitude and phase curves were smooth and in the $p^2$ plane the scattering of points was significant. The WIGLE calculations were then performed again, with smaller time increments, carried farther in time and every time step punched for input to the MORWIG program. A total of 997 and 998 points respectively were now available for the Fourier transformation for the 0.5 and 1.0 msec input pulse cases. The largest time increment was 32 microsec. The calculation was carried to about 19.2 msec after the initiation of the pulse. A significant improvement was noted in the results of the numerical transformation. The results were smooth throughout up to ~ 800 cps, where a certain amount of scattering was observed.
Comparison of the Theoretical and the Measured
Results of the Neutron Wave Analysis

To determine the effect that the input pulse width could have on the results of the wave analysis, the study was performed for both the 0.5 and 1.0 msec input pulse cases. These experimental and theoretical results in the time domain were presented in Part 2, Chapter IV of this work.

Shown in Figs. 38 through 41 are the amplitude and phase of the experimental data obtained by numerical Fourier transformation, for both the 0.5 and the 1.0 msec input pulses. The results for the 0.5 msec case are consistently "smoother" than those for the 1.0 msec case. In the 0.5 msec case the results "blow-up" past 1000 cps while the 1 msec case seems to be good only to 800 cps. The behavior of the amplitude and phase is as expected. The data was still good in a frequency region were most wave type measurements in multiplying media reported to date had already broken down. For example the experiments of Dunlap were only good to ~ 250 cps [23]. Undoubtedly, the frequency content of the narrow, fast burst is superior to the smeared pulse that is normally obtained if a thermalizing tank is used. The wave analysis on the WIGLE space-time results showed an identical behavior throughout the frequency range.

The real and the imaginary components of the complex inverse relaxation length are shown for both theory and experiment in Tables XVI and XVII, for the 0.5 and the 1.0 msec input pulses respectively. A graphical comparison of these results is shown in Figs. 42 and 43.

The following comments are pertinent:

a) A distinct but not significant difference is found between the results of the 0.5 and the 1.0 msec cases in both theory and experiment.
FIG. 38  AMPLITUDE OF ZEROTH FOURIER MOMENT vs. DISTANCE FOR SEVERAL FREQUENCIES - 0.5 MSEC INPUT PULSE -
FIG. 39  AMPLITUDE OF ZEROTH FOURIER MOMENT vs. DISTANCE FOR SEVERAL FREQUENCIES
- 1.0 MSEC INPUT PULSE -
FIG. 40  PHASE OF ZEROTH FOURIER MOMENT vs. DISTANCE FOR SEVERAL FREQUENCIES
- 0.5 MSEC INPUT PULSE -
FIG. 41  PHASE OF ZERO TH FOURIER MOMENT vs. DISTANCE FOR SEVERAL FREQUENCIES
- 1.0 MSEC INPUT PULSE -
### TABLE XVI

**The Real and the Imaginary Components of the Complex Inverse Relaxation Length - 0.5 msec Input Pulse**

<table>
<thead>
<tr>
<th>Frequency (cps)</th>
<th>$\alpha$ (cm(^{-1}))</th>
<th>$\xi$ (rad/cm)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Theory</td>
<td>Exp</td>
</tr>
<tr>
<td>0</td>
<td>0.02286</td>
<td>0.02248</td>
</tr>
<tr>
<td>10</td>
<td>0.02290</td>
<td>0.02254</td>
</tr>
<tr>
<td>20</td>
<td>0.02305</td>
<td>0.02270</td>
</tr>
<tr>
<td>30</td>
<td>0.02328</td>
<td>0.02296</td>
</tr>
<tr>
<td>40</td>
<td>0.02359</td>
<td>0.02331</td>
</tr>
<tr>
<td>50</td>
<td>0.02396</td>
<td>0.02373</td>
</tr>
<tr>
<td>60</td>
<td>0.02438</td>
<td>0.02423</td>
</tr>
<tr>
<td>70</td>
<td>0.02483</td>
<td>0.02476</td>
</tr>
<tr>
<td>80</td>
<td>0.02532</td>
<td>0.02536</td>
</tr>
<tr>
<td>100</td>
<td>0.02640</td>
<td>0.02663</td>
</tr>
<tr>
<td>120</td>
<td>0.02755</td>
<td>0.02797</td>
</tr>
<tr>
<td>140</td>
<td>0.02866</td>
<td>0.02937</td>
</tr>
<tr>
<td>160</td>
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<tr>
<td>1000</td>
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<td>0.07404</td>
</tr>
</tbody>
</table>
TABLE XVII

THE REAL AND THE IMAGINARY COMPONENTS OF
THE COMPLEX INVERSE RELAXATION LENGTH
- 1.0 MSEC INPUT PULSE -

<table>
<thead>
<tr>
<th>FREQUENCY (cps)</th>
<th>( \alpha ) (cm(^{-1}))</th>
<th>( \xi ) (rad/cm)</th>
</tr>
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<tbody>
<tr>
<td></td>
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<td>Exp</td>
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<tr>
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<td>0.06000</td>
<td>0.06396</td>
</tr>
<tr>
<td>900</td>
<td>0.06530</td>
<td>0.06692</td>
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</tbody>
</table>
Comparison of the theoretically predicted and the measured damping coefficient $\alpha$.
FIG. 43 COMPARISON OF THE THEORETICALLY PREDICTED AND THE MEASURED PHASE SHIFT PER UNIT LENGTH $\xi$
The predicted and measured values are in excellent agreement up to 200 cps; theory follows closely the measured difference between the 0.5 and the 1.0 msec cases. The differences in the damping coefficient at zero frequency are somewhat puzzling. A steady-state measurement of the inverse relaxation length yielded a value of 0.0221 cm\(^{-1}\), in excellent agreement with the experimental 0.5 msec case and \(\sim 5\%\) lower than the 1.0 msec case.

b) The theoretical and experimental \(\alpha\)'s and \(\xi\)'s start to diverge "significantly" past 200 cps. The deviation is larger for the 1.0 msec input case. The deviation is \(\sim 6\%\) at 400 cps for \(\alpha\) and \(\xi\) (0.5 msec case), and about \(8\%\) at 800 cps. Still both the theoretical and experimental results were smooth functions of frequency throughout the range; both collapse dramatically at \(\sim 1000\) cps.

c) An analysis of the least-squares fitting performed on both the predicted and the measured values reveals that a "poor" fit is obtained at 0 cps. The fit gets increasingly better with frequency up to 400 cps were fluctuations are observed in the values. The criteria used to select the value of \(\alpha\) and \(\xi\) quoted were that of spatial convergence, i.e., the chopping technique was employed to determine a spatially converged value. This was not always encountered, especially past 500 cps in both theory and experiment. A judgement was made to select an appropriate value in each case where convergence was not obtained. This made the theoretical and experimental results look worse at high frequencies because they had different trends and the same criteria had to be used for both.

The system \(\rho\) dispersion law is shown in Fig. 44. The agreement can be termed good, considering the percentage differences and other previously reported results [23]. A smooth trend is followed until theory
FIG. 44 THE UFSA R1 CORE ρ DISPERSION LAW
and experiment approach 1000 cps.

The differences observed reflect those noticed when the comparison in the time domain was conducted. The fact that the WIGLE pulses are consistently narrower than the experimental ones is reflected in the smaller damping coefficients at large frequencies.

The differences showed up much more markedly in the ultrasensitive $\rho^2$ plane. Since $\alpha_{th} < \alpha_{exp}$ and $\xi_{th} > \xi_{exp}$, the real part of $\rho^2=(\alpha^2-\xi^2)$ seems to indicate a discrepancy that is not as large as it appears.

Shown in Fig. 45 is the real part of $\rho^2$ for both theory and experiment as a function of frequency. Although they start to diverge significantly at $f = 100$ cps, it should be remarked that any small differences in $\alpha$ and $\xi$ are magnified out of proportion in the $\rho^2$ plane. In contrast, the imaginary part of $\rho^2=(2\alpha\xi)$ appears to signify perfect agreement in the results, since the deviations in $\alpha$ and $\xi$ cancel each other (see Fig. 46).

The $\rho^2$ dispersion law of the UFSA R1 clean core is shown in Fig. 47. This is a most sensitive index. Any small error in the data or failure of the model will immediately appear "blown-up" in $\rho^2$. Although the theory and experiment diverge in the $\rho^2$ plane, after 100 cps, both behave smoothly; the differences should be considered within the context of the overall analysis. Certainly it can be said that the WIGLE results do not follow the experimental results throughout the frequency range but that is to be expected; the actual discrepancies observed in the time domain and the numerical uncertainties propagate into the $\rho^2$ plane so significantly that perfect agreement seems to be unreachable with the present models and knowledge of parameters.

The steady-state inverse relaxation length $= .023$ (and the corresponding $\alpha_{f=0}$) is significantly smaller than the dynamic inverse relaxation $= .030$. 
FIG. 45  COMPARISON OF THE THEORETICALLY PREDICTED AND THE MEASURED $(\alpha^2 - \xi^2)$
FIG. 46 COMPARISON OF THE THEORETICALLY PREDICTED AND THE MEASURED $2 \alpha \xi$
FIG. 47  THE UFSA R1 CORE $\rho^2$ DISPERSION LAW
CHAPTER VI

SPATIAL DEPENDENCE OF PULSED-NEUTRON REACTIVITY MEASUREMENTS

Introduction

A resume of pulsed-neutron reactivity measurements was presented in Part 2, Chapter V of this work. During the nuclear calibration of the UFSA assembly certain "spatial" effects were observed in the determination of the reactivity of the long assembly. In this chapter these effects are investigated.

The pulse propagation measurements performed in the UFSA R1 clean core were analyzed using the Garelis-Russell technique [8] to determine the ratio $k\beta/\lambda$; the "fundamental" decay constant was obtained by applying Peierl's statistical analysis to the tail of the pulse and the effective delayed neutron fraction was calculated and assumed constant with respect to reactivity. A strong "spatial" dependence of the measurements conducted along the center line of the core was observed. An asymptotic spatial distribution was obtained long times after the pulse; at these times only the positions farthest from the neutron source had not reached background level. The Garelis-Russell technique was found to be very sensitive to the finite width of the input pulse.

A computer program, UNIPUL, was coded and used to perform the Garelis-Russell type analysis on the impulse response curves determined as a function of position in the UFSA R1 core. The same program
yielded the decay constants at different times after the pulse and computed a value of $\rho(\delta)$, as well as absolute $\rho$ and $k_{\text{eff}}$ using a calculated $\rho_{\text{eff}}$ for each decay constant.

The Decay Constant

A series of decay constants were obtained by analyzing the tail of the pulse at increasing times from the peak. The convergence of the decay constant in time was sought, before background is reached, and the value that gave the least deviation according to Peierl's statistical analysis was considered to give the decay constant at that position.

The decay constants determined close to the source decreased monotonically as the fitting was performed farther away in time from the peak. Once the "asymptotic" source region, as defined in the pulse propagation measurements, was reached, an apparent decay constant was found at each position. This decay constant is observed to converge or to vary very slowly with respect to time and decreased in magnitude as the distance to the source increased. At large distances from the source a fundamental decay constant is observed, seemingly when an asymptotic distribution of the neutrons is established in the assembly, more than 9 msec after the burst.

An analysis of the time-space behavior of the neutron flux (ref. Fig. 28) reveals that, indeed, more than 9 msec are required for the assembly to achieve an asymptotic neutron distribution. Therefore, only at large distances from the source can a true fundamental decay constant be measured; prior to P83 (142.9 cm) background is reached before a uniform spatial distribution is established.
Shown in Fig. 48 are the experimentally determined decay constants as a function of position in the assembly, for a 0.5 and 1.0 msec input pulse. As distance to the source increased, more "waiting" time was available to extract the decay constant. The values found past P83 are believed to represent the fundamental decay constant of the assembly.

It should be noted that the instantaneous decay constant at different times after the pulse was calculated for all the detector positions in the assembly, using the WIGLE code. The results agree closely with those measured experimentally at different times after the peak. The two-group, space-time dependent model describes well the observed phenomena.

The Ratio \( k_2/\lambda \)

The Garelis-Russell technique was applied to the two-medium system referred to as the UFSA R1 core. The measured \( k_2/\lambda \) values, determined as a function of distance from the source, showed a "spatial" dependence. This was somewhat surprising since the detector was supposedly located to minimize the Becker-Quisenberry correction [16] and a more uniform value of \( k_2/\lambda \) was expected. Although the model lacks a good energy representation of the system being studied such a strong spatial variation in \( k_2/\lambda \) was not expected.

Shown in Fig. 49 are the measured \( k_2/\lambda \) values for input pulses of 0.5 and 1.0 msec. A prominent feature is the noticeable difference between the results obtained with the two different, finite width input pulses. Attention was focused on the sensitivity of the Garelis-Russell model to the finite input pulse widths used in the experiments. As mentioned previously, the model utilizes a delta function input
FIG. 48  DECAY CONSTANT VS. AXIAL POSITION
FIG. 49 \( \frac{k_2}{\lambda} \) vs. AXIAL POSITION

UFSA RI Clean Core
12.72 cm between data points

0.5 msec input pulse width
1.0 msec input pulse width
pulse.

Tabulated in Table XVIII are values of $k\beta/\lambda$ determined using the Garelis-Russell technique on pulses arising from widely differing input pulses; the measurements were made at one position far from the neutron source. The effect of the finite pulse width on $k\beta/\lambda$ is significant. An analysis of the results shown in Fig. 49 points to the fact that at very large distances from the source a value of $k\beta/\lambda$ close to the one that would be obtained from a delta function input pulse is obtained.

Some comments on observed facts about the Garelis-Russell technique are pertinent. A dominant factor in the determination of $k\beta/\lambda$ is the delayed neutron background, which is strongly dependent on the repetition rate of the neutron generator (which should be determined very accurately). Meaningful values of $k\beta/\lambda$ are obtained far from the source for not too wide input pulses. A correction on the determined $k\beta/\lambda$ values to account for the finite width of the input pulses was developed by G. A. Mortensen, Nuclear Safety Division, Phillips Petroleum Co. [48] after the experimental observation of the phenomena. The suggested method was applied to several of the cases under consideration but an overcorrection seems to result; the "corrected" values keep changing drastically at long distances from the source where the influence of the input pulse is negligible. Further work in this correction should improve the results.

---

The Measured Reactivity and $k_{\text{eff}}$ Values

The reactivity in dollars is obtained, following the Garelis-Russell method, by the expression

$$\phi(\$) = \frac{a}{k\beta/\lambda} - 1$$
TABLE XVIII

THE DECAY CONSTANT AND kβ/λ VALUES MEASURED AS A FUNCTION OF INPUT PULSE WIDTH
- 0.5 MSEC INPUT PULSE WIDTH -

UFSA R1 Clean Core
0.5 M/W Ration
16.35 cm wide reflected core

<table>
<thead>
<tr>
<th>PULSE WIDTH (msec)</th>
<th>α (sec^{-1})</th>
<th>kβ/λ (sec^{-1})</th>
</tr>
</thead>
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<tr>
<td>0.1</td>
<td>511±10</td>
<td>213±4</td>
</tr>
<tr>
<td>0.5</td>
<td>488±11</td>
<td>202±5</td>
</tr>
<tr>
<td>1.0</td>
<td>501±6</td>
<td>193±5</td>
</tr>
<tr>
<td>2.0</td>
<td>503±8</td>
<td>183±7</td>
</tr>
<tr>
<td>3.0</td>
<td>510±4</td>
<td>172±9</td>
</tr>
<tr>
<td>4.0</td>
<td>523±16</td>
<td>169±9</td>
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<tr>
<td>5.0</td>
<td>494±6</td>
<td>144±10</td>
</tr>
<tr>
<td>6.0</td>
<td>504±5</td>
<td>136±10</td>
</tr>
<tr>
<td>7.0</td>
<td>505±5</td>
<td>126±9</td>
</tr>
<tr>
<td>8.0</td>
<td>522±17</td>
<td>118±11</td>
</tr>
<tr>
<td>9.0</td>
<td>524±13</td>
<td>111±8</td>
</tr>
<tr>
<td>10.0</td>
<td>496±16</td>
<td>108±9</td>
</tr>
</tbody>
</table>
Using a calculated value of $\beta_{\text{eff}} = 0.0079$ for this assembly configuration, the absolute value of the reactivity $\rho$, is obtained. The corresponding value of $k_{\text{eff}}$ is then found by the relation

$$k_{\text{eff}} = \frac{1}{1 + \rho}$$

Shown in Fig. 50 are the measured absolute reactivity and $k_{\text{eff}}$ as a function of position in the lattice. It is obvious from what has been said above regarding the behavior of $\alpha$ and $k\beta/\lambda$, that a meaningful value for either $k_{\text{eff}}$ or $\rho$ cannot be obtained in this case until $\approx 143$ cm separate the source and detector. The assigned value of $k_{\text{eff}}$ was $0.990 \pm 0.003$. This value compares very favorably with a value of $0.990 \pm 0.0025$ found by inverse multiplication measurements and a calculated value of 0.9906. Although the agreement obtained could be somewhat fortuitous, it can be said that a representative value of the reactivity was found. It seems that, if proper care is taken, the pulsed reactivity measurements will yield good results in large multiplying media.

Region-wise Dependence of the Reactivity Measurements

Becker and Quisenberry, as well as Waltar and Ruby [49] have pointed out the spatial dependence of pulsed source reactivity measurements in two-media systems. Differences in the measured reactivity can be present whether the detector is located in the core or in the reflector.

A series of pulsed source traverses were conducted across the UFSA R1 clean core and reflectors, at three different distances from the source. No differences were found for this core in the reactivity measured in the core or in the reflector. Shown in Table XIX is a
FIG. 50  REACTIVITY (-$) AND $k_{eff}$ VS. AXIAL POSITION
TABLE XIX

REGION-WISE DEPENDENCE OF THE REACTIVITY MEASUREMENTS

Measurement across the core width
UFSA R1 Clean Core
0.5 M/W Ratio
16.35 cm wide reflected core

<table>
<thead>
<tr>
<th>POSITION NO</th>
<th>DISTANCE TO CORE CENTER (cm)</th>
<th>$\alpha$ (sec$^{-1}$)</th>
<th>$k\beta/\lambda$ (sec$^{-1}$)</th>
<th>$\rho$</th>
<th>$k_{\text{eff}}$</th>
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</thead>
<tbody>
<tr>
<td>A</td>
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<td>205</td>
<td>0.01169</td>
<td>.9884</td>
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<tr>
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<td>497</td>
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<tr>
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<td>0.01168</td>
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<td>205</td>
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<td>.9887</td>
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<td>199</td>
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<td>31.8</td>
<td>474</td>
<td>187</td>
<td>0.01214</td>
<td>.9886</td>
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</tbody>
</table>
list of the decay constants, $k\beta/\lambda$, $\rho$ and $k_{eff}$ measured across the core and side reflector at a distance of 130.2 cm from the source.

The core-reflector configuration investigated is too narrow to observe any transverse propagation of the pulse; consequently no difference in the value of the reactivity was measured across the system. The value of $\alpha$ and of $k\beta/\lambda$ obtained near the outer edge of the reflector was low compared to the other values but somehow compensated each other to maintain $\rho$ approximately constant.
CHAPTER VII

CONCLUSIONS

The space-time kinetics behavior of a large-in-one-space dimension, side-reflected, highly multiplicative ($k_{\text{eff}} \sim 0.99$) subcritical assembly has been studied by the use of the pulse propagation technique. The experimental results were used to test the predictions of the one-dimensional two-group, space-time diffusion theory calculational scheme (WIGLE). The analysis of both the theoretically predicted and experimentally measured results was performed in the time as well as the frequency domain.

The comparison of theory and experiment in the time domain shows good agreement. WIGLE accurately predicted the delay times and the attenuation of the peak of the pulse when a first flight kernel was used to describe the spatial distribution of the fast source. At large distances from the source the time profiles predicted by WIGLE are consistently narrower than the measured pulse shapes. The sensitivity of the one-dimensional model to small changes in the transverse buckling was studied; WIGLE (and experiment) showed a large sensitivity to these changes. These results point to the necessity of obtaining the best possible estimate of the nuclear parameters to be used in space-time kinetics calculations.

The comparison of theory and experiment in the sensitive frequency domain confirms the good agreement found in the time domain. The
comparison of the WIGLE results and the pulse propagation data were put on the same basis by performing identical Fourier transformations and fitting procedures on both. The predicted and measured real part of \( \rho^2 = (a^2 - \xi^2) \) diverge past 100 cps. This is almost a "natural" occurrence in the ultrasensitive \( \rho^2 \) plane in which the opposing discrepancies of \( a^2 \) and \( \xi^2 \) are blown out of proportion. The agreement in the \( \rho \) dispersion law of the system is good and within the experimental accuracy and capabilities of the model.

It is believed that WIGLE did predict accurately the dynamic behavior of the narrow, long core studied. It is also believed that the sensitivity of the one-dimensional calculational scheme to small changes in the transverse buckling should be born in mind when feedback effects and/or two-dimensional effects are to be considered.

The followings topics deserve future investigation:

1. The sensitivity of the dynamic behavior of the system to small changes in the transverse buckling. This can be accomplished by a detailed pulse propagation study at two core heights or at two core widths.

2. Impose a more severe test of the model by conducting measurements with a narrower input pulse.

3. Further interpretation of reactivity measurements in large systems is needed. A correction for the influence that the input pulse width has on the determination of \( k\delta/\lambda \) by the Garelis-Russell method is necessary. The extrapolated area-ratio method should provide a basis for comparison.

4. A comparison should be made between the calculational results of two-group diffusion solved directly in the frequency domain and those obtained by numerical transformation of the WIGLE results.
APPENDIX A

CALCULATIONAL PROCEDURES USED IN THE DETERMINATION OF THE NUCLEAR PARAMETERS AND THE $k_{\text{eff}}$ VALUES
Determination of the Four Group Parameters

A four-group diffusion approach was used to calculate the effective multiplication factor of the six UFSA configurations to be studied as Phase I of the large core dynamics experimental program. The calculation of the parameters was performed by the Nuclear Safety Research Branch of the Phillips Petroleum Company [31].

The four energy groups were divided as follows:

- **Group 1**: $8.21 \times 10^5$ - $10^7$ EV
- **Group 2**: $5.53 \times 10^3$ - $8.21 \times 10^5$ EV
- **Group 3**: $0.532$ - $5.53 \times 10^3$ EV
- **Group 4**: $0$ - $0.532$ EV.

To obtain the four-group constants, the following sequence of calculations was performed for each of the six cases mentioned.

1. Fast group (groups 1-3) constants were calculated by the PHROG [50] computer program. These calculations include resonance integral and Dancoff correction calculations as provided by the RAVEN [51] calculational scheme.

2. Thermal constants were calculated by the TOTEM [52] computer program.

The constants used for the core object of this work are given below. It should be mentioned that the procedure described was also employed to determine the two-group parameters used for the WIGLE calculations.
UFSA R1 CORE PARAMETERS

0.5 M/W ratio
16.35 cm wide reflected core

<table>
<thead>
<tr>
<th>Group</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
</tr>
</thead>
<tbody>
<tr>
<td>D</td>
<td>1.6079</td>
<td>0.94239</td>
<td>0.61401</td>
<td>0.20809</td>
</tr>
<tr>
<td>$\Sigma_a$</td>
<td>0.0035411</td>
<td>0.0023701</td>
<td>0.23109</td>
<td>0.20809</td>
</tr>
<tr>
<td>$\Sigma_r$</td>
<td>0.090927</td>
<td>0.096727</td>
<td>0.084015</td>
<td>-</td>
</tr>
<tr>
<td>$\nu\Sigma_f$</td>
<td>0.0068929</td>
<td>0.0012609</td>
<td>0.015301</td>
<td>0.22662</td>
</tr>
<tr>
<td>$\frac{1}{\nu}$</td>
<td>5.25 $\times 10^{-10}$</td>
<td>2.637 $\times 10^{-9}$</td>
<td>1.843 $\times 10^{-7}$</td>
<td>3.324 $\times 10^{-6}$</td>
</tr>
</tbody>
</table>

Void Coefficient of Reactivity: $-3.14 \times 10^{-3} \Delta k/\%$ void
Temperature Coefficient of Reactivity: $-1.03 \times 10^{-4} \Delta k/\degree C$

Determination of $k_{eff}$

Using the constants tabulated above, eigenvalue calculations were performed with the AIM-6 computer code [53].

AIM-6 is a multigroup, multiregion, one-dimensional diffusion theory code that has been widely used for criticality calculations. The code will handle as many as 18 energy groups, 101 space points and 20 regions. Basically, the code solves the equation

$$ -D^2 \phi^i(r) + \Sigma^T_i \phi^i(r) = \chi^i S(r) + \sum_{j=q}^{i-1} \Sigma_s j \phi^j(r) $$

$1 < i < \text{NOG} < 18$. NOG = number of groups

The symbols follow conventional notation and

$\chi^i$ = the integral of the fission spectrum over the lethargy range represented by group i.
\[ S(r) = \sum_{i=1}^{\text{NOG}} \frac{(\Sigma F)^i \psi_i(r)}{\lambda} \]

where \( \lambda \) is the eigenvalue. The code then sets and solves a set of finite difference equations.
APPENDIX B

DESCRIPTION OF THE COMPUTER PROGRAMS
UNIPUL
MORE
MORWIG
ALXILS
MULTIPLLOT
General

A set of computer programs was coded for the data processing and the analysis of the results of the pulsed measurements. The programs were written for the IBM 360-50 computer, with the exception of MULTIPILOT which was coded for the IBM 1800 computer.

UNIPUL

The FORTRAN IV, IBM 360 computer program UNIPUL is used for a unified processing of the data obtained from conventional pulsing or pulse propagation measurements. A complete description of the program and the input requirements are available in Ref. 54.

The main operations of UNIPUL are:

1. Resolution time correction of the data, using a non-paralyzing counting correction.

2. Statistical determination of the neutron background. If the system under study is multiplicative, the delayed neutron background is calculated and stored in COMMON for the calculation of $\kappa_\beta/\kappa$.

3. Determination of the fundamental decay constant using Peierl's statistical method [18]. The decay constants can be calculated at different waiting times from the peak of the pulse to determine the time convergence. The $\alpha$'s are stored in COMMON for the reactivity determination.

4. The background is subtracted from the data. A test is made to find if any channel has a negative number of counts. If a few channels have negative counts, their value is set to zero. The number of

1. The cooperation of Dr. M. J. Ohanian in the coding of these programs is gratefully acknowledged.
channels is now reduced to an odd number of points terminating where the background has been determined to start.

5. Determination of the ratio $k_{\beta}/l$ following the Garelis-Russel method. Simpson's integration is used for the pulse and the Regula-Falsi iterative method is used to find the root.

6. The data is normalized according to a selected normalization scheme (see Part I, Chapter IV). The normalized data can be punched in cards for further analysis.

7. If a neutron wave type analysis is desired, a numerical Fourier transformation is performed and the amplitude and phase of the pulse for each frequency selected is printed and punched for further processing.

UNIPUL was used to process the pulse data punched in cards from original perforated paper tape (at the IBM 1800 computer of the Department of Nuclear Engineering Sciences, University of Florida); the paper tape is the output of the multichannel analyzer used in the experimentation.

MORE

The FORTRAN IV, IBM 360 computer program MORE performs a conventional numerical Fourier transformation of the pulse neutron data (zeroth moment only) for the analysis of the experiment in the frequency domain. The transformation used equal time steps, a maximum of 1023 time points and Simpson's integration scheme. An odd number of points are required by the program. Thermal neutron data can be entered directly as input or the total and epicaldium neutron flux are required to obtain the thermal flux by subtraction.
MORE punches in cards the amplitude and phase of the zeroth Fourier moment of the pulse for input to the ALXILS program. A complete description of the program and input requirements can be found in Ref. 55.

MORWIG

The MORWIG program is a version of the MORE code written to perform a numerical Fourier transformation on a pulse with unequal time steps. It is intended specifically for the neutron wave type analysis of the space-time data calculated by the WIGLE scheme. See Ref. 56 for a complete description.

ALXILS

The FORTRAN IV, IBM 360 program ALXILS performs a least-squares fit of the amplitudes and phases obtained from the Fourier transformation performed by either the MORE of the MORWIG program. A linear least-squares scheme is used. The values of ALPHA are obtained by fitting the log amplitude vs. distance for each frequency. The values of XI are obtained by a linear fit of the phase angle vs. distance. The channel chopping technique is used to determine the convergence of $\alpha$ and $\xi$ in space. The weighting of the points can be set to $(1/$observed value$)$ or to equal weights. Statistical quantities determining the "goodness of the fit" are calculated for each fit. A complete description can be found in Reg. 57.

MULTIPLIT

The FORTRAN IV, IBM 1800 computer program MULTIPLIT used the output of the UNIPUL and/or the WIGLE codes to plot the time profiles of
the pulses. A CALCOMP plotter is used in connection with the 1800 computer for the actual plotting of the pulse. The program is capable of normalizing to the peak counts of a reference pulse for the superposition of related time profiles.
APPENDIX C

UFSA R1 CLEAN CORE
TIME PROFILES OF THERMAL NEUTRON FLUX
AT NINETEEN CORE POSITIONS
FOR INPUT PULSES OF 0.5 AND 1.0 MSEC
FIG. 01  TIME PROFILE OF THERMAL NEUTRON FLUX  3.0 CM FROM THE SOURCE
TIME PROFILE OF THERMAL NEUTRON FLUX 15.71 CM FROM THE SOURCE
FIG. 03  TIME PROFILE OF THERMAL NEUTRON FLUX  28.43 CM FROM THE SOURCE
FIG. 64  TIME PROFILE OF THERMAL NEUTRON FLUX  41.15 CM FROM THE SOURCE
FIG. 65  TIME PROFILE OF THERMAL NEUTRON FLUX 53.85 CM FROM THE SOURCE
FIG. 06  TIME PROFILE OF THERMAL NEUTRON FLUX  66.58 CM FROM THE SOURCE
FIG. 07 TIME PROFILE OF THERMAL NEUTRON FLUX 79.30 CM FROM THE SOURCE
FIG. C8  TIME PROFILE OF THERMAL NEUTRON FLUX  92.01 CM FROM THE SOURCE

THERM-5-55

PEAK AT 1.480 MSECS
PULSE WIDTH = 0.5 MSECS

- THEORY FWHM = 2.335 MSECS
- EXP. FWHM = 2.520 MSECS
Fig. C9  Time profile of thermal neutron flux 104.73 cm from the source.
FIG. C10    TIME PROFILE OF THERMAL NEUTRON FLUX 117.44 CM FROM THE SOURCE
FIG. 011  TIME PROFILE OF THERMAL NEUTRON FLUX 130-16 CM FROM THE SOURCE
FIG. C12  TIME PROFILE OF THERMAL NEUTRON FLUX 142.88 CM FROM THE SOURCE

THERM-5-63
PEAK AT 2.650 MSEC
PULSE WIDTH = 0.5 MSEC

THEORY FWHM = 3.465 MSEC
EXP. FWHM = 3.720 MSEC
FIG. C13  TIME PROFILE OF THERMAL NEUTRON FLUX 155-59 CM FROM THE SOURCE
FIG. C14 TIME PROFILE OF THERMAL NEUTRON FLUX 162.31 CM FROM THE SOURCE
FIG. C15  TIME PROFILE OF THERMAL NEUTRON FLUX 181.03 CM FROM THE SOURCE
FIG. C16  TIME PROFILE OF THERMAL NEUTRON FLUX 193.74 CM FROM THE SOURCE
Fig. C17  Time profile of thermal neutron flux 206.46 cm from the source.
FIG. 018  TIME PROFILE OF THERMAL NEUTRON FLUX 21.17 CM FROM THE SOURCE
FIG. C19  TIME PROFILE OF THERMAL NEUTRON FLUX 231.98 CM FROM THE SOURCE
FIG. C20  TIME PROFILE OF THERMAL NEUTRON FLUX  3.0 CM FROM THE SOURCE
THERM1-0-13
PEAK AT 0.970 MSEC
PULSE WIDTH = 1.0 MSEC
THEORY FWHM = 1.065 MSEC
EXP. FWHM = 0.960 MSEC

FIG. C21 TIME PROFILE OF THERMAL NEUTRON FLUX 15.71 CM FROM THE SOURCE
FIG. C22  TIME PROFILE OF THERMAL NEUTRON FLUX 28.43 CM FROM THE SOURCE
Figure C23: Time profile of thermal neutron flux 41.15 cm from the source.
FIG. C24  TIME PROFILE OF THERMAL NEUTRON FLUX  53.86 CM FROM THE SOURCE
FIG. C25  TIME PROFILE OF THERMAL NEUTRON FLUX  66.58 CM FROM THE SOURCE

THERM1•0- 41
PEAK AT  1.330 MSECs
PULSE WIDTH = 1.0 MSECs

THEORY FWHM = 1.975 MSECs
EXP. FWHM = 1.920 MSECs
Time profile of thermal neutron flux 79.30 cm from the source.
FIG. C27 TIME PROFILE OF THERMAL NEUTRON FLUX 92.01 CM FROM THE SOURCE
FIG. C28  TIME PROFILE OF THERMAL NEUTRON FLUX 104.73 CM FROM THE SOURCE
FIG. C29  TIME PROFILE OF THERMAL NEUTRON FLUX 117.44 CM FROM THE SOURCE
FIG. C30  TIME PROFILE OF THERMAL NEUTRON FLUX 130.15 CM FROM THE SOURCE
FIG. 631  TIME PROFILE OF THERMAL NEUTRON FLUX 142.08 CM FROM THE SOURCE

FLUX (RELATIVE UNITS)
FIG. C32  TIME PROFILE OF THERMAL NEUTRON FLUX 155.59 CM FROM THE SOURCE
FIG. C33  TIME PROFILE OF THERMAL NEUTRON FLUX 168.31 CM FROM THE SOURCE
FIG. C34  TIME PROFILE OF THERMAL NEUTRON FLUX 181.03 CM FROM THE SOURCE
FIG. C35  TIME PROFILE OF THERMAL NEUTRON FLUX 193.74 CM FROM THE SOURCE
FIG. C36  TIME PROFILE OF THERMAL NEUTRON FLUX 206.46 CM FROM THE SOURCE
FIG. C37  TIME PROFILE OF THERMAL NEUTRON FLUX 219.17 CM FROM THE SOURCE

THERM1.0-125
PEAK AT  4.570 MSEC
PULSE WIDTH = 1.0 MSEC
THEORY FWHM = 4.275 MSEC
EXP. FWHM = 4.530 MSEC
FIG. C38  TIME PROFILE OF THERMAL NEUTRON FLUX 231.88 CM FROM THE SOURCE

FLUX (RELATIVE UNITS)

TIME (MSEC)

THERM 1: 0.132
PEAK AT 4.660 MSECs
PULSE WIDTH = 1.0 MSECs
THEORY FWHM = 4.275 MSECs
EXP. FWHM = 4.560 MSECs
APPENDIX D

UFSA R1 CLEAN CORE
TIME PROFILES OF FAST NEUTRON FLUX
AT SEVERAL CORE POSITIONS
FOR INPUT PULSES OF 0.5 AND 1.0 MSEC
RIFS 13-1
PEAK AT 0.459 MSECS
PULSE WIDTH = 0.5 MSECS
THEORY FWHM = 0.594 MSECS
EXP. FWHM = 0.600 MSECS

FIG. D1  TIME PROFILE OF FAST NEUTRON FLUX  15.71 CM FROM THE SOURCE
FIG. D2  TIME PROFILE OF FAST NEUTRON FLUX  28.43 CM FROM THE SOURCE
FIG. D3  TIME PROFILE OF FAST NEUTRON FLUX  53.86 CM FROM THE SOURCE
TIME PROFILE OF FAST NEUTRON FLUX 66.58 CM FROM THE SOURCE
FIG. D5 TIME PROFILE OF FAST NEUTRON FLUX 79.30 CM FROM THE SOURCE
FIG. D6  TIME PROFILE OF FAST NEUTRON FLUX 193.74 CM FROM THE SOURCE
FIG. 17  TIME PROFILE OF FAST NEUTRON FLUX  15.71 CM FROM THE SOURCE

RIFS 13
PEAK AT  0.970 MSEC
PULSE WIDTH = 1.0 MSEC
THEORY FWHM = 1.050 MSEC
EXP. FWHM = 1.110 MSEC
RFS 27
PEAK AT 0.970 MSECS
PULSE WIDTH = 1.0 MSECS
THEORY FWHM = 1.365 MSECS
EXP. FWHM = 1.380 MSECS

FIG. D8 TIME PROFILE OF FAST NEUTRON FLUX 41.15 CM FROM THE SOURCE
TIME PROFILE OF FAST NEUTRON FLUX 53.96 CM FROM THE SOURCE
FIG. D10  TIME PROFILE OF FAST NEUTRON FLUX  66.58 CM FROM THE SOURCE
FIG. D11 TIME PROFILE OF FAST NEUTRON FLUX 168.31 CM FROM THE SOURCE.
FIG. D12  TIME PROFILE OF FAST NEUTRON FLUX 181.03 CM FROM THE SOURCE
APPENDIX E

UFSA R1 CLEAN CORE
TIME PROFILES OF THERMAL NEUTRON FLUX
AT FOUR POSITIONS IN THE CORE
FOR A 0.1 MSEC INPUT PULSE
FIG. EI  TIME PROFILE OF THERMAL NEUTRON FLUX  SEL. 86 CM FROM THE SOURCE
FIG. E2  TIME PROFILE OF THERMAL NEUTRON FLUX 79.30 CM FROM THE SOURCE
FIG. E3 TIME PROFILE OF THERMAL NEUTRON FLUX 130.16 CM FROM THE SOURCE
FIG. E4    TIME PROFILE OF THERMAL NEUTRON FLUX 142.83 CM FROM THE SOURCE

THERM-1-83

PEAK AT 2.530 MSEC

PULSE WIDTH = 0.1 MSEC

THEORY FWHM = 3.465 MSEC

EXP. FWHM = 3.750 MSEC
APPENDIX F

UFSA R1 CLEAN CORE
TIME PROFILES OF THERMAL NEUTRON FLUX
AT THREE POSITIONS IN THE CORE
FOR A WIDE (10 MSEC) INPUT PULSE
FIG. F1 PROPAGATION OF A PULSE WIDER THAN THE SYSTEM PROPAGATION TIME - EXPERIMENTAL -
APPENDIX G

UFSA R1 CLEAN CORE
SHAPE OF THE PROPAGATING PULSE
AS A FUNCTION OF INPUT PULSE WIDTH

<table>
<thead>
<tr>
<th>Pulse Width (msec)</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.1, 0.5, 1.0, 2.0, 3.0,</td>
</tr>
<tr>
<td>4.0, 5.0, 6.0, 7.0, 8.0,</td>
</tr>
<tr>
<td>9.0, 10.0</td>
</tr>
</tbody>
</table>
FIG. G1 SHAPE OF THE PROPAGATING PULSE AS A FUNCTION OF PULSE WIDTH
- EXPERIMENTAL -
FIG. 62  SHAPE OF THE PROPAGATING PULSE AS A FUNCTION OF PULSE WIDTH  
- EXPERIMENTAL -
LIST OF REFERENCES


46. CORA, an improved version of the IMP computer program. IMP is described in IDO-17199.


50. PHROG is a Phillips-Hanford revision of GAM-1 (G. D. Joanou and J. S. Dudek, GAM-1, GA-1850 (1961)).


52. TOTEM links the TOPIC and TEMPEST codes for calculating thermal constants (G. E. Putnam, TOPIC, IDO-16968 (1964), and R. H. Shudde and J. Dyer, TEMPEST II, TID-18284 (1961)).


Nils J. Diaz was born in Moron, Cuba on April 7, 1938. He was graduated in June, 1955 from LaSalle High School in Havana, Cuba. In July, 1960, he obtained the degree of Professional Mechanical Engineering from the University of Villanova, Havana, Cuba, with honors. He worked as a plant design engineer from January, 1960 to April, 1961 and also as instructor of Machine and Plant Design at the University of Villanova for two semesters after his graduation. He arrived in the United States in October, 1961 and worked as a machine designer until September, 1962. He then entered the Graduate School of the University of Florida and received a Master of Science in Engineering in June, 1964. He held a graduate assistantship and a Fellowship from the Organization of American States until December, 1965. From January, 1966 to date he held a Junior Faculty appointment, as an Engineering Assistant, in the Nuclear Engineering Sciences Department of the University of Florida, while working toward the degree of Doctor of Philosophy.

Nils J. Diaz is married to the former Zena G. Gonzalez and is the father of three children, Nils, Ariadne, and Allene. He is a member of the American Nuclear Society and the Society of the SIGMA XI.
This dissertation was prepared under the direction of the chairman of the candidate's supervisory committee and has been approved by all members of that committee. It was submitted to the Dean of the College of Engineering and to the Graduate Council, and was approved as partial fulfillment of the requirements for the degree of Doctor of Philosophy.

March 1969

[Signatures]

Dean, College of Engineering
Dean, Graduate School

Supervisory Committee:

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