THE EVALUATION OF THE IDAHO STATE UNIVERSITY SUBCRITICAL ASSEMBLY

by

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To the Graduate Faculty:

The members of the Committee appointed to examine the thesis of Prinya Jotikasthira find it satisfactory and recommend that it be accepted.

Major Advisor

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I. INTRODUCTION

In a nuclear reactor, the ratio of number of fissions in any one generation to the number of fissions in the immediately preceding generation is defined as the multiplication factor, K. If the multiplication factor is exactly equal to unity the reactor is said to be critical and a neutron chain reaction within the system will be sustained at a constant rate. If K is less than unity the system is in a subcritical state and the chain reaction cannot be self-sustaining. In order to maintain a self-sustaining chain reaction for a given composition of fuel and other materials, a certain amount of fuel,compatible with the geometry of the arrangement of the fuel-material composition, is needed. The minimum amount of fuel necessary to maintain a self-sustaining chain reaction is called the critical mass, and the corresponding geometry of the arrangement of fuel and other materials is called the critical geometry or critical size.

A subcritical assembly or a subcritical reactor, as it is sometimes called, could therefore, be considered to be a portion of the core of a critical reactor. It usually consists of a geometric arrangement of nuclear fuel, moderator and possibly coolant. The assembly is small enough in volume, or contains such a small amount of fuel that it cannot sustain a chain reaction without the aid of an external neutron source. The external neutron source offsets the loss of neutrons (by leakage) from the sides of the assembly, and thus a constant neutron

II. HISTORICAL BACKGROUND

The history of the subcritical assembly can be traced for three billion years, at which time the oldest known rocks were formed. (1) At such time natural uranium contained six percent of the uranium 235 isotope, compared to 0.728 percent present today. It is quite likely that a water moderated system with six percent U^{235} as 'fuel' existed in nature. However, man-made subcritical assembly began only thirty-four years ago when Halban, Toliot and Kowarski of France and Sizilard, and Fermi of Columbia University, started generating fission neutrons. Haban's neutron source amplification experiment utilized heavy water as moderator and thus laid the foundation for the D_2O systems. Initially Fermi worked with light water and later on, turned to graphite as moderator. The assembly in which Fermi observed a ten percent increase in neutron production over the source density, consisted of 200 Kg. of U_3O_8 encased in fifty-two tinned iron cans. These cans were 5 cm. in diameter and 60 cm. in length. The 60 cm. cylindrical lattice was submerged in a 540 cm^3 cylindrical tank which contained a solution of $MnSO_4$. Fermi made the following statement in a paper describing the experiments with the above assembly.

> "From this result we may conclude that chain reaction could be maintained in a system in which neutrons are slowed down without much absorption until they reach thermal energies and are then mostly absorped by uranium rather than by another element. It remains an open question, however, whether this holds for a system in which Hydrogen is used for slowing down the neutrons."(2)

Shortly after, an "Intermediate Pile" was assembled at Columbia University. In 1954, New York University started to construct their subcritical assembly named "Pickel Barrel." The original name surely must have come from the twenty-five dollar wooden barrel. An Italian olive importer sold the University the assembly tank and coopers were hired to assemble it. It seems that the barrel had to be taken apart because it was too large for their building entrance.

From this date on Universities around the country began building their own subcritical assemblies. Most of them utilized natural uranium as fuel. Only two institutions constructed enriched uranium subcritical assemblies. The University of Texas designed a homogeneous enriched assembly. Rutgers, The State University of New Jersey, designed a heterogeneous enriched system which "traveled west and settled" at Idaho State University. At Idaho State University, this subcritical assembly will be referred to as the Idaho State University subcritical assembly.

III. DESCRIPTION OF COMPONENTS OF THE SUBCRITICAL ASSEMBLY

The Idaho State University subcritical assembly was originally designed and assembled by Dr. F. J. Jankowsky and his associates at Rutgers in the year 1961. This assembly, unlike most others, consists of a water moderated and reflected enriched uranium system. The enrichment makes it possible to have less fuel mass. This assembly has been designed to accommodate a variety of core configurations with the same type of fuel plates. The subcritical assembly was reassembled at Idaho State University essentially as it existed at Rutgers. A water handling system, a shutdown system, a working platform, and a core lifting device were added to the original components. Also, in accordance with the United States Atomic Commission regulations, a gamma criticality alarm was installed for safety.

The Idaho State University subcritical assembly consists of an aluminium core tank(placed on a graphite thermal column), three sets of fuel spacing assemblies, core support devices, and one hundred and fifty fuel plates.

The Core Tank

A cylindrical aluminium tank is used to house the core and the moderator. This tank is large enough to accommodate additional water around the core to act as a reflector. The tank wall is made of a 1/4 in. aluminium sheet, rolled and welded. The bottom of the tank and the removable lid for the top of the tank are fabricated from a 3/8 in.

aluminium sheet. The tank is 36.0 in. in diameter and 39.0 in. high Other dimensions of the core tank are shown in Figure 1. The aluminium top can be locked to the core tank and thus the core tank can be used to store the fuel plates.

Thermal Column

The Thermal Column is used primarily to house the neutron sources and supply the core with neutrons of thermal energy. The thermal column is formed by stacking eight criss-cross layers of graphite blocks as shown in Figure 1. Each layer consists of twelve blocks arranged side by side. Each graphite block is 4 in. square and 48 in. long. The total weight of the thermal column, comprised of ninety-six blocks, is approximately 4500 lbs. One of the blocks, labeled (y8, x1), was accidentally broken into two pieces prior to arrival at Idaho State University. The block labeled (y2, x6) is in two sections made out of two identical pieces each 24 in. long. Eleven other blocks are specially made to accommodate neutrons sources, or 1/4 in. fission counter and enriched U^{235} foils. Figures 2, 3 and 4, show the original positions for these special blocks. In Figure 2, the block (y7, x6) is shown with two enriched uranium foils. Next to it is a blank, a block without special machining. Directly below, a block (y5, x6) is shown with the fission counter in place. Figures 3 and 4 show eight other special blocks. Blocks (y6, x6), (y6, x7) and(y4, x5) are used to house cylindrical neutron sources of 1.02 in. diameter and 1.45 in. high. Blocks (y8, x6) and (y4, x6) are machined to hold foils.

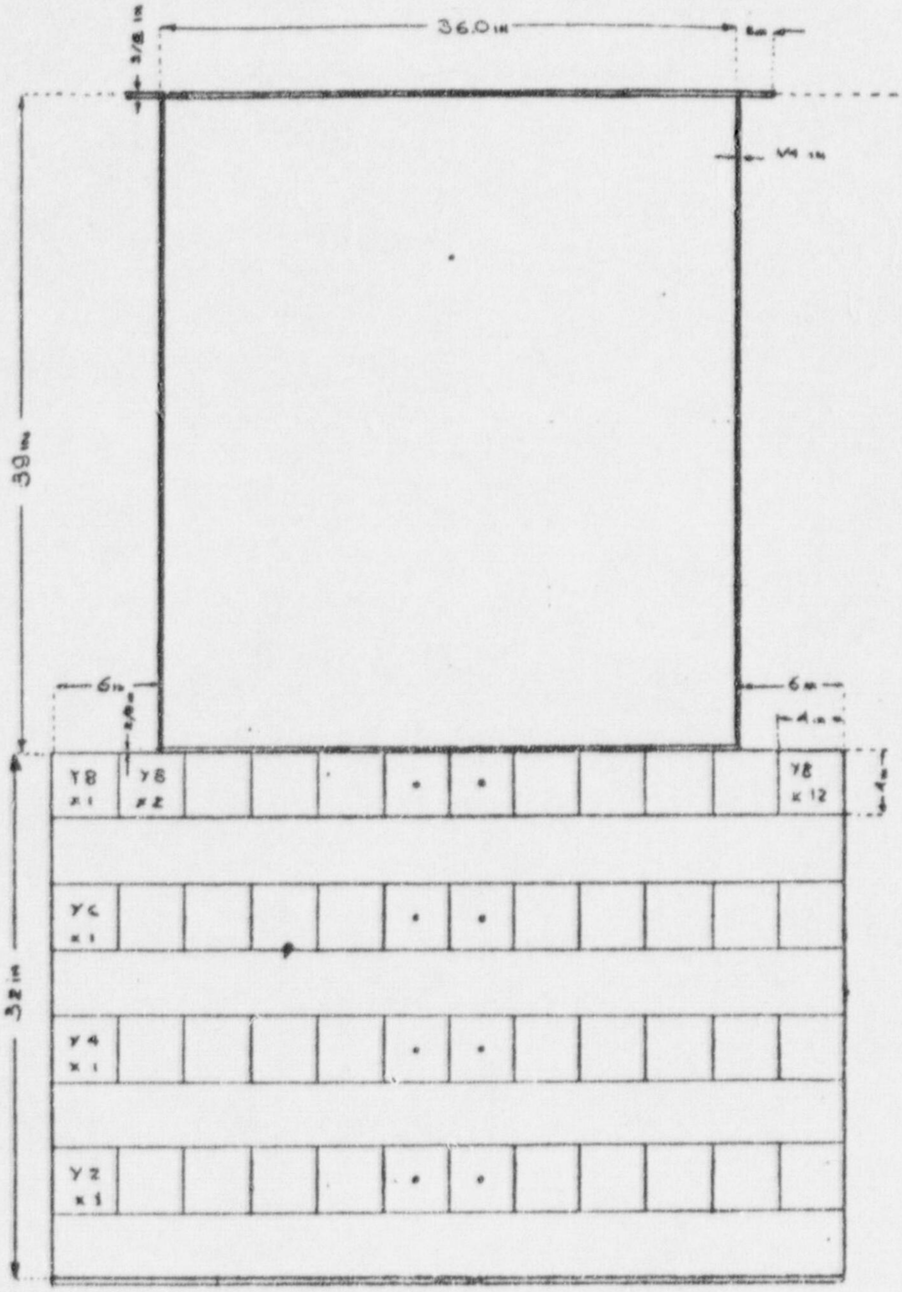

Figure 1: Core tank with the thermal column

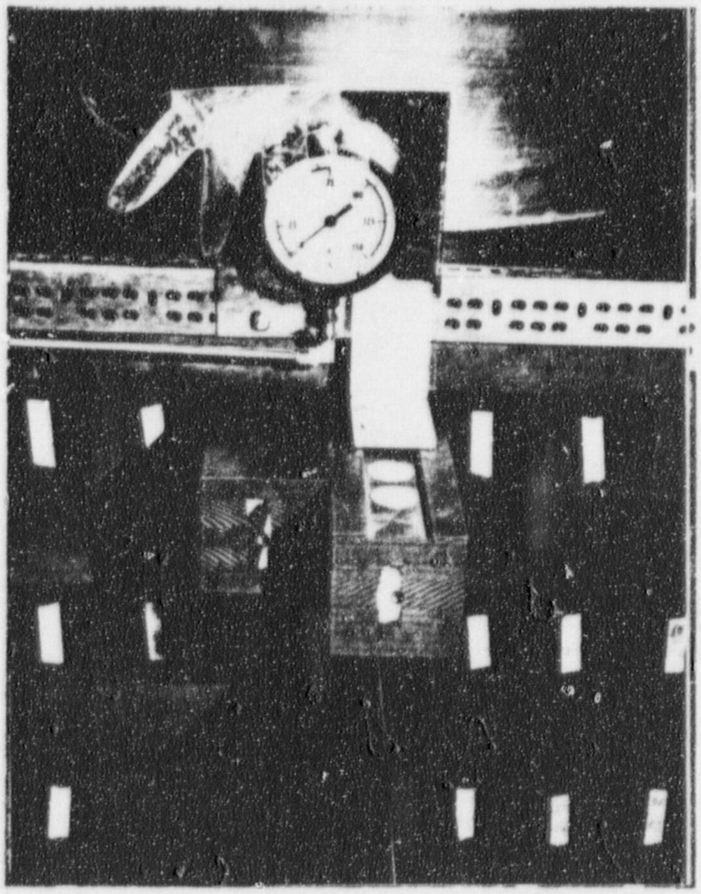

Figure 2: East view of the thermal column.

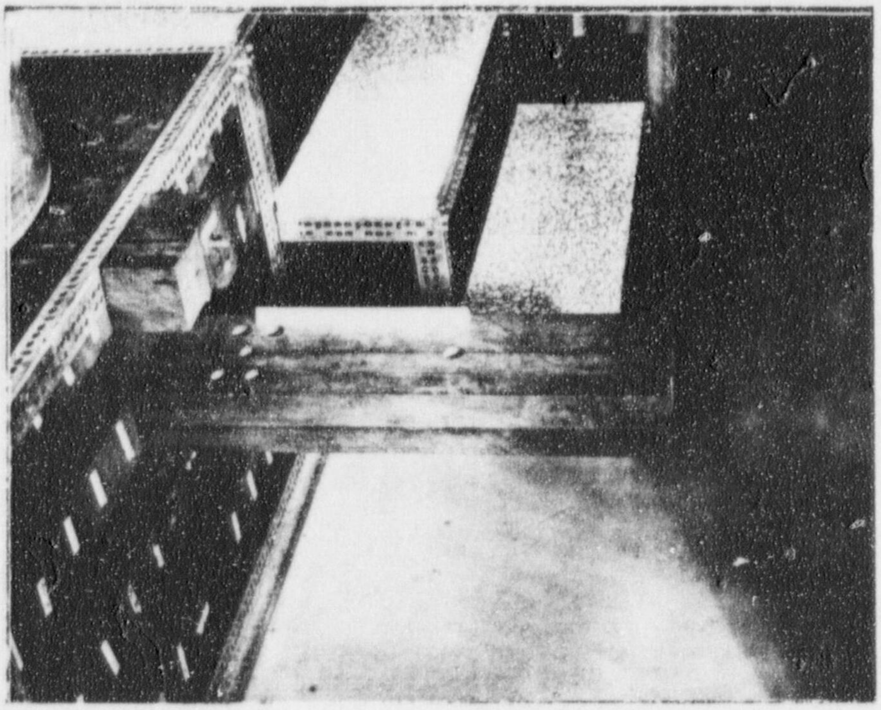

Figure 3: North side of the thermal column.

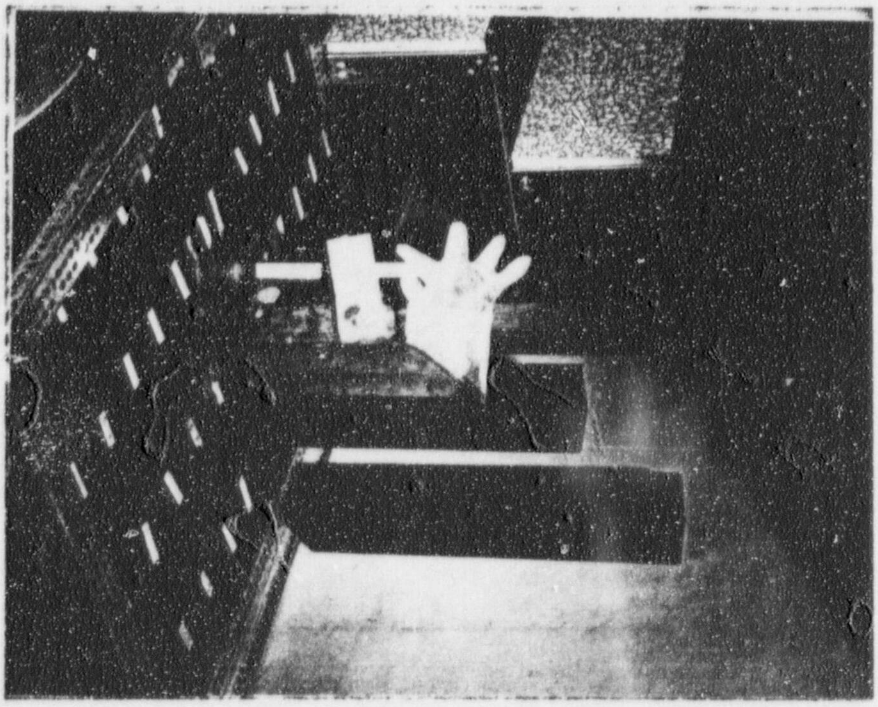

Figure 4: North side of the thermal column.

Each of these eleven blocks has a 3/8 - 16 tapped hole to facilitate handling. Since the positions of any two blocks can be interchanged, a versatile source and measuring system is available.

Fuel Plates

A total of 150 fuel plates are provided with this assembly. Each plate is .08 in. thick, 3.0 in. wide, and 26.0 in. long. The plates are constructed with a fuel bearing region of 0.04 in. thick, 2.75 in. wide and 24.0 in. long, clad with aluminium. A typical plate is shown in Figure 5.

The fuel region consists of a uranium-aluminium mixture with the uranium enriched to 20% U^{235} by weight. The total amount of uranium used is 7614.79 gm. with 1510.27 gm. being U^{235}. It is assumed that all the plates are identical and that the uranium is distributed equally among them.

Fuel Spacing Assembly

The fuel plates can be arranged in three different lattice configurations using the appropriate spacing grids. The fuel plates may be used singly to form a quasi-homogeneous lattice, or in groups of two or three to obtain a more heterogeneous lattice. Each lattice requires a set of grids which is bolted to a common frame to complete the fuel spacing assembly. The three fuel spacing assemblies are shown in Figures 6, 7 and 8; the dimensions of the grids are shown in Figures 9 to 14; Figures 15 and 16 give dimensions of the frame. A detailed description of each type of core is discussed in the next chapter.

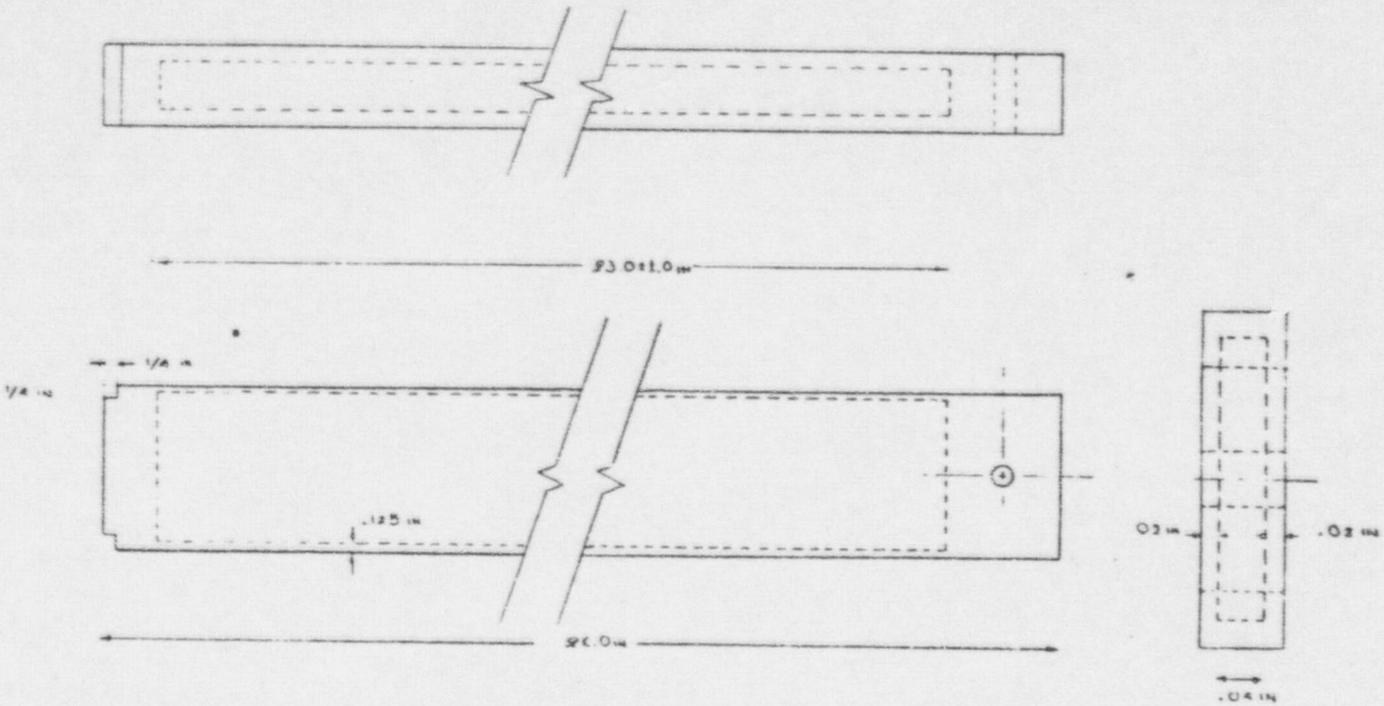

Figure 5: The subcritical assembly fuel plate fabricated by M&C Nuclear Co.

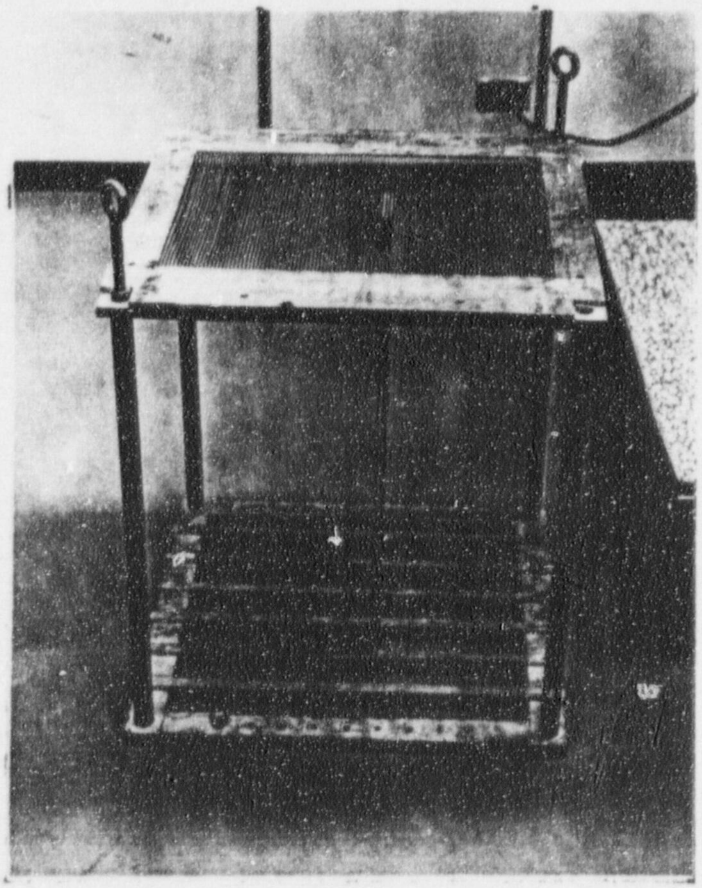

Figure 6: The single plate spacing assembly.

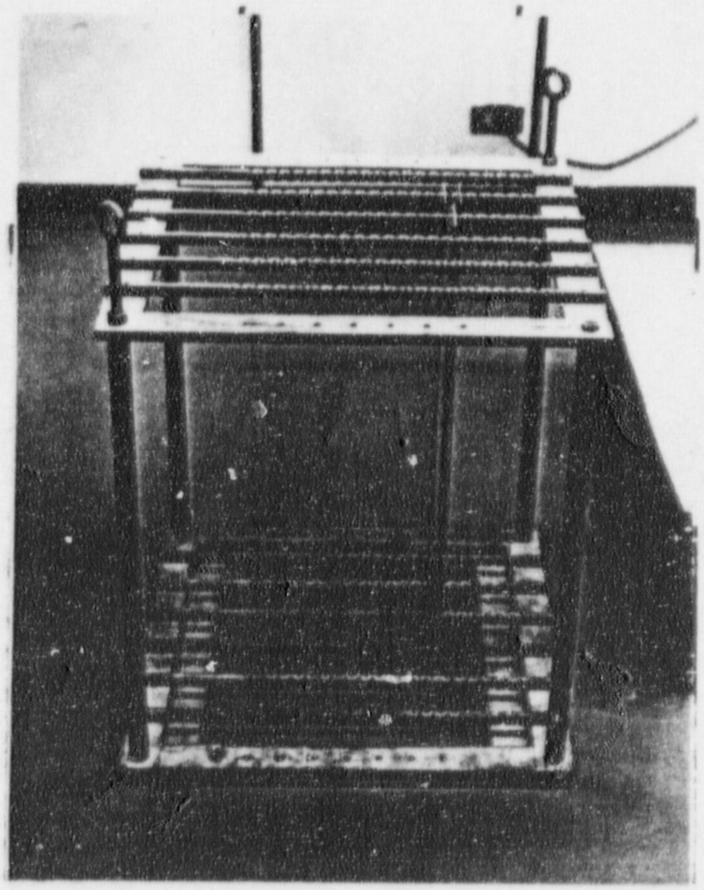

Figure 7: The double plate spacing assembly.

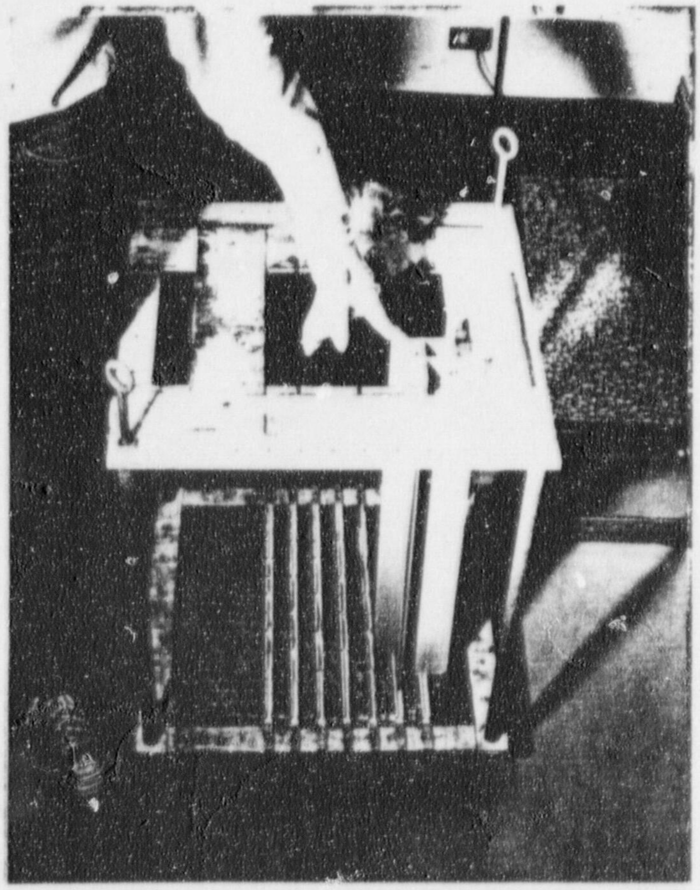

Figure 8: The triple plate spacing assembly.

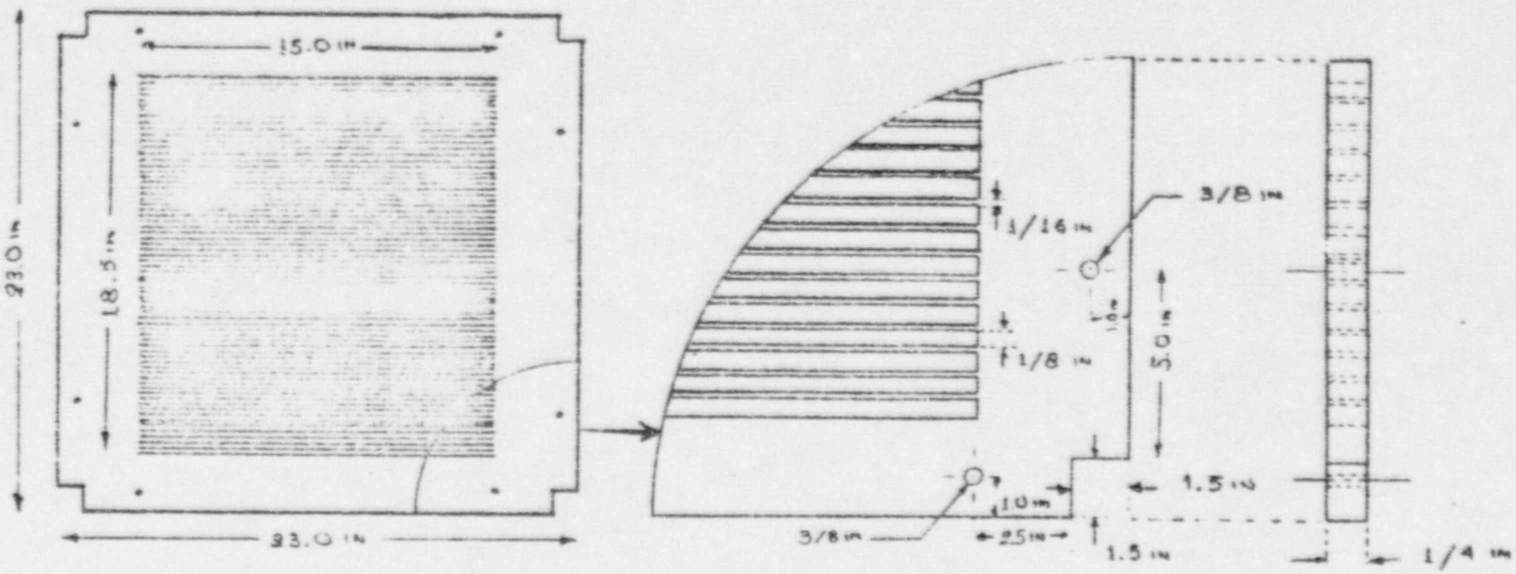

Figure 9: The top single plate spacing grid.

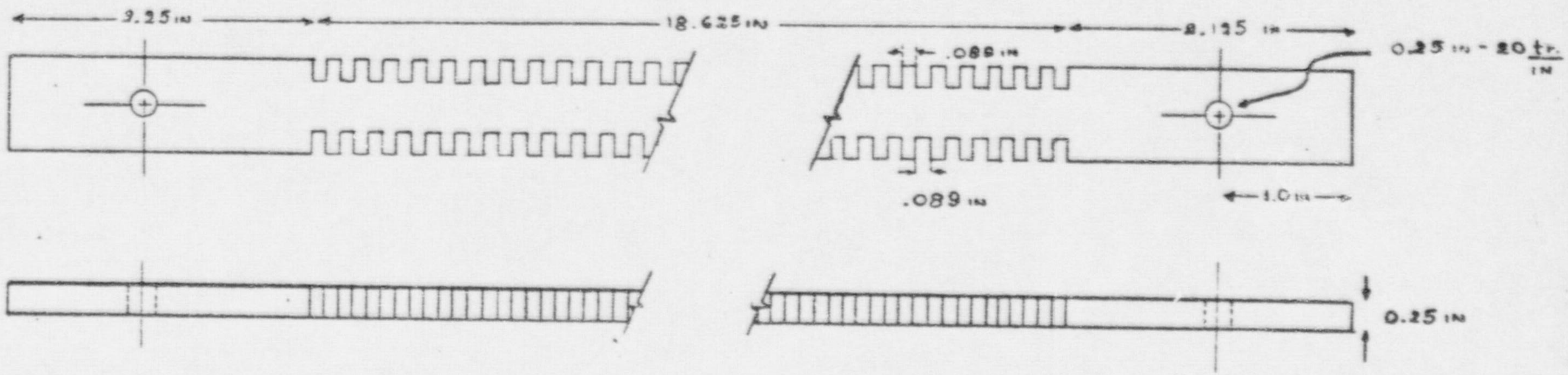

Figure 10: The bottom single plate grid. There are a total of ten identical pieces similar to the one shown above. A maximum of six pieces can be bolted to the spacing grid frame to complete the bottom portion of the fuel spacing assembly.

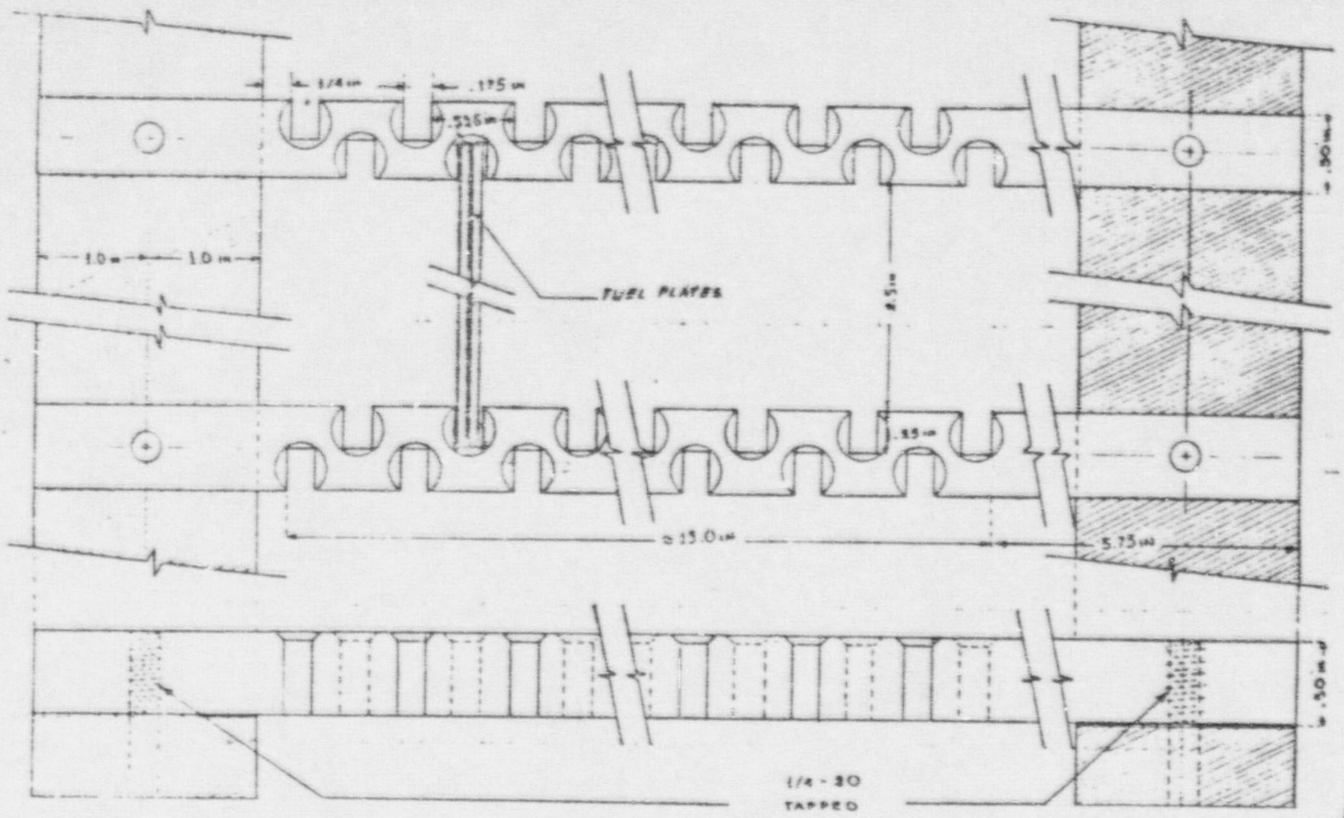

Figure 11: The top spacing grid for the dual plate assembly.

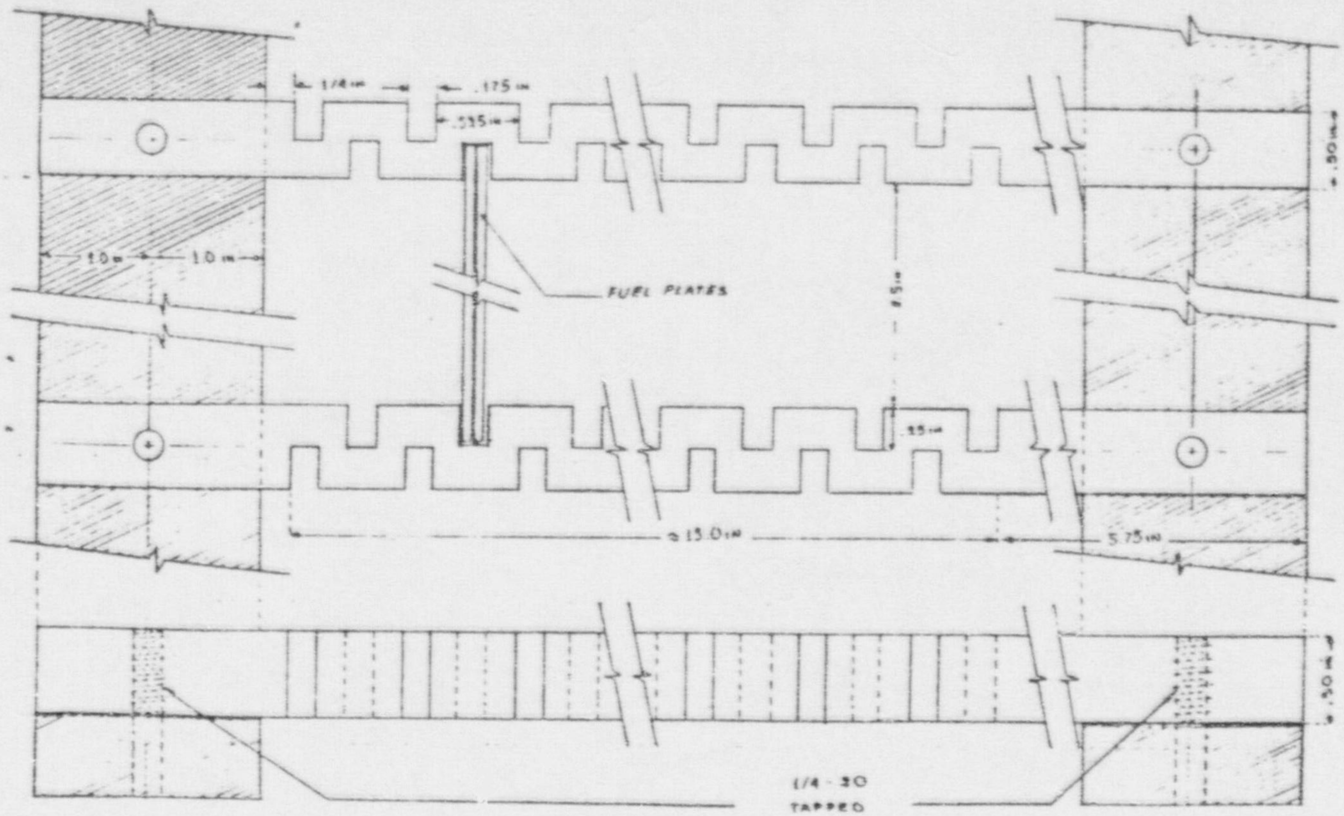

Figure 12: The bottom dual plate spacing grid.

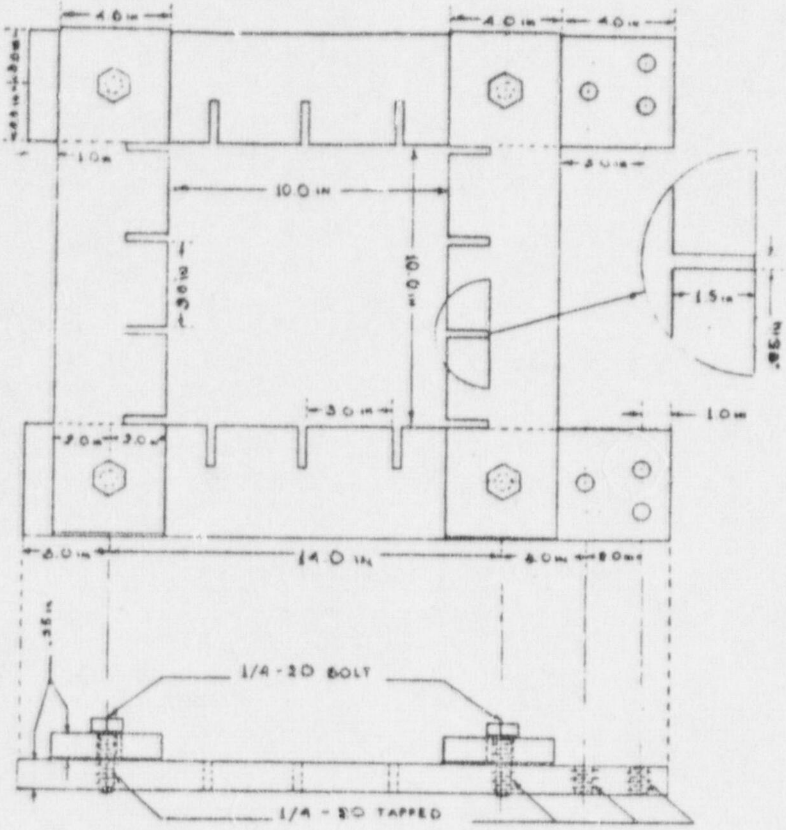

Figure 13: The top triple plate spacing grid.

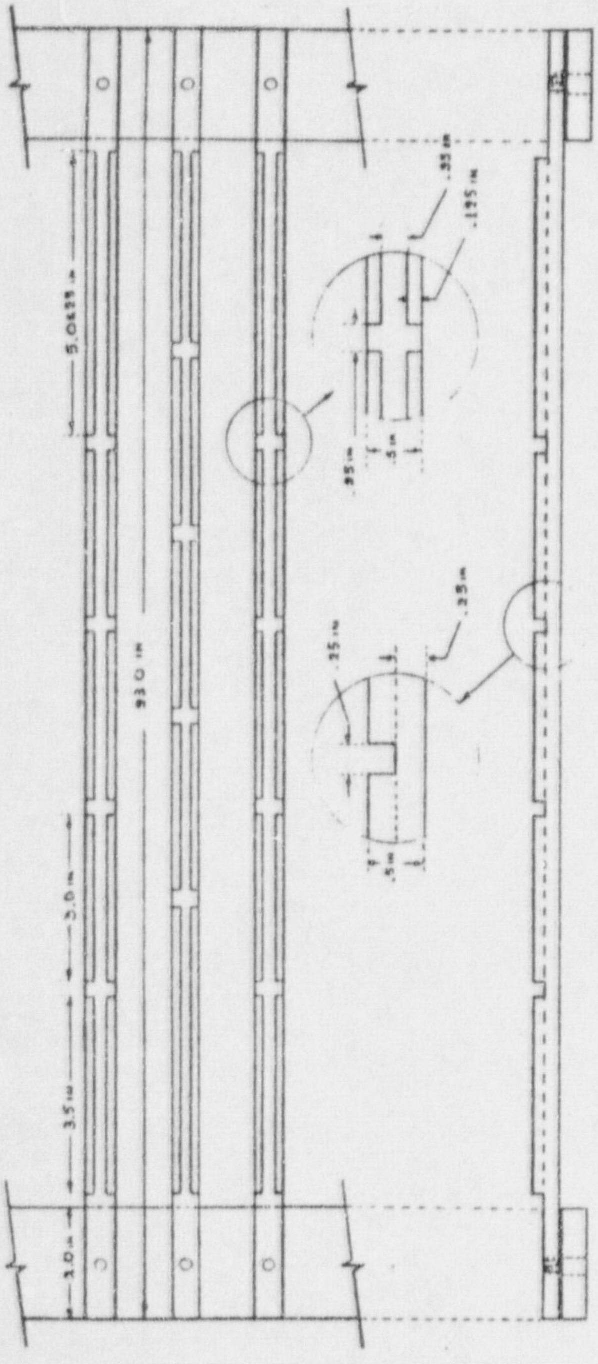

Figure 14: The bottom triple plate spacing grid.

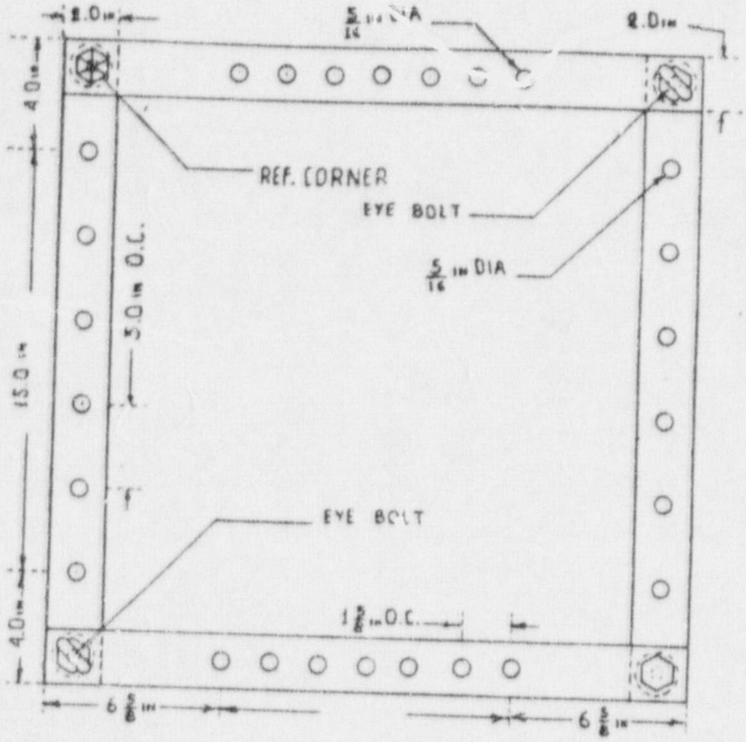

Figure 15: The top view of the fuel spacing frame. The 1-5/8 in. on center holes are used to assemble the T-1 core, while the 3.0 in. OC holes are used to assemble the other five cores.

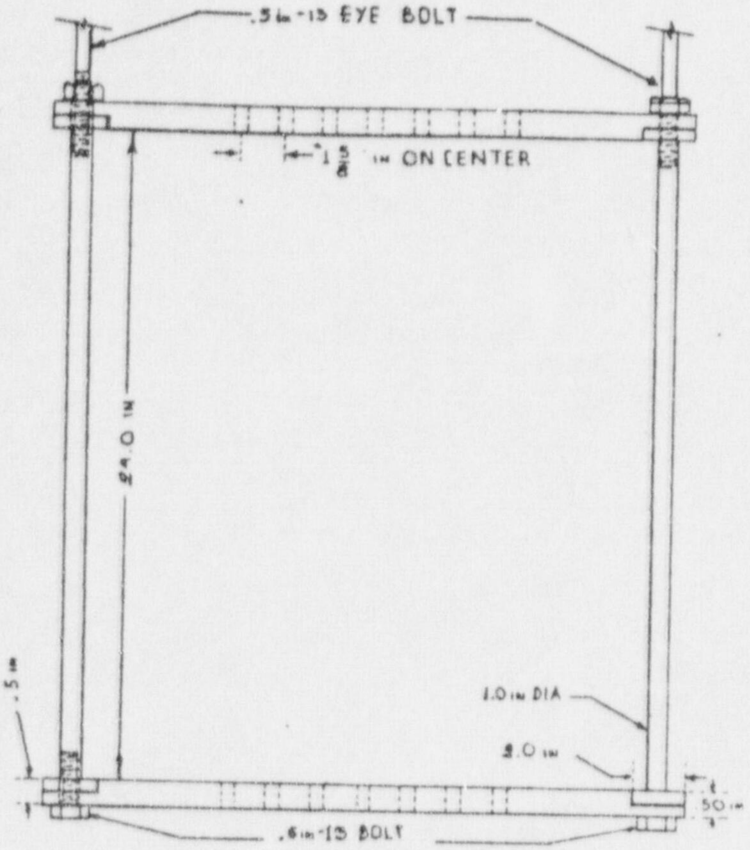

Figure 16: Side view of the spacing assembly frame.

Core Support Devices

A honeycomb structure shown in Figure 17, is used to support the core at the bottom of the tank. To secure the core in its proper place relative to the sides of the core tank, brackets are provided which are bolted to the rim of the tank. This arrangement is shown in Figure 18. Figure 19 shows the picture of both devices in place.

Water Handling System

Distilled water is used in the subcritical assembly as a moderator and reflector. The water is stored in three inter-connected polyethylene drums of 55 gallon capacity each. At the beginning of each experiment the water is pumped from the drums into the core tank. At the end of the experiment the water is drained back into the storage drums. Schedule 40 polyvinylchloride pipes and fittings are used. Half inch pipe is used to feed water into the assembly tank and three quarter inch pipe is used to return the water to the storage drums The pump chosen employs a magnetic drive mechanism with a maximum flow capacity of 940 gallon per hour. Ideally, a pump with the above flow rate will fill the core tank within thirteen minutes. However, because of pipes, fittings, valves and the 5 ft. head in the system the time required to fill the tank has almost doubled.

The present system requires twenty-five minutes to fill the core tank with 33 in. of water. A"normally open" solenoid valve is installed to control the return flow of water to the storage drums. "Normally open" implies that the return flow is cut off only when the coil is

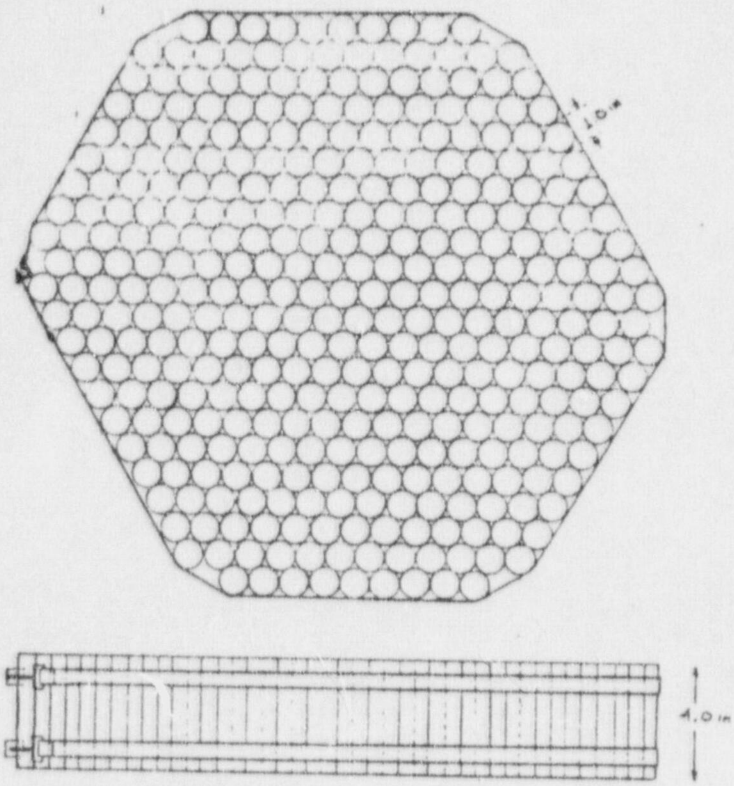

Figure 17: Bottom core support structure.

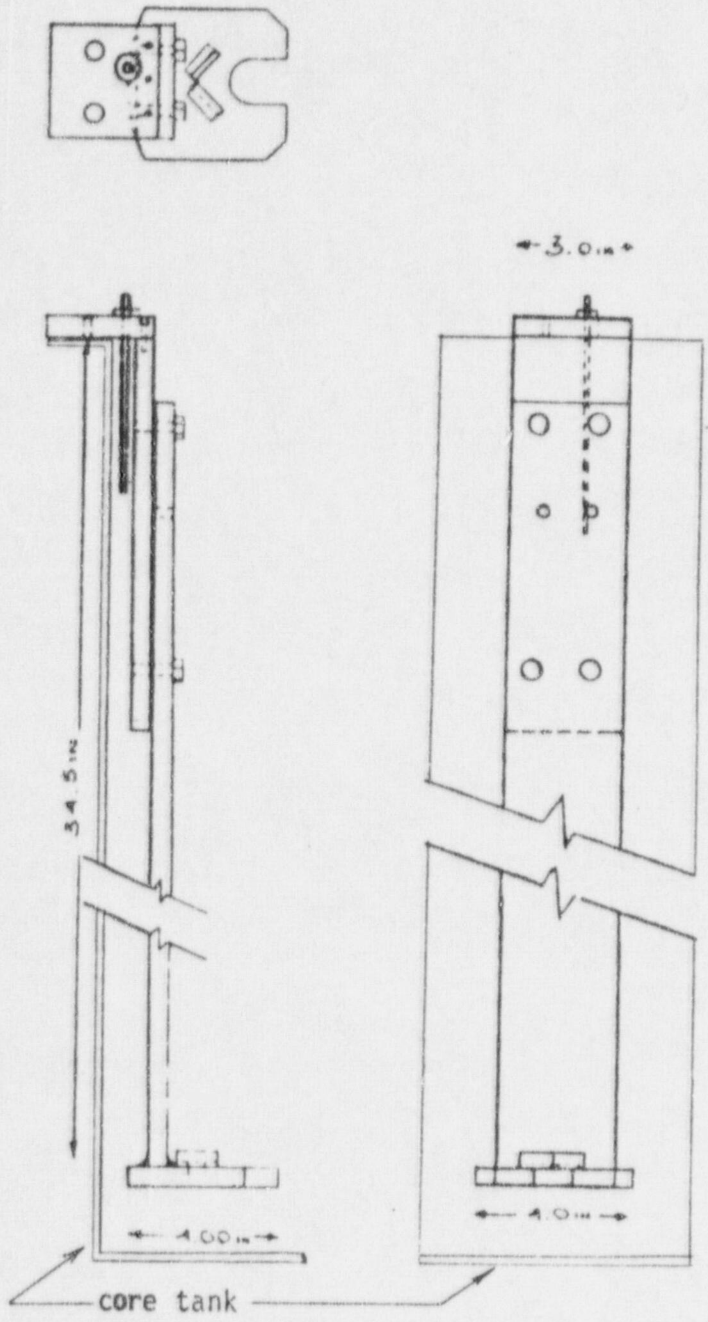

Figure 18: The core side support brackets.

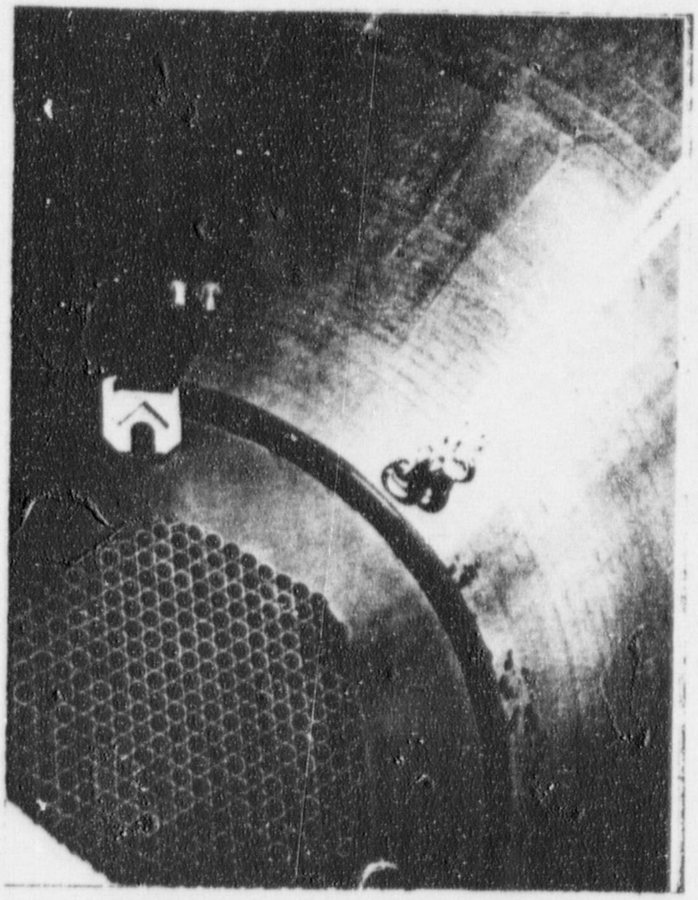

Figure 19: Core support devices in the core tank.

energized. Under operating conditions, the coil is energized and the valve is closed. In case of power failure the solenoid coil is de-energized and the valve is automatically opened to ensure the safety of the system. Two tee junctions are provided so that a demineralizer can readily be connected in parallel with the pump if necessary. The water level in the core tank is indicated by a pressure gauge calibrated in inches of water. This gauge is located at the edge of the graphite pedestal. The operating level of water in the core tank has been set at 33 in.; additional water will automatically flow back into the storage container through the overflow line. A complete schematic of the water handling system is shown in Figure 20.

The power to the solenoid valve and pump is supplied through normally open contacts of the double pole double throw relay. The normally open contacts of this relay are closed by the 18 volt-AC signal from the criticality alarm. During alarm condition the 18 VAC signal is turned off by the alarm, thereby cutting power to both solenoid valve and pump. A second relay is used to insure that power does not return to the pump and solenoid upon power return following a power failure. This relay is wired as an electrically latched switch and must be reset following any power interruptions. A mechanical switch is provided so that the pump may be turned off when the core tank is filled. The complete electrical wiring diagram for the system is shown in Figure 21.

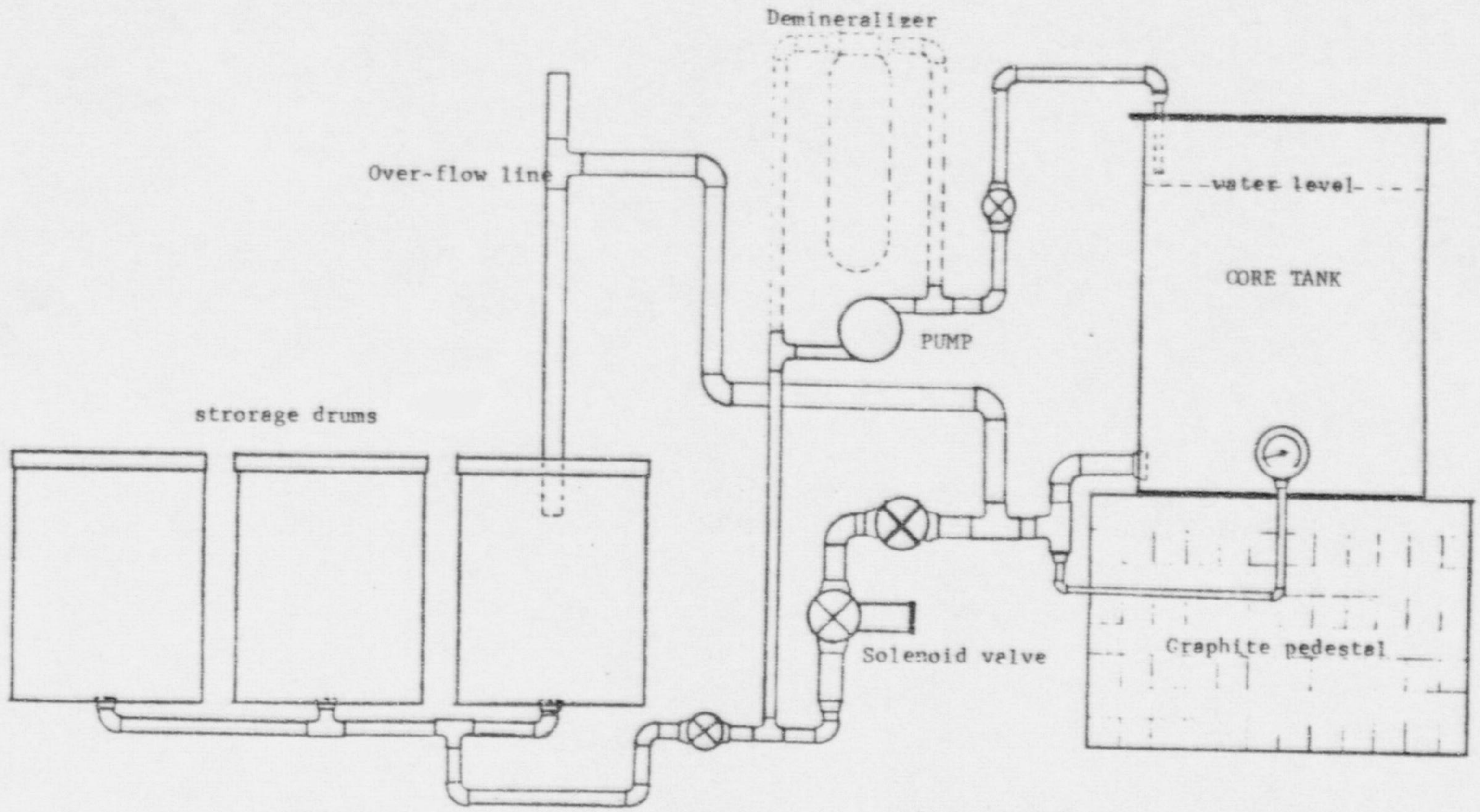

Figure 20: Schematic of the water handling system.

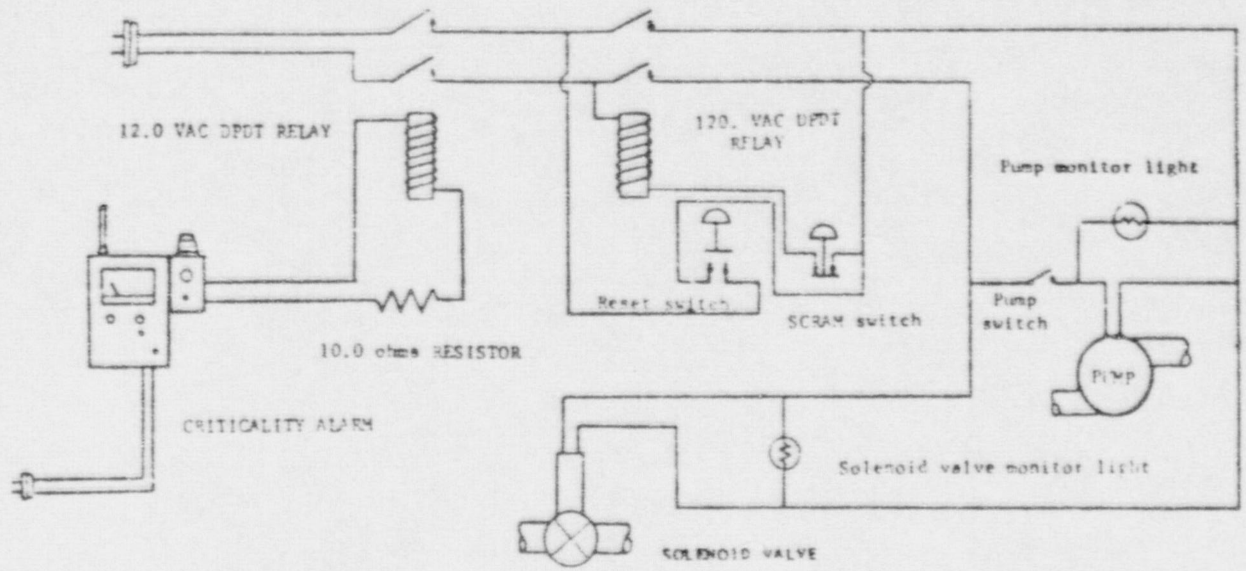

Figure 21: The wiring diagram of the water handling system.

Criticality Alarm

The United States Atomic Energy Commission requires that each licensee who is authorized to possess in excess of 500 gm. of contained U^{235} and/or 300 gm. of plutonium and/or 300 gm. of U^{233}, provide a criticality alarm in the area in which the special nuclear material is handled, used or stored. (3)

The criticality alarm installed in the subcritical assembly room is manufactured by Ludlum Measurements Inc. This instrument is called the model 300 area monitor. The model 300 area monitor utilize 115V 60 Hz single phase electrical line as power source. Stand-by batteries are incorporated within the unit to provide power in case of line power failure.

The gamma field strength is displayed on a three decade logarithmic scale meter covering the range of 0.02 MR/hr to 20 MR/hr. This unit is mounted on the wall approximately six feet away from the surface of the assembly tank and approximately 15 feet away from the fuel storage container.

Emergency Shut Down System

During normal operation of the subcritical assembly, with one curie Pu-Be neutron source placed in the middle of the most reactive core, a combined dose rate of 3.0 MR/hr from source neutrons, fast neutrons, thermal neutrons, prompt and delayed gamma is expected. Calculations are shown in Appendix A. However, in the event of a radiation level reaching 20 MR/hr the solenoid valve opens automatically to drain water from the core tank back into the water storage container.

The automatic draining is initiated by the high limit alarm signal from the model 300 area monitor, as explained previously, in the water handling system section. Draining of the water will reduce multiplication of the core to such an extent that a continued criticality would not be possible.

IV. DESCRIPTION OF THE SUBCRITICAL ASSEMBLY CORES

Six different core configurations can be formed with the fuel-spacer assemblies presently available. Other core configurations could also be realized if new fuel spacing grids are fabricated. All the six cores are of rectangular parallelpiped geometry with different base dimensions. The height of the core, however, is fixed at twenty-four inches. Using the single plate spacing assembly three different lattices can be obtained which, in the present work, are referred to as the S-1 core, the S-2 core and S-3 core. With the double fuel spacing assembly two fuel plates together are used to act as one fuel element. In this manner, the thickness of the fuel and the moderator in each differential slab is increased, while the fuel density remains similar to the corresponding size "S" type core. In other words, heterogeneity of the core has been increased. Two core sizes can be formed with the double plate grids, which are referred to as the D-1 core and the D-2 core. To assemble a more heterogeneous core, three fuel plates are used as one fuel element in the triple plate spacing assembly. This core will be referred to as the T-1 core.

S-1 Core

To form the S-1 core the fuel plates are loaded singly in every slot of the single plate spacing assembly. The resulting fuel span is .175 inches. To obtain a core size of 9 x 9 x 24 in.^{3} a loading of three plates (in a row) by fifty rows is used. The effective multiplication factor of the S-1 core is calculated to be 0.73.

<u>S-2 Core</u>

The S-2 Core utilizes every other slot of the single plate spacing assembly. The fuel span in this configuration is twice that of the S-1 core. The fuel plates are arranged four plates in a row by thirty-seven rows. Thus, 148 fuel plates are used in this S-2 core resulting in a core size of 12 in. by 13 in. by 24 in. The calculated value of K-eff for this core is 0.865.

<u>S-3 Core</u>

To increase the amount of moderator in the core the fuel plates are placed in every third slot in the single plate spacing assembly. All the 150 fuel plates are utilized in a five plate per row by thirty rows configuration. The core obtained in this manner is 15 in. by 16 in. and the calculated K-eff is 0.825.

<u>D-1 Core</u>

The dual fuel grids are cut in such a manner that two fuel plates are held together as one in each slot. The fuel span of the plate is 0.65 in. The core formed is 12 in. wide by 12.75 in. thick by 24 in. long and uses a total of 144 fuel plates in a four plate per row by 18 row thick lattice. The K-eff of this core is calculated to be 0.870.

<u>D-2 Core</u>

The D-2 core is the largest core which can be assembled at present in the Idaho State University subcritical assembly. The size of the D-2 core is 15 in. by 19.5 in. by 24 in. with a fuel span of 1.3 in. All 150 fuel plates are used. This core yields a calculated K-eff of 0.78.

T-1 Core

Only one core can be formed with the triple plate grid. One hundred and forty seven fuel plates are used in sets of three to form a core of 12.5 in. by 12.5 in. by 24 in. This core is the most reactive core of the subcritical assembly with a calculated K-eff of 0.87.

Further specifications of the cores such as the width of the moderator channel are given in Table I together with the dimensions in each core, the fuel span and the calculated and measured K-eff.

TABLE I. Specification of Idaho State University Subcritical Assembly Core

CORE	Number of fuel plates used	Core size (inches)	Center to center fuel span (inches)	Width of Moderator Channel (inches)	Measured K-eff	Calculated K-eff
S-1	150	9.0 x 9.0	.178	0.098	0.8061	0.753
S-2	148	12. x 13.	0.356	0.276	0.875	0.865
S-3	150	15 x 16	.534	.454	---	0.825
D-1	144	12 x 12.75	.65	.49	---	0.870
D-2	150	15 x 19.5	1.3	1.14	---	0.730
T-1	147	12.5 x 12.5	1.62	1.3	0.825	0.873

V. MATHEMATICAL MODEL

Although the subcritical assembly was designed and operated for a brief period of time at Rutgers, The State University, there has been very little information published about the assembly. The only available information being an article by Jankowski (4) and the special Nuclear Materials License Application submitted by Rutgers (5). Both the articles were directed more towards the economy and safety aspects of the assembly than systematic and accurate analysis of the criticality worth of the different cores. This investigation is directed toward a detailed and accurate reactor analysis of the core by means of a computor code and experimentally verifying the results obtained from the computer calculations. The computer code used in this study is called DISNEL. The acronym DISNEL is derived from the name of the code: "Diffusion Iterative Solution for Nineteen Energy Levels." The language of the computer program is in FORTRAN-IV and is designed for use on an IBM 360-75 computer system. DISNEL is an operating code, in use at present at the National Reactor Testing Station, and is written by Kunze, et. al. (6) of Aerojet Nuclear Corporation (formerly Idaho Nuclear Corporation).

The DISNEL code is a "one-pass," one-dimensional diffusion code. The nuclear cross sections required for the calculations are provided in the code itself. A fixed 19 group energy structure, as shown in Table II is used in the code. The group structure in the resonance region for computing resonance self-shielding corrections is essentially

TABLE II. DISNEL 19 groups energies.

Group No.	Lethargy Width	Energy in (ev)	
		Upper	Lower
1	0.25	1.000×10^{7}	2.788×10^{6}
2	0.50	7.788×10^{6}	4.724×10^{6}
3	0.50	4.724×10^{6}	2.865×10^{6}
4	0.50	2.865×10^{6}	1.738×10^{6}
5	0.50	1.738×10^{6}	1.054×10^{6}
6	0.50	1.054×10^{6}	6.393×10^{5}
7	0.50	6.393×10^{5}	3.877×10^{5}
8	0.50	3.877×10^{5}	2.352×10^{5}
9	1.25	2.352×10^{5}	6.738×10^{4}
10	2.00	6.738×10^{4}	9.119×10^{3}
11	2.00	9.119×10^{3}	1.234×10^{3}
12	2.00	1.234×10^{3}	1.67×10^{2}
13	2.00	1.670×10^{2}	2.260×10^{1}
14	1.75	2.260×10^{1}	3.928
15	1.25	3.928	1.125
16	1.00	1.125	4.14×10^{-1}
17	1.00	4.140×10^{-1}	1.523×10^{-1}
18	1.03	1.523×10^{-1}	5.437×10^{-2}
19	Maxwell Boltzman Thermal Average.		

identical to that given in ANL 5800 (7). The element cross-sections are from a Nuclear Data Tape compiled at General Electric, NMPO, Cincinnati. DISNEL is suitable for use with four different core geometries, viz., infinite slab, cylindrical in radial direction, spherical and cylindrical with radial and axial calculations cross-matched on buckling, so that, in effect, it is a two dimensional cylindrical geometry. The machine requirements for this code are 35K words of usable memory with some of the variables in the diffusion calculation being double precision system words. The limitations of this code are that the mesh points be limited to 120 for any one dimensional calculation and that the number of different elements, compositions and regions be limited to 2C each.

DISNEL Input

As mentioned previously, all the six cores of the subcritical assembly are of parallelpiped type. Unfortunately, this code is not amenable to calculations using this geometry. Therefore, a choice must be made among the four kinds of options of geometry available. If the infinite slab geometry is used for the calculations, the width of the core will be the only variable dimension, and leakage from the finite thickness as well as the length, would have to be represented by one buckling term. As a result, the reflector savings in two directions of the core will be no more than an approximation. A more accurate calculation would be a radial-axial cross-matched calculation. If the cylindrical geometry option is used for the calculations, then the actual shape of the system which is parallelpiped, will have to be represented by an equivalent cylindrical geometry. A cylindrical core is always more reactive than a parallelpiped core of the same height, volume and composition.

Therefore, the cylindrical core geometry used for the calculations would result in a larger value for K-eff than the actual rectangular parallelpiped core. A decision was made to use the cylindrical geometry with a radial-axial cross-matched option. After one complete cross-match calculation was performed, a radial-only calculation was performed on the same core with a fixed axial buckling term. A value of 2.0×10^{-3} cm^{-2} was used as the axial buckling term in the subsequent calculations. This number was suggested by one of the authors of the code (8). The K-eff obtained from this radial-only calculation was only 0.09% less than the one obtained from the radial-axial cross-matched calculation. Therefore, the cylindrical radial direction calculation with an axial buckling of 2.0×10^{-3} cm^{-2}, was finally used in calculation of other cores.

In order to use cylindrical approximation with plate-type fuel it was necessary to assume a homogeneous core and then apply a self-shielding correction factor in the calculation. The self-shielding correction factor used in this calculation is outlined in the DISNEL User's Guide (9). The formula is

$$\sigma_p = \frac{f\sigma^1}{1.27} + \frac{S}{4NV} \frac{1-C}{1+0.1C}$$

where

σ^1 = Potential scattering cross-section per absorber atom.

f = An empirical correction factor.

s = Surface of the fuel element

V = Volume of the fuel element

N = Atom density of the absorber

C = The Dancoff correction factor

$$\sigma^1 = \frac{4\pi \ (1.25 \times 10^{-3} \ (A)^{1/3} \ cm)^2}{10^{-24} cm^2 / \ BARN} \qquad \frac{BARN}{ATOM}$$

f = .700 for Oxygen

$C = 2E_3 \ (\Sigma_s d)$

The potential scattering cross-section is the surface area of the absorber nucleus. The empirical correction factor "f" was given as .700(10) for oxygen bearing moderators. The Dancoff correction factors differ for each fuel casing geometry. The formula presented above is the Dancoff correction factor for a slab type fuel element (11), Where "d" is the moderator channel width, Σ_s is the macroscopic scattering cross-section of the moderator, and E_3 (x) is the generalized exponential integral function whose values are tabulated in reference (12).

The self-shielding factors for U^{238} and U^{235} can also be found in reference (13).

For each core of the subcritical assembly, the K-eff of five core sizes was computed by the code in an attempt to generate enough data for a simulation of approach to criticality experiment. For example, the

S-1 has the dimensions of 9 in. x 9 in. x 24 in. The radius R_e of an equivalent cylindrical core is given by:

$$R_e = \frac{(9.0\ \text{in.} \times 9.0\ \text{in.})^{1/2}}{(\pi)^{1/2}} \times 2.54\ \frac{\text{cm}}{\text{in}} = 12.83\ \text{cm.}$$

The first core calculated by the code is 8 cm. in radius, then 10 cm., 12 cm., 14 cm., and 16 cm. while the height and composition of the core would be held constant. If the results of K-eff versus core size are plotted on a graph, the values for a core of 12.83 cm. in radius can then be read from the graph.

The composition of each core was calculated in the following manner:

$$\text{Volume fraction of the water} = \frac{\text{Total core area - area occupied by fuel plates}}{\text{Total Core Area}}$$

$$\text{Atom Density of Aluminium Metal} = \frac{\text{Area occupied by aluminium}}{\text{Total core area}}$$

$$\times \frac{\text{density of Aluminium}}{\text{Atomic weight of Aluminium}}$$

$$\times \frac{6.02 \times 10^{23}\ \text{Atom/ gm-atom}}{10^{-24}\ \text{Barns/cm}^2}$$

$$\text{Atom Density of } U^{235} = \frac{\text{Amount of } U^{235}}{\text{Vol. of core}} \times \frac{.602\ \text{Atom cm}^2/\text{Barns - gm-atom}}{\text{A of } U^{235}}$$

$$\text{Atom Density of } U^{238} = \frac{\text{Amount of } U^{238}}{\text{Vol. of core}} \times \frac{.602\ \text{Atom cm}^2/\text{Barns - gm-atom}}{\text{A of } U^{235}}$$

As the assembly core increases in size, the reflector thickness of the core decreases by the same amount. Therefore, for ease in calculations each core has been assumed to have 15 cm. of water as radial reflector. Since the mean free path of a neutron in water is only .66 cm., 15 cm. of water will be an infinite reflector in the calculation. The top and bottom reflector thickness are incorporated within the axial buckling term. Parameters used in the calculations of the core are listed in Appendix B.

Results of the Calculation

The results of the calculations provide an insight for the approach to criticality experiment. As mentioned previously, the geometry used for the mathematical model is different from the real core geometry. Consequently, the results of the calculation will indicate that the system is more reactive than it actually is because the surface to volume ratio in the model is smaller. This results in lower neutron leakage than in the actual system. The effective multiplication constant (K-eff) obtained from the calculations is plotted in Figures 22, 23, 24 and also plotted in Figure 25 which compares these results with those obtained by Jankowski (14). The K-eff obtained from the DISNEL code calculation is slightly less than Jankowski's calculation except for the S-1 core. The deviation in the two calculations can be explained by the combinations of different assumptions made in Jankowski's calculations. These assumptions were:

1. A fixed reflector saving of 5 cm. was used in all cases.
2. The U^{238} (content) in the core was ignored.
3. .07 in. thick plates with 9.45 gm. of U^{235} were assumed.
4. Rectangular parallelpiped geometry was used.
5. Two group calculations were employed.

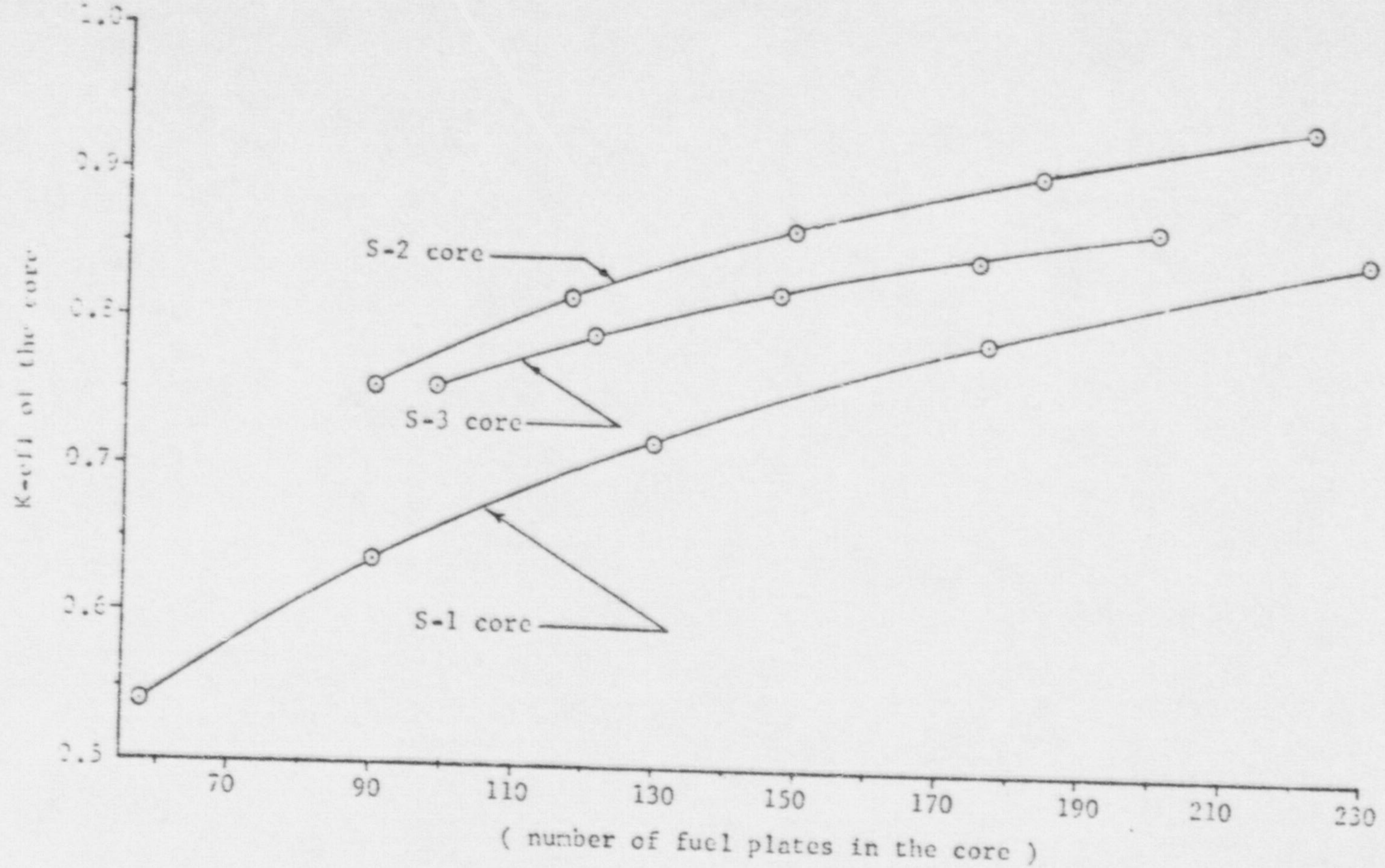

Figure 22. K-eff vs. number of fuel plates for "S" type core from DISNEL calculations.

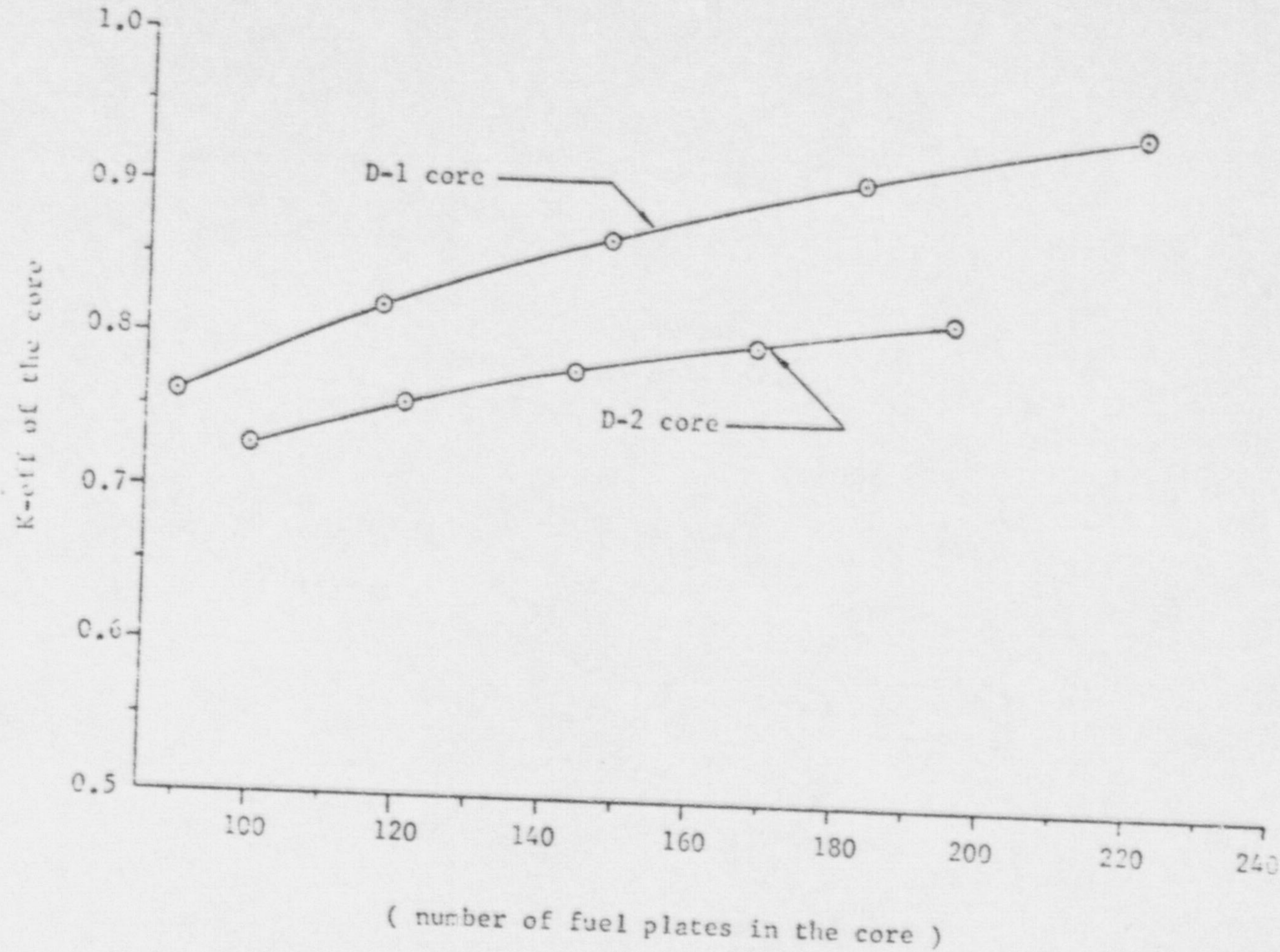

Figure 23: K-eff vs. number of fuel plates for "D" type core from DISNEL calculations.

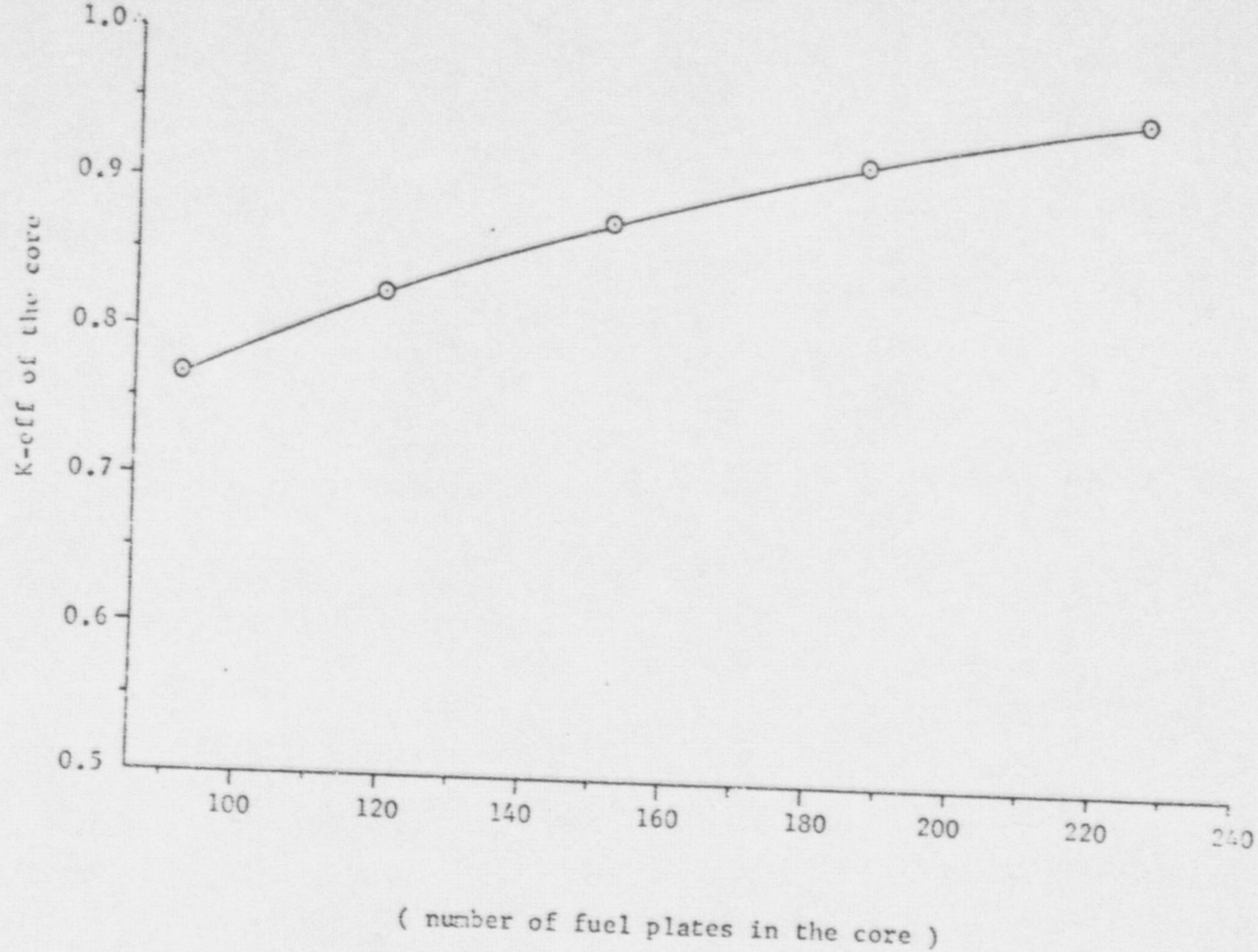

Figure 24: K-eff vs. number of fuel plates for T-1 Core from DISNEL calculations.

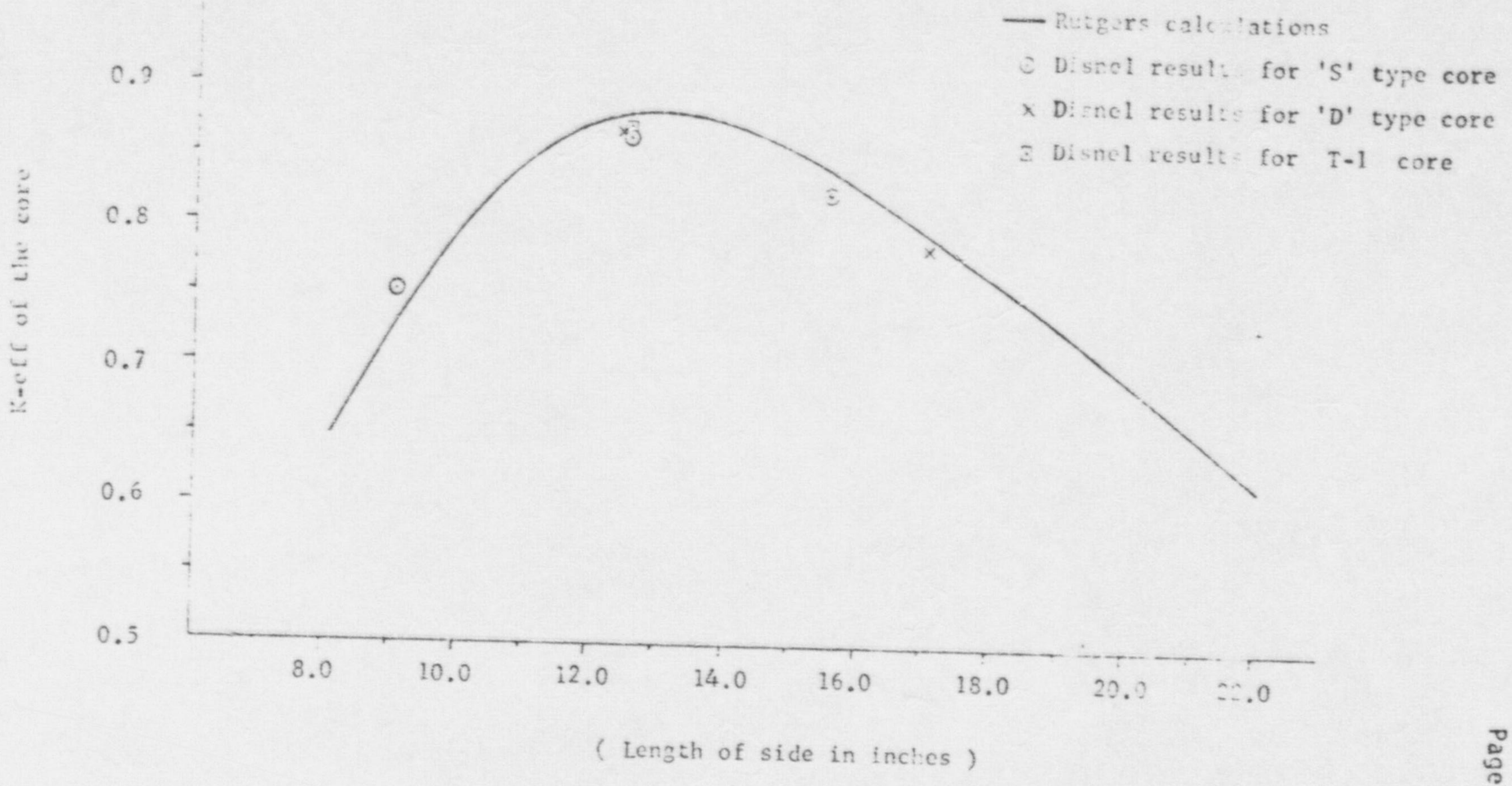

Figure 25: Comparison between DISNEL code calculation and Rutgers calculation.

VI. APPROACH TO CRITICALITY EXPERIMENT

Operations of the subcritical assembly at Rutgers has not revealed any criticality hazard. However, for safety reasons, criticality experiments will be performed on all six cores of the subcritical assembly. Additionally, these experiments are required by the Idaho State University Special Nuclear Material License application of the fuel for the subcritical assembly.

General Theory (15)

In a subcritical assembly, the introduction of an external neutron source will cause the neutron population of the assembly to increase and then reach an equilibrium level as long as the source strength is constant.

Suppose that, starting at some arbitrary time zero, No neutrons from an external source are introduced into a subcritical assembly. Each subsequent generation the neutron population in the assembly will be as follows.

Time in Neutron Generation	Neutron Population
0	No (source population)
1	No + No K-eff
2	No + No K-eff + No K^2-eff
.	
.	
.	
M	No (1 + K-eff + K^2-eff +. . .+K^m-eff)

One can observe that the increase in the neutron population is a power series which can be written as follows:

$$M = \frac{Nm}{No} = \frac{1 - K\text{-eff}^{(m+1)}}{1 - K\text{-eff}}$$

where m is the number of generation since t = 0.

In a subcritical assembly, where K-eff is less than unity and (m) is sufficiently large, the subcritical multiplication M is equal to one divided by one minus K-eff.

In the core where the system is just critical, the neutron population would continue to increase without limit. If the subcritical multiplication can be measured for various fuel loadings, and its reciprocal plotted vs. the amount of fuel used, the curve obtained may be extrapolated to 1/M = 0 to predict the mass required for criticality..

Neither N or N_0 may be measured directly, but a neutron detector placed at a suitable location in or near the core should provide a count rate which is proportional to the neutron density. A count rate without fuel, Co, is first measured and then a count rate, C, is measured with fuel in place. The value of C/Co obtained corresponds to M for this particular loading. Additional fuel is then added and a new value of C measured for this fuel loading. The observations are repeated and a graph of 1/M vs. amount of fuel is plotted.

Selecting a suitable location for the detector is complicated by the fact that most neutron detectors are sensitive to thermal neutrons while the neutron energy spectrum from the source is generally different from the fission spectrum produced in the core. Placing the detector in the center of the core is usually the best location. If this is not possible, the closest location to the core, within the reflector will provide the best results.

Procedure for Appraoch to Criticality

(1) Three independent neutron detection systems were installed in the subcritical assembly area for this particular experiment to make sure that the multiplication of the system is known at all times. Each of the systems consisted of a boron trifloride filled neutron proportional counter coupled to a suitable pulse-counting system. A detailed description of each pulse-counting system is included in the instrumentation section of this procedure. The detector for system number one was located inside the core tank next to the fuel-spacing assembly with the detector center approximately the same level as the core center. Detector number two was located a few inches above the water covering the core. It was attached to a movable steel member which can be removed for core access and replaced at a precisely located position prior to obtaining data. Detector number three was located within a water-tight aluminium tube next to (or inside of) the core. The center line of detector number three was approximately at the lower boundary of the fuel. The aluminium tubing which houses detector number three was attached to the fuel spacing assembly in such a manner as to remain fixed during the experiment. Detector number three was taken out of the aluminium tubing during lifting or lowering of the core and replaced when the core was properly secured inside the core tank.

(2) After the three neutron detectors had been placed in their proper locations as indicated in step 1, a neutron source was placed in the graphite block beneath the core tank. The source consisted of 18 gm. of plutonium mixed with beryllium so that the neutron emission rate is 2.2×10^6 neutrons per second. The source was placed in a hole previously prepared in the graphite blocks designated Y6, X6 and Y6, X7, which was then inserted into the graphite assembly. The source was approximately eight inches below the very center of the core tank which houses the uranium fuel. The source was inserted into the prepared hole using tongs long enough so that the operator can never be closer than three feet to this source. Calculations have shown that the fast neutron dose rate to the operator will be less than three Mr/hr during the loading. Since it is anticipated that the loading of the source will take only a few minutes, exposure to the operator is negligible.

(3) All three of the neutron detection systems were energized at this time and checked to make sure they were counting neutrons from the neutron source. A 0.02 thick sheet of cadmium was placed between each of the detectors and the source location to insure that the pulses obtained from each of the detectors are indeed due to the arrival of thermal neutrons and not due to spurious noise. No further operations were performed until it was evident that each of the three channels was operating properly.

(4) The water pump was energized and the water was pumped from the storage tank into the core tank until the water level reached the overflow point of the tank. Pumping was maintained long enough to make sure that the overflow line was clear and that water was indeed draining back into the storage tanks. The water pump was then stopped and the fuel spacing assembly for the core was attached to the lifting mechanism and lowered, without fuel, into the water-filled tank. It was observed that the displaced water drained automatically back into the storage tanks through the overflow line. It is necessary that this line remain open at all times so that the water level between the core and detector number two remains constant throughout the experiment.

(5) With this configuration the count rate at each of the three detectors was determined in order to estimate the sensitivity of each of the three detectors. From previous experience, it was anticipated that detector number two would provide a count rate in excess of two counts per second in its location in the critical assembly. It was anticipated, therefore, that a counting time of ten minutes would be adequate to provide a statistical accuracy in the count rate of approximately $\pm 2.5\%$. However, it should be pointed out that the statistical accuracy of this initial count is not a determining factor in the success of the experiment since extrapolation technique will be used in determining the value of the effective multiplication constant. As the effective multiplication constant approaches unity, the overall

multiplication of the system becomes infinite, and the extrapolation of reciprocal multiplication approaches zero. Any error in the initial count rate, therefore, becomes negligible as the count rate from the system becomes large. It was felt that the other two counters, being placed closer to the source than counter number two, provide count rates such that their statistical accuracy would be adequate in the ten minute count. In any event, the counts obtained from systems one and three would be such that the statistical accuracy should exceed $\pm$10 per cent for these systems.

(6) The fuel spacing grid was lifted out of the water with a winch assembly and bars placed below it so that there was no possibility of it accidentally falling back into the core tank. At this time one-third of the fuel, or 50 plates, was placed in the fuel spacing assembly designated for the particular core. Thus, one-third of the fuel was loaded in the initial loading.

(7) The fuel spacing assembly containing the 50 fuel plates was slowly lowered into the water-filled tank with constant monitoring of the instrumentation indicating neutron count rate. It was noted in the description of the instrumentation that counter one has a count rate meter which gives a continuous indication of the neutron count rate. The lowering of the fuel assembly was such that the fuel would not be lowered more than one foot in three time constants of the number one rate meter. If the count rate meter attached to counter one indicates that the count rate is higher than a factor of 3 above the initial count

rate, lowering of the core should stop until the effective multiplication constant is determined by obtaining an accurate count from detector one. Lowering of the fuel assembly continues until it is fully submerged or until one or both of the count rate meters indicate more than three times the count from the previous readings. If submersion has to be stopped because of high count rate, the assembled core should be lifted out of the water, 20 fuel plates removed from the core and the remaining lowered into the core following the same procedure. A multiplication of approximately three was predicted for this loading.

(8) When the fuel assembly had reached a stable position resting on the bottom support, a count was obtained from each of the three detectors of such magnitude that the statistical accuracy was better than ± 3 per cent for each. Using these counts, and those obtained previously with no fuel, a graph was initiated for each of the detectors, plotting the reciprocal of the system multiplication versus the number of fuel plates in the assembly. Each of the graphs was extrapolated to a value of zero for reciprocal multiplication to indicate the number of plates required to achieve a critical system. A straight line extrapolation was used even though detector placement indicates that a concave or convex curve would be anticipated. That curve indicating the minimum number of plates necessary for criticality was used for determining the assembly procedure in the following steps.

(9) The core assembly system was lifted from the water using the winch, and additional fuel plates were added to the assembly. The number of plates to be added shall not exceed more than half of the plates required from the lowest prediction of the previous step, or 25 plates, whichever is smaller. The plates added were placed in such a manner in the grid assembly so as to maintain as uniform a loading across the core as possible.

(10) The core assembly was lowered back into the water at a rate of less than one foot per three time constants of the number one rate meter as explained previously. Submersion was terminated if the count rate exceeds 2.5 times that obtained from the previous steady state measurement. Should the count rate exceed this limit, the core motion was stopped, an accurate count will be determined and the multiplication factor estimated from an extrapolation of the previously initiated curves for the detectors one and three. One half of the plates loaded in step nine was removed and step ten repeated.

(11) When the fuel spacing assembly containing the fuel plates was in place and resting on the core supports, count rates were determined for each of the three detectors. Sufficient counting time was allowed to make sure that each of the count rates was within ± 3 per cent statistical accuracy. From these counts, the multiplication of the assembly was determined and a point placed on each of the three graphs of the reciprocal multiplication versus the number of fuel plates. A straight line extrapolation was again used, using the previous point and

the point just located to determine the number of fuel plates necessary to achieve criticality. The lower value of the statistical uncertainty limit was used in the new point for determining this extrapolation.

(12) Steps nine, ten and eleven, were repeated until all 150 fuel plates have been added to the assembly and lowered into the core tank or until the multiplication factor for the system exceeds 0.93. Since calculations indicate that the effective multiplication factor with all 150 plates in the most reactive (T-1) configuration would not exceed 0.88, it is assumed that all 150 fuel plates will be in position following the assembly.

Safety Considerations

While it is anticipated that during the initial assembly of the ISU subcritical reactor, no safety problems will arise, each of the participants in the assembly experiment did observe standard safety precautions. Each person wore a film badge containing a neutron film and a gamma sensitive film. In addition, each participant wore a dossimeter sensitive to gamma radiation. All dossimeters were read immediately prior to the experiment and subsequent to the experiment to determine if any gamma radiation of significance had been received by any participant. Film badges will be processed by the commercial film badge supplier for Idaho State University and records kept of the exposure received by any participant. In addition to the personnel monitoring badges and dossimeters worn by the participants, two portable gamma ray monitors were in operation in the assembly room during the experiment. There is also a criticality alarm which is permanently attached to the wall of the room. These three gamma ray instruments, together with the neutron detection equipment used in the subcritical experiment, insured the participant that their exposure is below acceptable values.

Should either of the two portable gamma ray instruments or the wall-mounted criticality alarm indicate excessive gamma radiation levels, or should any of the three neutron monitoring instruments indicate excessive neutron levels, the experiment will be terminated and the room evacuated. Upon evacuation, the project director shall be responsible for shutting off the power to the control console. Removal of this power deactivates the solonoid valves and allows all the water from the core tank to drain back into the water storage system. Draining of this water reduces multiplication from the fuel assembly to such an extent that continued criticality would not be possible. The project director will direct one member of the experimental team to leave through the emergency exit of Room 24 so that the automatic alarm in the hallway of the building will be sounded. The project director himself will leave through the corridor next to the health physicist's office and will turn off all building air circulation equipment at the emergency station there when exiting from the area. The project director shall take one gamma ray monitoring instrument with him during this exit and when reaching the hall shall determine whether the radiation level is safe for continued occupancy. Should the gamma ray level at this point still exceed acceptable values, the project director will trip the building fire alarm system on his way up the stairs which will insure evacuation of the entire building. All other members of the critical assembly team will leave the building as quickly and directly as possible following the initiation of this emergency procedure.

If at any time during this experiment any person feels that an unsafe condition exists for any reason, he shall contact the project director. The project director shall order the experiment to be terminated until investigation has shown that the situation is safe. At no time during the assembly shall any member present be subjected to excessive radiation levels as indicated by any of the portable monitoring instruments present in the assembly room.

Instrumentation

The primary instrumentation for the critical assembly experiment for the ISU subcritical assembly will consist of the following three systems, the locations of which have been previously described.

System One: The detector for system one shall be a Reuter Stokes proportional counter filled with BF_3 gas to 70 cm. of mercury pressure. The boron in this detector is enriched to 93 percent boron 10. The active dimensions of this detector are approximately thirteen inches long and two inches in diameter. A Mech-Tronics Corporation Model 253 high voltage supply with a maximum voltage of 3000 volts will be used to power the detector. Canberra Industries Model 806 preamplifier will be used together with a Harshall Model NA-16 linear amplifier to amplify pulses from detector one. A Metronic Nuclear Company scaler Model 709 containing an integral timer will be used to determine the counts from the counter number one. A count rate meter will also be used to provide continuous count rate data from counter number one. The entire system number one will be housed in a "mini-bin" meeting AEC specifications.

System Two: System number two shall consist of a Reuter Stokes proportional counter filled with BF_3 to 40 cm. of mercury with the boron again being in excess of 93 percent boron 10. The active dimensions of the counter for system number two are approximately eight inches long by two inches in diameter. The high voltage supply for this system will be a Power Design Pacific Inc., Model 2K-10, with a maximum voltage of 2000 volts. A Metronic Company preamp with a Canberra Industries linear amplifier Model 1410 will amplify the pulses from counter number two. A Canberra time model 1492 coupled to a Metronic Nuclear Model 700 scaler determines the count rate. This system will be housed in a "mini-bin" meeting AEC specifications.

System Three: System three will consist of a Reuter Stokes detector Model RSN-127A with active dimensions of six inches by one inch in diameter. This detector will be connected to Ludlum Model 2200 portable scaler-rate meter with integral high voltage supply, power supply and timer.

The manner in which the neutron monitor systems one, two and three are set up are shown respectively in Figures 26, 27 and 28.

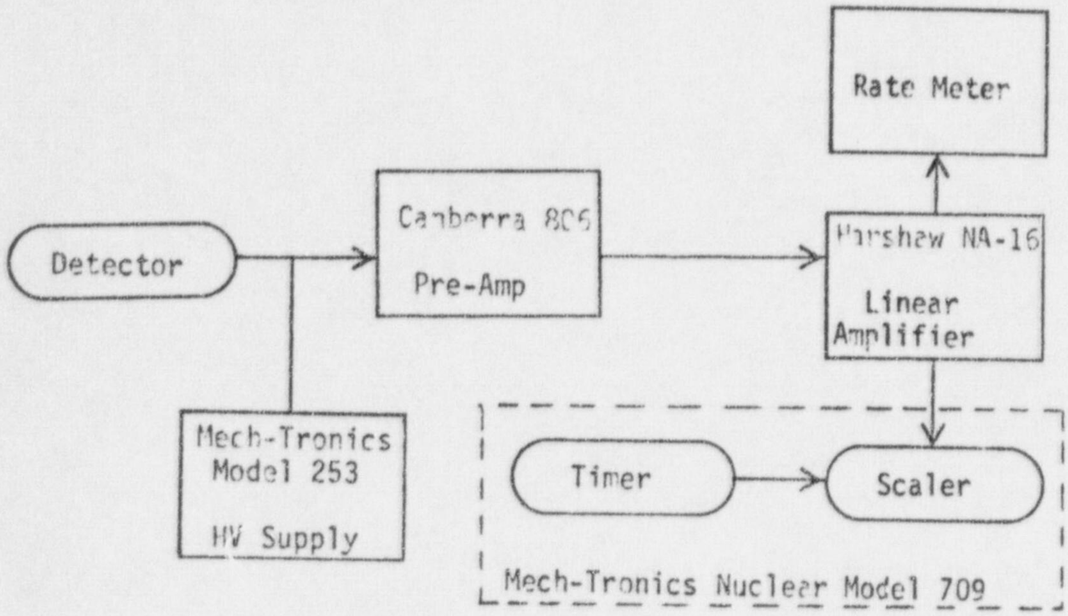

Figure 26: Components of Number 1 System

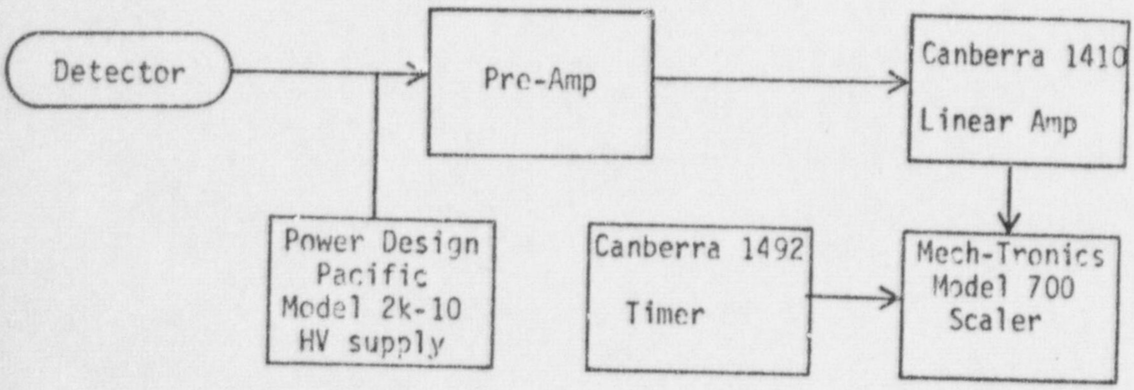

Figure 27: Components of Number 2 System

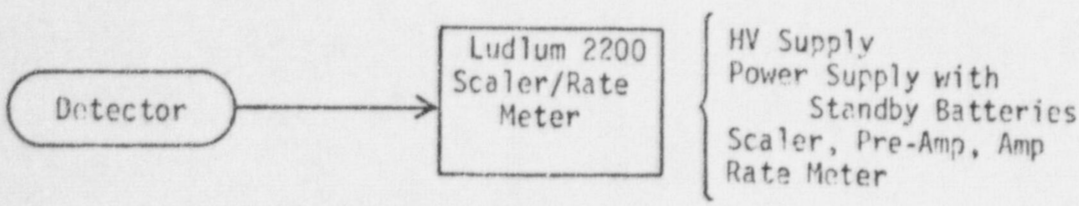

Figure 28: Components of Number 3 System

Result of the Criticality Approach Experiment

T-1 Core

Calculations have shown that this T-1 core was the most reactive core. For this reason this core was used for the first assembly to insure that no hazards exist in utilizing most of the fuel plates in the subcritical assembly. Two complete approachs to criticality experiment were performed with this core. The difference between the two runs was the manner in which the fuel plates were added to the core. During the first run the initial 49 plates were placed in each slot of the triple plate fuel spacing assembly. Subsequent additions of fuel plates were placed contiguous to the plate of the previous loading. In run number 2, three plates were inserted at the same time, in each slot of the triple plate fuel spacing assembly. Thus, the core size was increased with each loading while a constant fuel to moderator ratio was maintained within the core. Consequently, run number 2 gave a more uniform increase in the values of K-eff. The measured K-eff of .825 was obtained in both runs.

S-1 Core

The detector number 3 in this experiment was placed inside a water tight aluminium tubing. This tubing was attached to the top single plate grid. In doing so, the configuration of the S-1 core had to be modified. A quasi cylindrical core with a hollow center of 2 x 3 in^2 was chosen. Because the pre-experimental calculation utilized a cylindrical geometry there was speculation that the obtained experimental results would better follow the predictions. The loading configuration for the modified S-1 core is shown in Figure 32. The location of the detectors 1 and 2 has not been moved. . The highest K-eff value for this core is .806.

S-2 Core

The S-2 core was assembled essentially the same as was described in Chapter III. However, five plates had to be taken out near the middle of the core and added to one side of the core. The incore space was used for detector 3 tubing. Also, all 150 plates were used in this experiment instead of the planned 148 plates. Detectors 1 and 2 have not been relocated since the experiment with T-1 core. The highest measured K-eff of .875 was obtained from detector 3.

From the experiments it was evident that detector-location is the most critical factor. This phenomena can be explained by the small subcritical multiplication of the subcritical assembly core. Also, calculations have predicted that T-1 core was the most reactive of the cores. However, experimental results contradict the calculations by showing a K-eff of 0.875 for the S-2 core, 0.806 for the modified S-1 core, and 0.82 for the T-1 core. This contradiction can be explained by difference in the detector location in the three experiments. In the T-1 experiment the detector was located outside of the core while in the S-2 and S-1 experiments the same detector was in or very close to the center of the core. By placing the detector inside the core more fission neutrons are detected. As a result, a higher subcritical multiplication is observed. It is therefore, very difficult to conclude which of the cores is the most reactive. However, the experiments have demonstrated that criticality cannot be achieved by the T-1, S-2 and modified S-1 core.

TABLE III

Experimental Data From Approach to Criticality
Experiment of T-1 Core (Run 1)

Number of fuel plates		49	74	99	124	147
Detector 1	C in 100_3	17532	20194	20458	---	---
$Co = \frac{15937c}{100\ sec}$	$\frac{Co}{C}$	.9090	.7892	.7790	---	---
		1.09%	1.06%	1.06%	---	---
Detector 2	C in 400_3	18410	18765	19596	---	---
$Co = \frac{18315\ c}{400\ sec}$	$\frac{Co}{C}$	.9948	.9760	.9346	---	---
		1.04%	1.04%	1.03%	---	---
Detector 3	C in 15M	2056	2621	3962	5204	5932
Co=1055/15M	$\frac{Co}{C}$	.5131	.4025	.2662	.2027	.1778
	Error of Co/C	3.79%	3.65%	3.46%	3.38%	3.34%

Note: In this experiment the detector 1 was placed at the surface of the core tank, with its center approximately the same level as the core center.

TABLE IV

Experimental Data From Approach to Criticality
Experiment of T-1 Core (Run 2)

Number of fuel plates		49	63	84	126	147
Detector 1	C in 1000 sec	51552	61885	73550	100113	109878
Co = 35376	$\frac{Co}{C}$	.6862	.5716	.4810	.3534	.3220
in 1000 sec	Error $\frac{Co}{C}$	0.69%	0.65%	0.62%	0.62%	0.61%
Detector 2	C in 400 sec	22076	18093	18301	18754	18600
Co = 18431	$\frac{Co}{C}$	.8349	1.0187	1.0071	.9828	.9909
in 400 sec	Error $\frac{Co}{C}$	1.00%	1.05%	1.04%	1.04%	1.04%
Detector 3	C in 15 M	2788	3433	4163	5355	6487
Co = 1111.	$\frac{Co}{C}$	.3985	.3236	.2669	.2075	.1713
in 15 M	Error $\frac{Co}{C}$	3.55%	3.45%	3.38%	3.3 %	3.25%

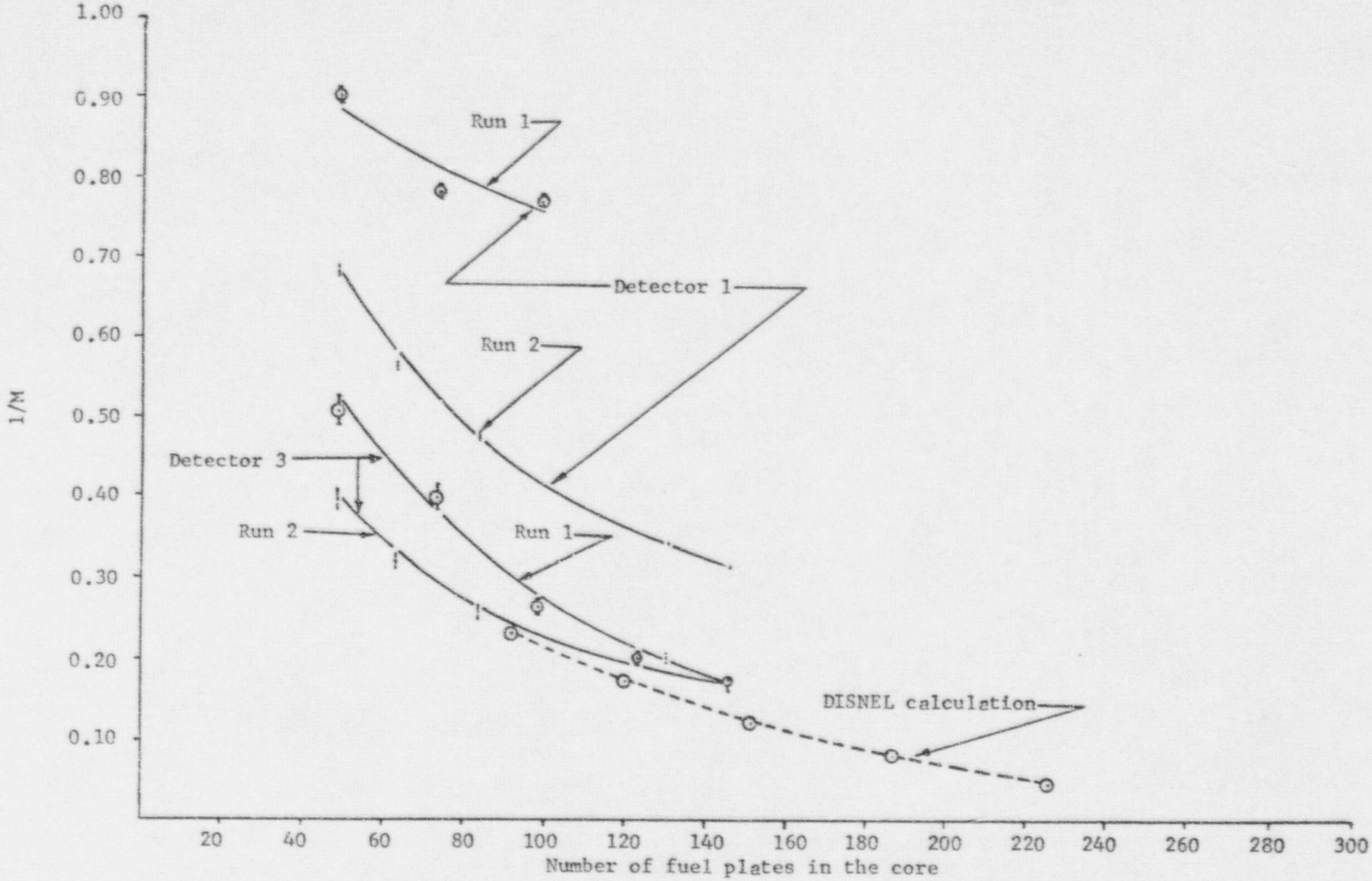

Figure 29: One over the subcritical multiplication versus number of fuel plates in the T-1 core.

TABLE V

γ Radiation Observed During
Run 2 of the T-1 Core

Number of fuel plates	0 Background	49	74	99	124	147
Reading taken at the surface of the core tank	0.01 $\frac{Mr}{hr}$	.03 $\frac{Mr}{hr}$	.03 $\frac{Mr}{hr}$	.03 $\frac{Mr}{hr}$	--	--
Reading taken at the top of the water	.015 $\frac{Mr}{hr}$	.035 $\frac{Mr}{hr}$	.038 $\frac{Mr}{hr}$	.04 $\frac{Mr}{hr}$	.075 $\frac{Mr}{hr}$	--
Reading taken inside Detector 3 protective tubing	--	.5 $\frac{Mr}{hr}$	--	--	--	--

TABLE VI

Experimental Data From Approach to Criticality
Experiment of Modified S-1 Core

Number of Fuel Plates		50	90	118	150
Detector 1 Co = 37637 in 1000 sec.	C in 10^3 sec	50425	60809	78833	92926
	$\frac{Co}{C}$	.7464	.6189	.4771	.4050
	Error $\frac{Co}{C}$	0.68%	0.66%	0.63%	0.61%
Detector 2 Co = 17778 in 400 sec.	C in 400 sec	17606	17578	16568	16725
	$\frac{Co}{C}$	1.0098	1.0114	1.0730	1.0630
	Error $\frac{Co}{C}$	1.06%	1.06%	1.08%	1.08%
Detector 3 Co = 1533 in 15 M	C in 15 M	2966	4580	6003	7905
	$\frac{Co}{C}$	.5169	.3347	.2554	.1939
	Error $\frac{Co}{C}$	3.15%	2.95%	2.86%	2.79%

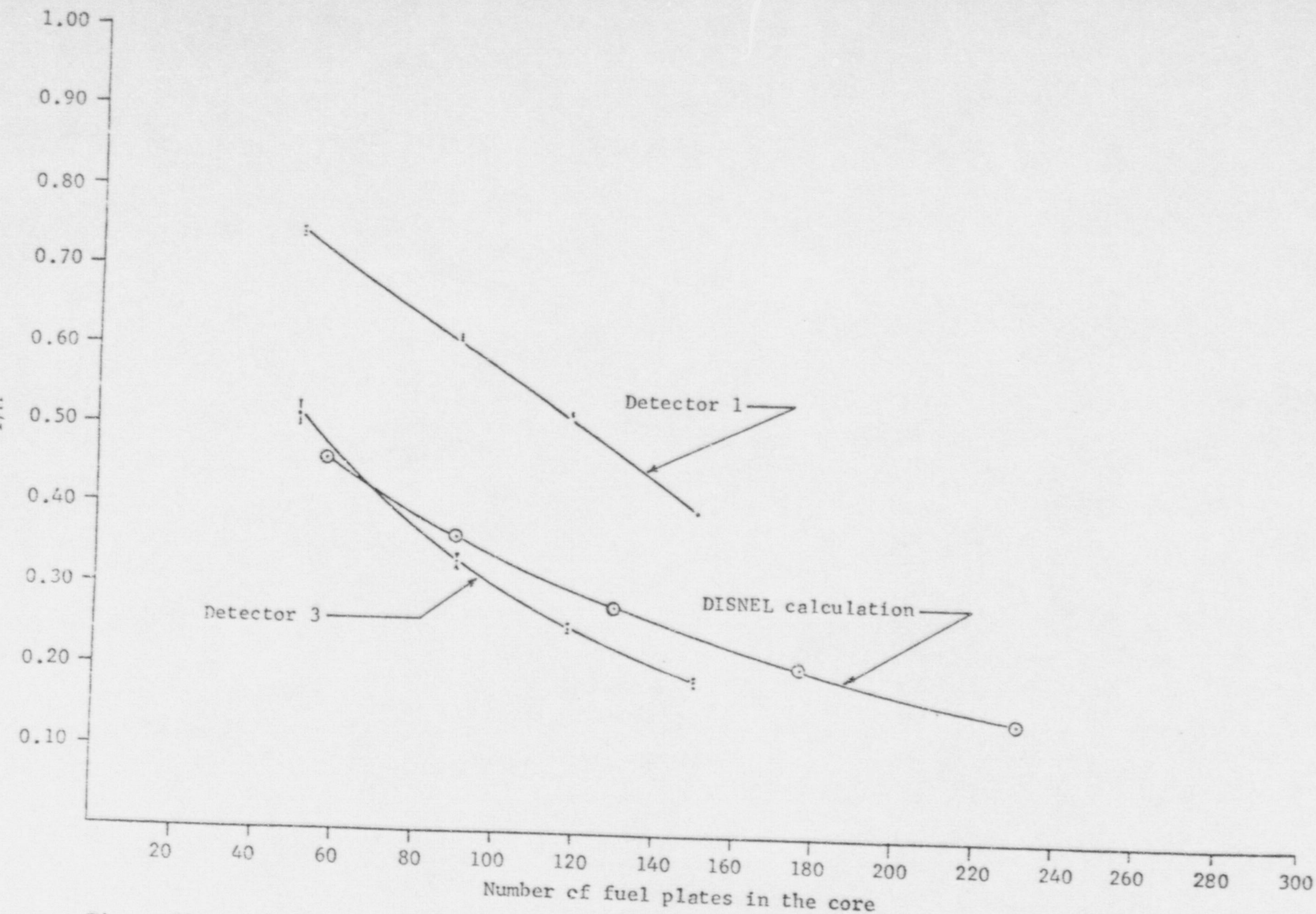

Figure 30: One over the subcritical multiplication versus number of fuel plates in the mod. S-1 core.

TABLE VII

Experimental Data From Approach to Criticality
Experiment of S-2 Core

Number of Fuel Plates		50	66	91	115	135	150
Detector 1	C in 10^3 sec	83935	92395	107204	125986	149166	158003
Co = 40455	$\frac{Co}{C}$	.4820	.4378	.3774	.3211	.2712	.2560
in 10^3 sec.	Error $\frac{Co}{C}$	0.61%	0.60%	0.58%	0.57%	0.56%	0.56%
Detector 2	C in 400 sec	17048	17870	17558	17557	17582	17739
Co = 17606	$\frac{Co}{C}$	1.0327	.9852	1.0027	1.0028	1.0014	.9925
in 400 sec	Error $\frac{Co}{C}$	1.07%	1.06%	1.07%	1.07%	1.07%	1.06%
Detector 3	C in 15M	3804	4863	7084	9351	10999	12188
Co = 1522	$\frac{Co}{C}$	.4001	.3130	.2149	.1628	.1384	.1249
in 15 M	Error $\frac{Co}{C}$	3.08%	2.94%	2.83%	2.76%	2.73%	2.72%

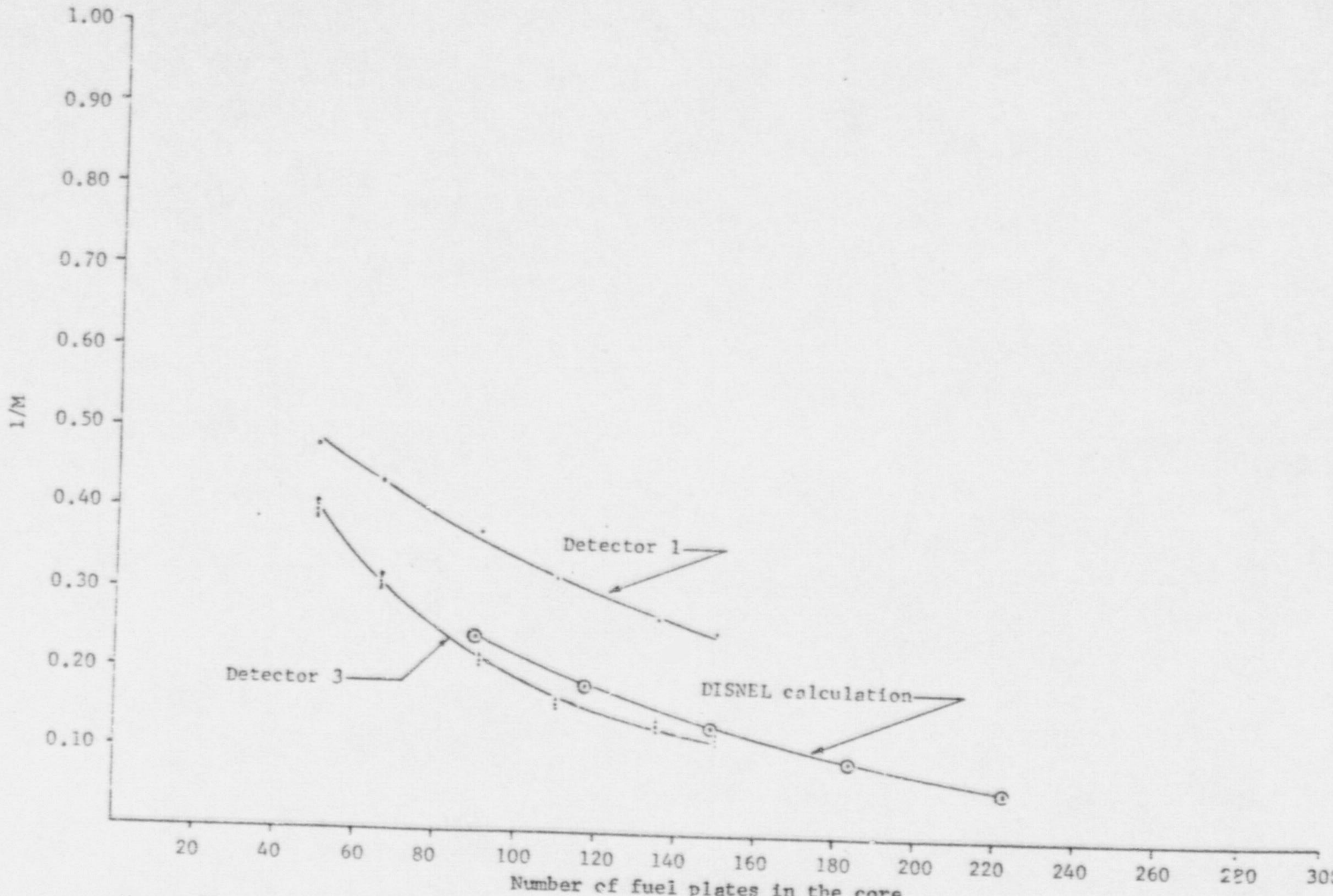

Figure 31: One over the subcritical multiplication versus number of fuel plates in the S-2 core.

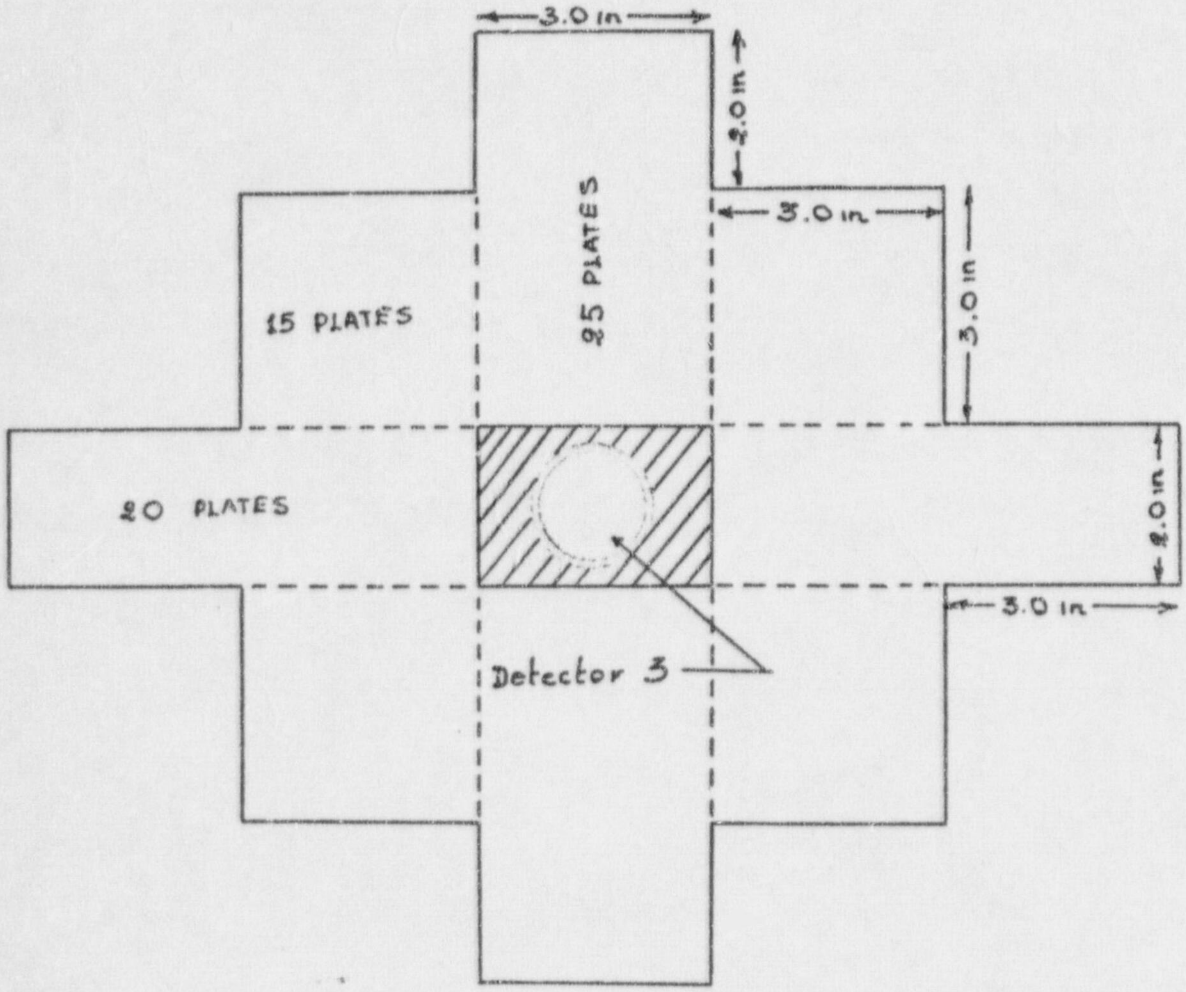

Figure 32: The modefied S-1 core

VII. EXPERIMENTAL PROCEDURE FOR THE USE OF THE IDAHO STATE UNIVERSITY SUBCRITICAL ASSEMBLY

Introduction

The Idaho State University subcritical assembly consists of an aluminium core tank 36 in. in diameter and 39.0 in. high. The core tank is placed on top of a graphite thermal column of 4.0 ft. square and 32 in. high. The subcritical assembly fuel consists of 150 aluminium clad uranium-aluminium plates. These fuel plates are 3.0 in. wide, .08 in. thick and 26 in. long. The fuel bearing portion of the plates is 2.75 in. wide, .04 in. thick and 24.0 in. long containing 10 gm. of U^{235}. These fuel plates may be assembled in a variety of geometric configurations to form the core of the subcritical assembly. Several different grid arrangements are available for maintaining proper spacing of the fuel plates in the core tank.

Moderator and reflector is provided by water which is pumped into the core tank from storage containers located nearby. The operating level of the core tank has been set at 33 in. of water. The constant operating level is maintained by the valve and overflow line. Figure 20 shows a schematic view of the core tank, the thermal column, the pumping assembly and the water storage container. Figure 21shows a schematic diagram of the shut down mechanism associated with the water handling system.

Safety Rules

It has been shown by both calculations and initial assembly experiments that no danger is present in the operation of the subcritical assembly. However, some precautions are imposed to persons involved with the operation of the subcritical assembly in order to minimize the accidental possibilities and ensure personal protection during an accident. These rules are:

1. The following materials are not to be taken into the subcritical assembly area without the permission of Professor A. E. Wilson. Permission may be granted for only one material at a time and cannot exceed the limitation stated: Graphite 8 lbs., Beryllium 2 lbs., Beryllium Oxide 4 lbs., Heavy Water 2 lbs., Fissionable Isotopes (U-235, U-233, Pu239) 3 gm of any one or combination. This limitation applies to other chemical forms or mixture containing the above material, detailed plans of the experiments to be conducted must be submitted before approval will be granted. Under no conditions are larger quantities of these materials to be carried into the subcritical assembly area.

2. Handling of fuel plates and foils should be performed with water-proof gloves or other appropriate hand covering.

3. Neutron source should be handled with source holder or tongs with a minimum length of 1 m.

4. A minimum of two persons must be present in the room while the assembly is in operation. One of the two persons must be an authorized operator.

5. Personnel inside Room 23 during the operation must carry on person a personal monitoring instrument either a pocket dossimeter or γ-neutron sensitive film badge. The dosage received by persons wearing the dossimeter should be recorded and kept in file.

6. Chemicals or materials which could cause a class B fire should not be stored unattended in Room 23.

7. No beverage or food should be taken inside Room 23.

8. In the event of an emergency where evacuation of personnel is needed the procedure below will be observed under the direction of the person responsible for the assembly at that particular time. either the faculty member present at such time or the student receiving permission to use the facility.

   All power to the pump and solenoid drain valve will be cut off. (This step should already be performed by the criticality alarm signal.) For added precaution the drain valve and the pump power switch will be turned off. The switch is located on a console between the exit door and the assembly tank.

   The portable gamma survey meter will be taken out of Room 23 if it is felt that no danger will result from carrying out this action.

   After evacuation, the door of Room 23 will be closed and the alarm attached to the emergency exit door of Room 24 activated. If radiation level in the immediate outside area is above normal, the building fire alarm will be activated to evacuate the entire building. Also, the building ventilation system will be deactivated. This action can be achieved by pushing the "Penthouse Power Emergency Trip" switch. This switch is located on the wall facing the health physicist's office.

   The person in charge will be responsible for the evacuation of all personnel persent in the basement or the building if the entire building must be evacuated. He will find a safe location for the personnel to stay during the emergency. No one will be allowed to reenter the dangerous area.

   The health physicist and the Chairman of the Department will be informed immediately. Their names and phone numbers are written on the door of Room 23 and on the health physicist's door.

9. An operating γ monitoring instrument must be placed on the graphite pedestal during any experiment performed with the subcritical assembly.

10. At the end of each experiment the neutron source must be stored in the paraffin filled storage container inside the storage vault of Room 22. The fuel plate may be stored in the locked core tank with solenoid valve deenergized or stored in the locked fuel container in Room 23. It is the responsibility of the person in charge of the experiment to check that all fuel plates and neutron sources are securely locked in their appropriate places.

Procedure for Operation

1. All the fuel plates which will be used in the experiment are loaded in the desired fuel spacing assembly. This loading should be performed with the fuel spacing assembly properly supported by core support devices. Inside the core tank brackets on the side of the tank and a honey-comb structure at the bottom of the tank are used to support the core. Also, the fuel may be loaded with the fuel spacing assembly above the tank. If this loading manner is preferred the fuel spacing assembly should be secured to the core lifting device and resting positively on iron bars placed across the core tank. Upon completion of loading, the core is lowered into the core tank such that the core rests on the core support devices.

2. At this point the neutron souce can be brought into Room 23, and placed at the desired location for the experiment. The neutron source can be placed in several locations inside the thermal column as well as inside the core itself. The paraffin filled source storage container is mounted on wheels. Therefore, it is recommended that the source container be brought into the room before the source is removed from the storage container.

3. At this point the gamma survey meter is turned on in the appropriate range and placed on the thermal column.

4. The power to the solenoid valve and pump is restored. This is accomplished by pushing the"valve and pump power reset" switch. Then the manual pump switch is turned on. Approximately 25 minutes is required to fill the assembly tank with 33 in. of water.

5. If, during the course of the experiment, the gamma survey meter or the criticality alarm meter indicated a gamma field of 20.0 Mr/hr, the assembly should be shut down and evacuation procedure is observed.

APPENDIX

APPENDIX A

DOSE CALCULATIONS

Rated Power

K-eff = .874

source neutrons = 2.1×10^6 n/sec

$$M \text{ (subcritical multiplications)} = \frac{1}{1-K\text{-eff}} = 7.94$$

Neutron Population rate = Source neutrons x M

$$= 2.1 \times 10^6 \frac{n}{sec} \times 7.94 = 1.67 \times 10^7 \text{ n/sec}$$

Fission Neutron population rate =

$$1.67 \times 10^7 \frac{n}{sec} - 2.1 \times 10^6 \frac{n}{sec} = 1.46 \times 10^7 \frac{n}{sec}$$

$$\text{Fission Rate} = \frac{1.46 \times 10^7 \text{ n/sec}}{2.47 \text{ n/fission}} = 5.90 \times 10^6 \frac{\text{Fission}}{\text{sec}}$$

$$\text{Rated Power} = \frac{5.9 \times 10^6 \text{ fission/sec}}{3.1 \times 10^{10} \frac{\text{fission}}{\text{watt-sec}}} = .19 \times 10^{-3} \text{ watts.}$$

Fast Neutron Dose Rate Calculation

In this calculation, bulk shielding facility data will be used. This data is published in ANL 5800 as Figures 7 - 11 on page 464.

At 30 cm. (or 12 in.) from the core surface a number of 15 ergs/gm-hr-watt is reported for fast neutrons. Conversion of this number in $\frac{\text{Rad}}{\text{hr-watt}}$ yield $1.5 \times 10^{-1} \frac{\text{Rad}}{\text{hr-watt}}$ or $1.5 \frac{\text{Rem}}{\text{hr-watt}}$ with fast neutrons RBE (Relative Biological Effectiveness) of 10.

Therefore

$$DR(f) \quad 1.5 \frac{\text{Rem}}{\text{hr-watt}} \times 1.9 \times 10^{-4} \text{ watts (rated power)} = .285 \frac{\text{MREM}}{\text{hr}}$$

Thermal Neutron Dose Rate Calculation

Again, BSF data will be used. At 30 cm. from the core surface a number of $2.5 \times 10^{5} \frac{\text{n (thermal)}}{\text{cm}^2\text{-sec-watt}}$ is obtained from BSF data.

$$2.5 \times 10^{5} \frac{\text{n}}{\text{cm}^2\text{-sec-watt}} \times 1.9 \times 10^{-4} \text{ watts} = 47.5 \frac{\text{n}}{\text{cm}^2\text{-sec}}$$

A conversion $8.6 \times 10^{-4} \frac{\text{Mr/hr}}{\frac{\text{n(.025 ev)}}{\text{cm}^2\text{-sec}}}$ is given in ANL 5800 page 466.

$$DR\ (th) = 47.5 \frac{\text{n}}{\text{cm}^2\text{-sec}} \times 8.6 \times 10^{4} \frac{\text{Mr/hr}}{\frac{\text{n(.025 ev)}}{\text{cm}^2\text{-sec}}} = .0408 \text{ MREM/hr.}$$

γ Photon Dose Rate Calculation

BSF data gives $1.3 \times 10^3 \frac{\text{ERGS}}{\text{gm-hr-watts}}$ at 30 cm from the core surface

$$1.3 \times 10^3 \frac{\text{ERGS}}{\text{gm-hr-watt}} \times 1.9 \times 10^{-4} \text{ watts} = 2.56 \times 10^{-1} \frac{\text{ERGS}}{\text{gm-hr}}$$

If a Conversion of $\frac{1\text{RAD}}{100 \frac{\text{ERG}}{\text{gm}}}$ is used.

$$DR_{(\gamma)} = 2.56 \times 10^{-1} \frac{\text{ERGS}}{\text{gm-hr}} = 2.56 \frac{\text{MRAD}}{\text{hr}} = 2.56 \frac{\text{MREM}}{\text{hr}}$$

Source Neutron Dose Rate

Assuming that 2.1×10^6 n/sec PuBe neutron source is located in the center of the core tank. The flux at the surface of the core tank will be:

$$2.1 \times 10^6 \frac{\text{n}}{\text{sec}} \times \text{geometric attenuation} \times \text{water shielding}$$

$$= 0.3.63 \frac{\text{n}}{\text{cm}^2\text{-sec}}$$

where geometric attenuation $= \frac{1}{4\pi R^2 \text{ (core tank)}} = 3.76 \times 10^{-5} \text{ cm}^{-2}$

and water shielding $= e^{-\Sigma_a R \text{ (core/tank)}} = 4.75 \times 10^{-3}$

Σ_a = Macroscopic neutron cross section for 4.5 Mev neutrons = .117 cm^{-1}. ANL 5800 page 466 gives a conversion factor of $.11 \frac{\text{mrem/hr}}{\text{n/cm}^2\text{-sec}}$ for 5 Mev neutrons. Therefore,

$$DR\ (Source) = .363 \frac{n}{cm^2\text{-}sec} \times .11 \frac{MRem/hr}{n/cm^2\text{-}sec}$$

$$= .04 \frac{MREM}{hr}$$

Total Dose Rate

$$DR_t = .285 \frac{MREM}{hr} + .0408 \frac{MREM}{hr} + 2.56 \frac{MREM}{hr} + .04 \frac{MREM}{hr}$$

$$DR_t = 2.9258 \frac{MREM}{hr} = 3 \frac{MREM}{hr.}$$

APPENDIX B

TABLE B-1 Core Parameters Used in DISNEL Code

CORE	Moderator to Core Volume Ratio	Aluminium Atom Density in $\frac{A}{BAN\text{-}cm}$	U^{235} Atom Density in $\frac{A}{BARN\text{-}cm}$	U^{238} Atom Density in $\frac{A}{BAN\text{-}cm}$	Dancoff correction Fuel	$\sigma_p(U^{238})$	$\sigma_p(U^{235})$
S-1	.5556658	.0144849	.0001214	.0004897	.2668	3700.21	14913.15
S-2	.7732463	.0073923	.0000620	.0002474	3.622×10^{-2}	9821.90	39259.27
S-3	.8500372	.0048887	.0000410	.0001636	5.7264×10^{-3}	15393.27	61430.37
D-1	.7741388	.9073629	.0000617	.0002464	3.9778×10^{-3}	5273.7	21048.0
D-2	.8769536	.0040112	.0000336	.0001342	7.3484×10^{-6}	9721.85	38817.
T-1	.7696571	.0075090	.0000630	.0002513	1.13×10^{-6}	3561.7	14195.

TABLE B-3 Self-Shielding Correction Factor for U^{235}

GROUP / CORES	13	14	15	16	17	18	19
S-1	.998	.972	1.0	1.0	1.0	1.0	1.0
S-2	1.0	.990	1.0	1.0	1.0	1.0	1.0
S-3	1.0	1.0	1.0	1.0	1.0	1.0	1.0
D-1	1.0	.980	1.0	1.0	1.0	1.0	1.0
D-2	1.0	.990	1.0	1.0	1.0	1.0	1.0
T-1	.996	.968	1.0	1.0	1.0	1.0	1.0

TABLE B-2 Self-Shielding Correction Factor for U^{238}

GROUPS / CORE	12	13	14	15	16	17	18	19
S-1	.670	.475	.520	1.0	1.0	1.0	1.0	1.0
S-2	.880	.650	.70	1.0	1.0	1.0	1.0	1.0
S-3	.887	.728	.76	1.0	1.0	1.0	1.0	1.0
D-1	.720	.530	.583	1.0	1.0	1.0	1.0	1.0
D-2	.879	.650	.698	1.0	1.0	1.0	1.0	1.0
T-1	.668	.462	.518	1.0	1.0	1.0	1.0	1.0

APPENDIX C

DISNEL Input Copy

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//X.FT08F001  DD  DSN=PFR*.JFKXSECT.UNIT=DKPRV2.DISP=SHR.VOL=SER=PERM02
//FT06F001 DD SYSOUT=A.DCB=(RECFM=FBA.LRECL=133.BLKSIZE=1596)
//FT05F001  DD  *
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  11    4    0    3        75.
   1    9  .[illegible]    12
    .570      .475      .520       1.0       1.0       1.0       1.0       1.0
   1   25  .0001214   12
    .008      .072      1.0        1.0       1.0       1.0       1.0
   1   13  .0144840    0
   1  100  .5556650    0
  12    4    0    3        75.
   1   99  .0002674   12
    .880      .650      .700       1.0       1.0       1.0       1.0       1.0
   1   95  .0000620   12
    1.0       .990      1.0        1.0       1.0       1.0       1.0
   1   13  .0073023    0
   1  100  .7772267    0
 102    0    1    3        75.
   0  100       1.0    0
   2          2
                          2     4    11    1          1    0    0
                0.0    .002      .0001           .01       0.0
  11   20       8.0    .002  100     20           15.       .002
1 S-1-10- I.S.U.  PRINYA
   2          2
                          2     4    11    1          1    0    0        0
                0.0    .002      .0001           .01       0.0
  11   25      10.0    .002  100     20           15.       .002
1 S-1-12- I.S.U.  PRINYA
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                          2     4    11    1          1    0    0        0
                0.0    .002      .0001           .01       0.0
  11   30      12.0    .002  100     20           15.       .002
1 S-1-14- I.S.U.  PRINYA
   2          2
                          2     4    11    1          1    0    0        0
                0.0    .002      .0001           .01       0.0
  11   35      14.0    .002  100     20           15.       .002
1 S-1-16- I.S.U.  PRINYA
   2          2
                          2     4    11    1          1    0    0        0
                0.0    .002      .0001           .01       0.0
  11   40      16.0    .002  100     20           15.       .002
1 S-2-14- I.S.U.  PRINYA
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                          2     4    12    1          1    0    0        0
                0.0    .002      .0001           .01       0.0
  12   35      14.0    .002  100     20           15.       .002
1 S-2-16- I.S.U.  PRINYA
   2          2
                          2     4    12    1          1    0    0        0
                0.0    .002      .0001           .01       0.0
  12   40      16.0    .002  100     20           15.       .002
1 S-2-18- I.S.U.  PRINYA
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                          2     4    12    1          1    0    0        0
                0.0    .002      .0001           .01       0.0
  12   45      18.0    .002  100     20           15.       .002
1 S-2-20- I.S.U.  PRINYA
   2          2
                          2     4    12    1          1    0    0        0
                0.0    .002      .0001           .01       0.0
  12   50      20.0    .002  100     20           15.       .002
```

```
1 S-2-22- I.S.U. PRINYA                                                          0
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   13   55       22.0        .002  100    20          15.       .002

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//X.FT08F001  DD  DSN=PERM.JFKXSECT.UNIT=DKPRV2.DISP=SHR.VOL=SER=PERM02
//FT06F001 DD  SYSOUT=A.DCB=(RECFM=FBA.LRECL=133.BLKSIZE=1596)
//FT05F001  DD  *
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  13     4     0     3          75.
   1    08   .0001636    12
    .887       .729          .760        1.0        1.0        1.0        1.0        1.0
   1    35   .0000413    12
    1.0        1.0          1.0        1.0        1.0        1.0        1.0
   1    13   .0048887     0
   1   100   .8603372     0
  21     4     0     3          75.
   1    08   .0002464    12
    .720       .530          .593        1.0        1.0        1.0        1.0        1.0
   1    05   .0003617    13
    1.0        .080          1.0        1.0        1.0        1.0   .    1.0
   1    13   .0073629     0
   1   100   .7741388     0
 100     0     1     3          75.
   0   100        1.0     0
   2          2                2    4   13    1          1    0    0
                  0.0        .002      .0001          .01       0.0
   13   45       18.0        .002  100    20          15.       .002
1 S-3-20- I.S.U. PRINYA                                                          0
   2          2                2    4   13    1          1    0    0
                  0.0        .002      .0001          .01       0.0
   13   50       20.0        .002  100    20          15.       .002
1 S-3-22- I.S.U. PRINYA                                                          0
   2          2                2    4   13    1          1    0    0
                  0.0        .002      .0001          .01       0.0
   13   55       22.0        .002  100    20          15.       .002
1 S-3-24- I.S.U. PRINYA                                                          0
   2          2                2    4   13    1          1    0    0
                  0.0        .002      .0001          .01       0.0
   13   60       24.0        .002  100    20          15.       .002
1 S-3-26- I.S.U. PRINYA                                                          0
   2          2                2    4   13    1          1    0    0
                  0.0        .002      .0001          .01       0.0
   13   65       26.0        .002  100    20          15.       .002
1 D-1-14- I.S.U. PRINYA                                                          0
   2          2                2    4   21    1          1    0    0
                  0.0        .002      .0001          .01       0.0
   21   35       14.0        .002  100    20          15.       .002
1 D-1-16- I.S.U. PRINYA                                                          0
   2          2                2    4   21    1          1    0    0
                  0.0        .002      .0001          .01       0.0
   21   40       16.0        .002  100    20          15.       .002
1 D-1-18- I.S.U. PRINYA                                                          0
   2          2                2    4   21    1          1    0    0
                  0.0        .002      .0001          .01       0.0
```

```
  21   45      18.0      .002  100   20        15.        .002
1 D-1-20- I.S.U.  PRINYA                                                        0
   2        2              2    4   21    1          1    0    0
              0.0       .002      .0001          .01       0.0
  21   50      20.0      .002  100   20        15.        .002
1 D-1-22- I.S.U.  PRINYA                                                        0
   2        2              2    4   21    1          1    0    0
              0.0       .002      .0001          .01       0.0
  21   55      22.0      .002  100   20        15.        .002
/*
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// COP=210.CPU=001.T=000
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//STEPLIB DD  DSN=PERM.JFKDSNEL.UNIT=DKPRV2.DISP=SHR.VOL=SER=PERM02
//X.FT03F001  DD  DSN=PERM.JFKXSECT.UNIT=DKPRV2.DISP=SHR.VOL=SER=PERM02
//FT06F001 DD SYSOUT=A.DCB=(RECFM=FBA.LRECL=133.BLKSIZE=1596)
//FT05F001  DD  *
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  22    4    0    3        75.
   1   98  .0001342   12
     .872      .650      .628          1.0         1.0         1.0         1.0         1.0
   1   95  .0000336   12
     1.0       .900      1.0           1.0         1.0         1.0         1.0
   1   13  .0040112    0
   1  100  .8760536    0
  31    4    0    3        75.
   1   98.000251342   12
     .668      .462      .518          1.0         1.0         1.0         1.0         1.0
   1   95  .0000630   13
     .996      .968      1.0           1.0         1.0         1.0         1.0
   1   13  .0075090    0
   1  100  .7606571    0
 100    0    1    3        75.
   0  100       1.0    0
   2        2              2    4   22    1          1    0    0
              0.0       .002      .0001          .01       0.0
  22   50      20.0      .002  100   20        15.        .002
1 D-2-22- I.S.U.  PRINYA                                                        0
   2        2              2    4   22    1          1    0    0
              0.0       .002      .0001          .01       0.0
  22   55      22.0      .002  100   20        15.        .002
1 D-2-24- I.S.U.  PRINYA                                                        0
   2        2              2    4   22    1          1    0    0
              0.0       .002      .0001          .01       0.0
  22   60      24.0      .002  100   20        15.        .002
1 D-2-26- I.S.U.  PRINYA                                                        0
   2        2              2    4   22    1          1    0    0
              0.0       .002      .0001          .01       0.0
  22   65      26.0      .002  100   20        15.        .002
1 D-2-28- I.S.U.  PRINYA                                                        0
   2        2              2    4   22    1          1    0    0
              0.0       .002      .0001          .01       0.0
  22   70      28.0      .002  100   20        15.        .002
1 T-1-14- I.S.U.  PRINYA                                                        0
   2        2              2    4   31    1          1    0    0
              0.0       .002      .0001          .01       0.0
  31   35      14.0      .002  100   20        15.        .002
1 T-1-16- I.S.U.  PRINYA                                                        0
   2        2              2    4   31    1          1    0    0
```

```
                    0.0          .002        .0001             .01         0.0
   31    40        16.0          .002   100     20             15.         .002
1 T-1-18- I.S.U.  PRINYA
   2          2                    2      4     31     1              1      0      0          0
                    0.0          .002        .0001             .01         0.0
   31    45        18.0          .002   100     20             15.         .002
1 T-1-20- I.S.U.  PRINYA
   2          2                    2      4     31     1              1      0      0          0
                    0.0          .002        .0001             .01         0.0
   31    50        20.0          .002   100     20             15.         .002
1 T-1-22- I.S.U.  PRINYA
   2          2                    2      4     31     1              1      0      0          0
                    0.0          .002        .0001             .01         0.0
   31    55        22.0          .002   100     20             15.         .002
/*
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Enclosure A3

Facility Emergency Plan

EMERGENCY PLAN FOR

THE NUCLEAR FACILITY AT

LILLIBRIDGE ENGINEERING LAB

AT IDAHO STATE UNIVERSITY

April 26, 1994

(Revision 5)

9405260114 3/4pp

CONTENTS

SECTION 1. INTRODUCTION

This emergency plan shall be used as a plan of action to follow in the event of a nuclear incident at the nuclear facility located at Idaho State University, Pocatello, Idaho.

It shall be used in conjunction with the State of Idaho Radiation Emergency Response Plan for all events involving a radiation accident, i.e., any release of radioactivity which may injure or contaminate a person. The Idaho plan delineates the actions to be taken by the Idaho State Police, Department of Health and Welfare, and the agreements with the local hospitals and the fire department in the event of a radiation accident. Those events which do not involve a radiation accident shall be the responsibility of the local operating staff and the administration of Idaho State University.

The nuclear facility consist of an AGN-201 nuclear reactor manufactured by Aerojet General Nucleonics in 1956. It is owned by Idaho State University and is operated under License Number R-110. The maximum power it is licensed to operate at is 5 watts. The fuel consists of uranium enriched to 19.88% uranium 235.

The AGN-201 reactor system consists of two basic units, the reactor and the control console. The reactor unit includes the core consisting of uranium dioxide dispersed in polyethylene, a graphite reflector, and the lead and water shielding. Fuel loaded control and safety rods are installed vertically from the bottom of the reactor unit, passing by the instruments which measure the power level. The rods are inserted by control mechanisms which provide safe and efficient operation of the reactor. The weight of the reactor unit, with the water shield, is 20,000 pounds; the weight of the console unit is 800 pounds.

The AGN-201 is located in the basement of the Lillibridge Engineering Labcratory at Idaho State University. Refer to Appendix 1 for the floor plans of the laboratory.

SECTION 2. DEFINITIONS

Emergency planning zone - Rooms 14, 19, 20, 22, 23, and 24 on the first level of the Lillibridge Engineering Laboratory.

Nuclear incident - Any unusual circumstances or occurrence that could lead to or cause damage to the reactor and/or sub-critical facility nuclear fuel or nuclear fuel cladding.

Operations area - The area inside room #20 and #23.

Radiation accident - Any release of radioactivity which may injure or contaminate a person.

Operations boundary - The walls, ceilings, and doors of rooms #20 and #23.

Nuclear facility - Consists of a AGN-201 nuclear reactor and the sub-critical assembly area.

Operations team - Consists of NRC licensed operators and health physicists.

Nuclear emergency - Any emergency which combines a radiation accident with any nuclear incident.

SECTION 3. ORGANIZATION AND RESPONSIBILITIES

Table 1 is a diagram of the local operating organization and shows the relationship to the State of Idaho radiation emergency response organization. The Idaho plan contains the responsibilities and duties of the auxiliary organizations who are committed to respond to a radiation accident, i.e., Idaho State Police, Department of Health and Welfare, the local fire department and hospitals.

The Idaho State University personnel who will respond to a nuclear incident are the Reactor Administrator, Reactor Supervisor, Radiation Safety Officer, and Campus Security. They will be advised by the School of Engineering faculty. There are no other local support organizations directly committed to respond to a nuclear incident other than the normal response provided by the local fire and police departments.

In order for the emergency plan to function as intended, it is essential that all coordinating personnel at Idaho State University be aware of their areas of responsibility and assure that their facilities and equipment are available and operational. The following is a list of University personnel and their areas of responsibility:

1. The Reactor Administrator and/or Reactor Supervisor are responsible for:
    a. Operations at Idaho State University should a nuclear incident occur.
    b. Notification of the State Police and Idaho Department of

Health and Welfare in the event of a radiation accident.

c. Requests for medical assistance or notification of an applicable hospital to prepare for patient care.

d. Safety regulations and practice within the nuclear facility.

e. Internal operations and assignments.

f. Routine checking of safety equipment and safety within the facility and assuring that employees are knowledgeable in equipment operation.

g. Requests for additional fire fighting assistance, and instructing the fire marshall concerning the hazards of the nuclear facility.

h. Evacuation plans and assembly areas.

i. Maintaining up-to-date notification roster of appropriate personnel and agencies.

j. Personnel accountability procedures at the Lillibridge Engineering Laboratory.

k. In their absence the Radiation Safety Officer shall assume their duties.

2. The Radiation Safety Officer is responsible for:

   a. Health physics assistance at Idaho State University.

   b. Authorizing volunteer emergency workers to incur radiation exposure in excess of normal occupational limits.

c. Manning check points or control points for surveying personnel and equipment.

d. Health physics at Lillibridge Engineering Laboratory and scheduling of personnel and working times in radiation areas.

e. Monitoring teams and environmental sampling at Idaho State University, analysis of samples, and maintenance of records.

f. Decontamination procedures and control.

g. Health physics for contaminated personnel until they are attended to by the proper medical personnel.

h. Insuring that all necessary health physics information is communicated to the appropriate agencies.

i. Personnel monitoring, personnel radiation records, and for scheduling of personnel for the operations team.

j. In his/her absence the Reactor Supervisor or the Reactor Administrator will assume these duties.

3. Idaho State University Campus Security will be responsible for:

a. Establishing area control and manning of check points.

b. Traffic control and traffic counting.

c. Assistance in communications and information dispersal.

d. Assisting State Police in the event of a radiation accident.

Refer to Appendix 2 for a list of organizations who have indicated they can provide assistance upon request either via the State of Idaho Radiation Emergency Response Plan or as normal duty of their organization.

SECTION 4. EMERGENCY PROCEDURES

Table 2 shows the emergency classification system for potential emergency situations which may occur in order of increasing severity.

Emergency Class	Type of Incident	Purpose
1. Unusual incident	Civil Disturbance Bomb Threat Theft of SNM	To secure the area. Investigate the situation. Alert the police.
2. Personal injury	Fire or Explosion	Same as 1 plus alert firemen and minimize damage.
3. Personal injury with contamination	Nuclear Emergency	Same as 2 plus activate the State Radiation Emergency Plan.

Table 2. Emergency Classification Table.

The emergency planning zone, EPZ, Rooms 14, 19, 20, 22, 23, 24 on the first level of the Lillibridge Engineering Laboratory, see figures 1 through 4. This emergency plan shall apply to the EPZ. There are no postulated accidents for the AGN-201 Reactor which would result in exposure of 1 rem whole body or 5 rem thyroid beyond the operations boundary.

In the event of an incident which requires evacuation of the building, i.e., fire or explosion, all personnel within the EPZ shall proceed to the shop area near the double doors to be accounted for and if radioactive contamination is suspected the potentially contaminated personnel will be separated from all others. In the event of a radiation accident the State of Idaho Emergency Radiation Response Plan will be initiated.

The emergency exposure guidelines are the same as the radiation dose standards for individuals in restricted areas as specified in 10 CFR 20.101. These guidelines are sufficient when the size and postulated radiation accidents are considered for the nuclear facility at Idaho State University.

The facility will maintain emergency procedures for dealing with various emergencies including nuclear emergencies, bomb threats, fires or explosions, theft or attempted theft of special nuclear material, and civil disorder.

The response procedures describe the type of response to be accomplished, the duties and responsibilities of the security organization and the management involved in the response. An up-to-date notification roster will be maintained in the Reactor Supervisor's office, in the School of Engineering administrative office, and the Campus Security office. The notification roster indicates the names and telephone numbers of those who will be notified immediately of any emergency and also the names and telephones numbers of those who may be called upon to assist. Refer to Appendix 2 for the notification roster.

A number of radiation monitoring devices are maintained at the Radiation Safety Office. The Radiation Safety Officer will determine which devices are to be used. A Geiger Counter will be the standard device for monitoring dose rates and contamination levels around the facility. Radiation monitoring devices are also maintained in the Chemistry and Physics buildings in the event that the Radiation Safety Office is not accessible. These devices include pencil dosimeters, hand held Geiger Muller counters, scintillation counters, and thermoluminescent dosimeters. All personnel entering a radiation area or a suspected radiation area shall have some method of determining the radiation field and personal dose. No person shall enter a suspected radiation area unless under the direction of the Radiation Safety Officer. The exception shall be for the police and the fire departments. If it is necessary that the police or firemen enter a radiation area without personal radiation monitoring devices the Radiation Safety Officer shall be informed immediately and will then survey the affected person for contamination and arrange for a whole body assay if necessary.

Nuclear Emergency

A nuclear emergency shall be any emergency which combines a radiation accident with any other nuclear incident. Any emergency which includes a radiation accident is a sufficient condition to initiate the State of Idaho Radiation Emergency Response Plan by calling the Idaho State Police, Region V. Evacuation of the Lillibridge Engineering Laboratory may or may not be required for a nuclear emergency. If the emergency is strictly a radiation accident and not combined with fire or explosion, building evacuation will be ordered if the radiation levels are above 10 mR/hr outside the operations boundary or if there are airborne radioactive materials.

In the improbable event that the nuclear reactor laboratory must be evacuated two exits are accessible from the reactor laboratory itself. These are: (1) through the double doors to the reactor laboratory and (2) an emergency escape hatch located in the roof. Two sets of stairs and an elevator lead to the ground level floor from which three exit points are available; one to the Southeast and two to either side of the display foyer. A ladder leads to an escape hatch in the ceiling of the reactor laboratory opened only from the inside.

The emergency exit sequence shall be: (1) Personnel in adjacent laboratory spaces shall be warned by operators to initiate evacuation; (2) The first person to reach the Emergency Ventilation Cut Out Switch (Located on the south wall, across from the health physicist's office shown in figure 1 will trip all ventilation off the line in the building preventing any further air exchange; (3) If time permits, radiological monitoring equipment will be taken from the reactor laboratory. If this is not possible other monitoring equipment has been stored at the Physical Science Building for emergencies; (4) The building fire alarm will be sounded to evacuate the entire building. The locations of the local fire alarms are at the bottom of the staircase on the south side (or on the way to the staircase on the north side) of the building; (5) The last person leaving the reactor laboratory area will shut all doors. All persons leaving the EPZ area shall proceed immediately to the northeast corner of the building near the large overhead door of the machine shop to be accounted for and be checked for radioactive contamination; (6) Staff personnel shall proceed to the Fire Department Emergency Command Center (if established) and provide information and assistance to the on scene commander; (7) The University Administration will be notified as soon as is practicable. The Reactor Administrator and the Reactor Supervisor will be notified immediately. They will, in turn, determine which State and Federal agencies shall be notified. The Radiation Safety Officer will be responsible for thorough radiation monitoring. The nuclear reactor laboratory will be reentered as radiological levels permit and then only by authorization of the Reactor Supervisor or his designated representative. The reactor system will be checked for damage. Refer to Appendix 3, Emergency Evacuation Plan.

Bomb Threat

1. The person receiving the threat should obtain as much information as possible. Ask the following questions:
   a. Where is the bomb?
   b. What kind of bomb is it?
   c. What time will it go off?
   d. Why are you doing this?

2. Notify:
   a. Idaho State University Campus Security
   b. Pocatello Police Department

3. The security officer, upon being notified of the threat, will proceed immediately and notify the following offices:
   a. Chief of Campus Security
   b. Pocatello Police Department
   c. University Administration
   d. Reactor Administrator
   e. Reactor Supervisor

4. Shut down the reactor.

5. The security officer will record the name and location of the person receiving the threat.

6. The removal or transfer of any radioactive material will be the responsibility of the Reactor Administrator and/or the Reactor Supervisor.

7. School of Engineering staff will assist with any subsequent searches of the Lillibridge Engineering Laboratory.

Fire or Explosion

1. Scram the reactor by pushing the power off button and check that the rods have scrammed by any of the following methods:
   a. Rods engaged lights out.
   b. Decreasing current trace on channel 2 and 3 strip charts.
   c. Period meter pegged low.

2. As soon as the scram is verified evacuate the building.

3. Notify the Pocatello Police Department by the quickest available means, i.e., radio, fire alarm, telephone.

4. The Fire Department will:
   a. Proceed to the area.
   b. Notify Pocatello Police Department and Idaho State University Campus Security for traffic control.

5. ISU Campus Security will notify the campus maintenance department. The maintenance department will provide a person to secure or activate building systems and alarms as necessary.

6. Notify the Reactor Administrator and/or Reactor Supervisor who will in turn notify:
   a. Idaho State University Radiation Safety Officer.

b. Idaho State Radiation Control Section.

c. U.S. Nuclear Regulatory Commission Region IV.

Theft or Attempted Theft of Special Nuclear Material

1. If an indication of a theft or an attempted theft exists in or around Rooms 20 or 23 of the Lillibridge Engineering Laboratory, immediately notify the Campus Security who will in turn notify:

    a. Chief of Campus Security

    b. Pocatello Police Department

    c. Reactor Administrator

    d. Reactor Supervisor

2. The Reactor Administrator and/or the Reactor Supervisor will proceed immediately to the Lab and inspect and inventory all special nuclear material. If a theft or attempted theft of special nuclear material has occurred, the following will be notified immediately:

    a. U.S. Nuclear Regulatory Commission Region IV.

    b. Idaho State Radiation Control Section.

Civil Disorder

1. Notify the Chief of Campus Security.

2. Campus Security will post guards in the basement of the lab.

3. Notify the Pocatello Police Department for riot and incident control.

SECTION 5. EMERGENCY FACILITIES AND EQUIPMENT

The emergency support center will be in the northeast corner of the machine shop near the large overhead door, Figure 2. Emergency control directions will be given from this area.

An ambulance shall be called for any person who may be injured. If that person is also contaminated with radioactive materials, the State of Idaho Emergency Radiation Response Plan shall be initiated by calling the Idaho State Police, Region V, and the receiving hospital shall be informed the injured person is potentially contaminated.

If a person is not injured but contaminated with radioactive materials, the State Plan will be initiated and decontamination procedures will begin under the direction of the Idaho State University Radiation Safety Officer in accordance with the State Plan.

The only emergency communications system in addition to the normal telephones are the hand-held radios which are used by campus security. Emergency communications will have to be by word of mouth if the telephone system is inoperable or the radios used by Campus Security are unavailable.

SECTION 6. MAINTAINING EMERGENCY PREPAREDNESS

The Reactor Administrator and the Reactor Supervisor are responsible to ensure the proper execution of the Emergency Preparedness Plan.

The training of University personnel who are responsible to act under this emergency plan is the responsibility of the Reactor Administrator and the Reactor Supervisor with the assistance of the Technical Safety Office in the area of radiological control. For those personnel and agencies not a part of Idaho State University, training is a responsibility of the Department of Health and Welfare, State of Idaho.

Rev. 5

The Idaho State University Reactor Administrator and the Reactor Supervisor will provide a training program at least once a year to train other University personnel who may be called upon to assist in the improbable event of a nuclear incident.

Rev. 5

University personnel who would be involved in a nuclear incident will be tested by annual drills. This will be accomplished by the unannounced initiation of a drill by the Reactor Administrator or by the Reactor Supervisor with written permission from the Reactor Administrator. Outside agencies will be contacted in advance and informed of the drill. University personnel will carry through with this action as though it were an actual emergency. Records of these drills will be entered into the facility operating records by the Reactor Supervisor or a licensed Senior Reactor Operator or Reactor Operator.

Rev. 5

The Emergency Preparedness Plan shall be audited under the cognizance of the Reactor Safety Committee at least once every two years. They shall evaluate the effectiveness of the plan and note

Rev. 5

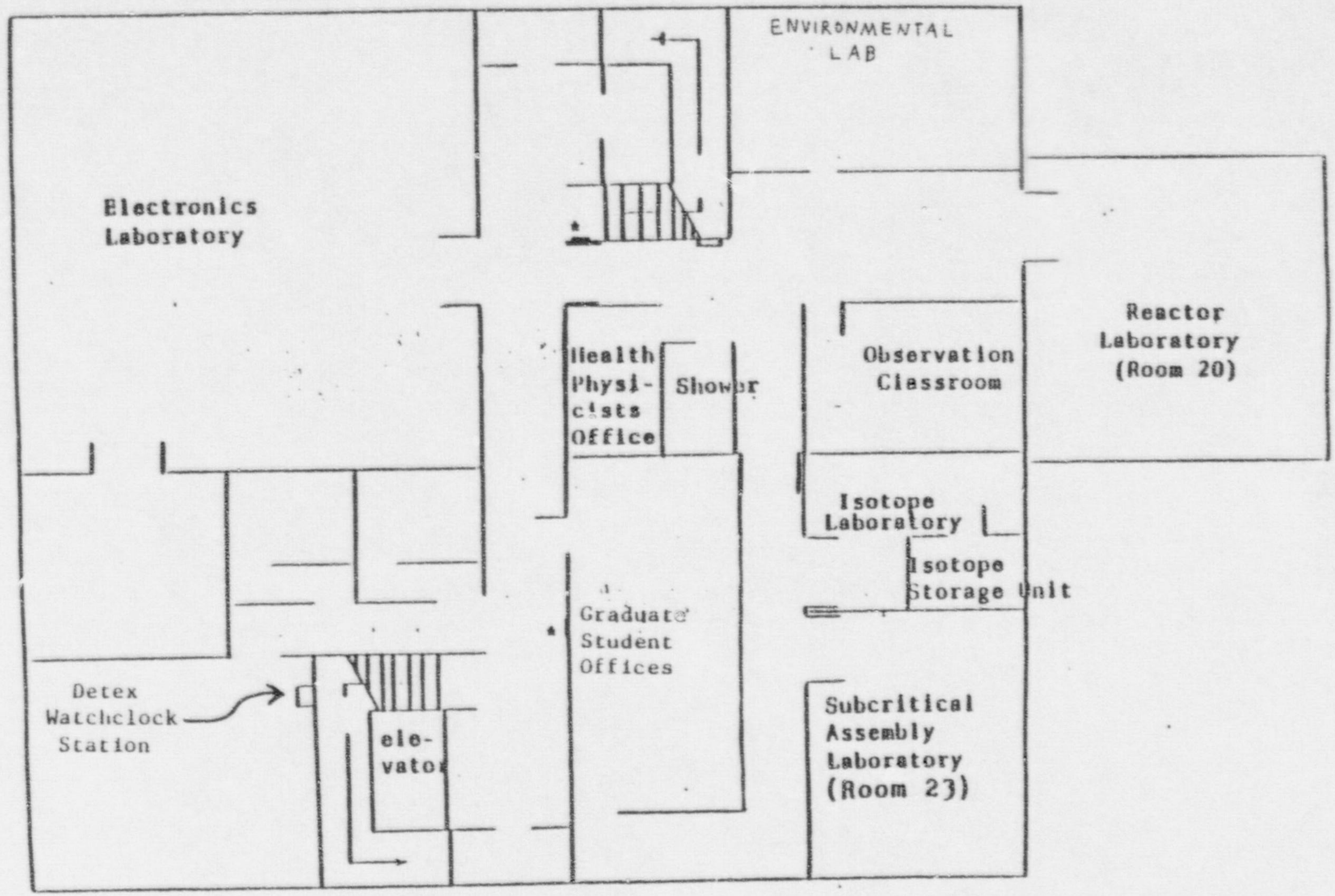
the results of the evaluation in their minutes. They shall also approve any changes which may be made to the plan.

Emergency equipment used for fire fighting, radiation detection and air sampling shall normally be checked for proper operation annually, but in no case shall the check be greater than 16 months. Batteries in portable equipment shall be checked prior to each use and annually unless previous experience dictates a more frequent check is required. A complete stock of replacement batteries shall be available for all battery powered emergency equipment. Emergency equipment will be inventoried annually.

APPENDIX 1.

FLOOR PLAN

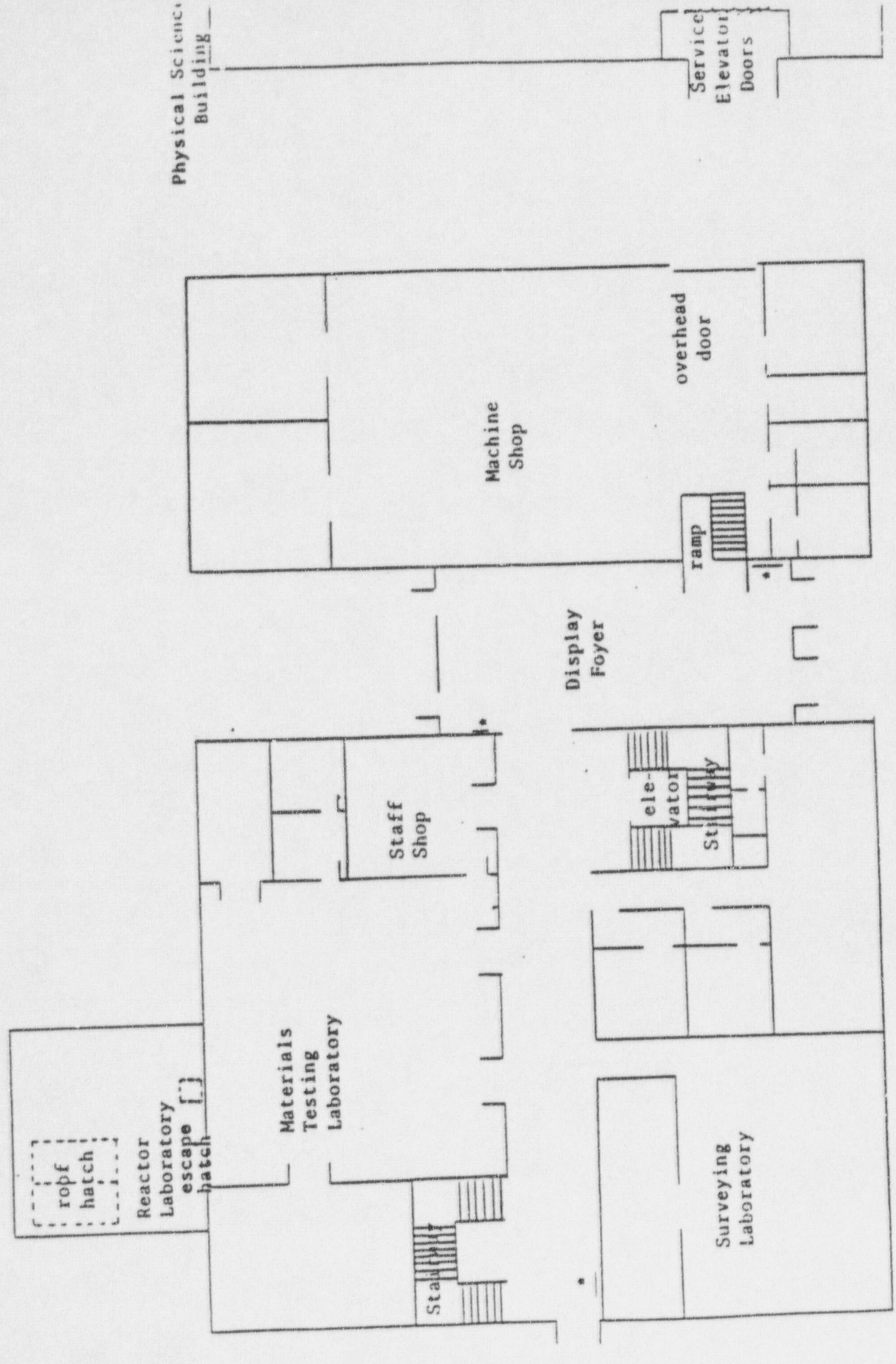
LILLIBRIDGE ENGINEERING LABORATORY



*Indicates fire-evacuation alarm actuators.

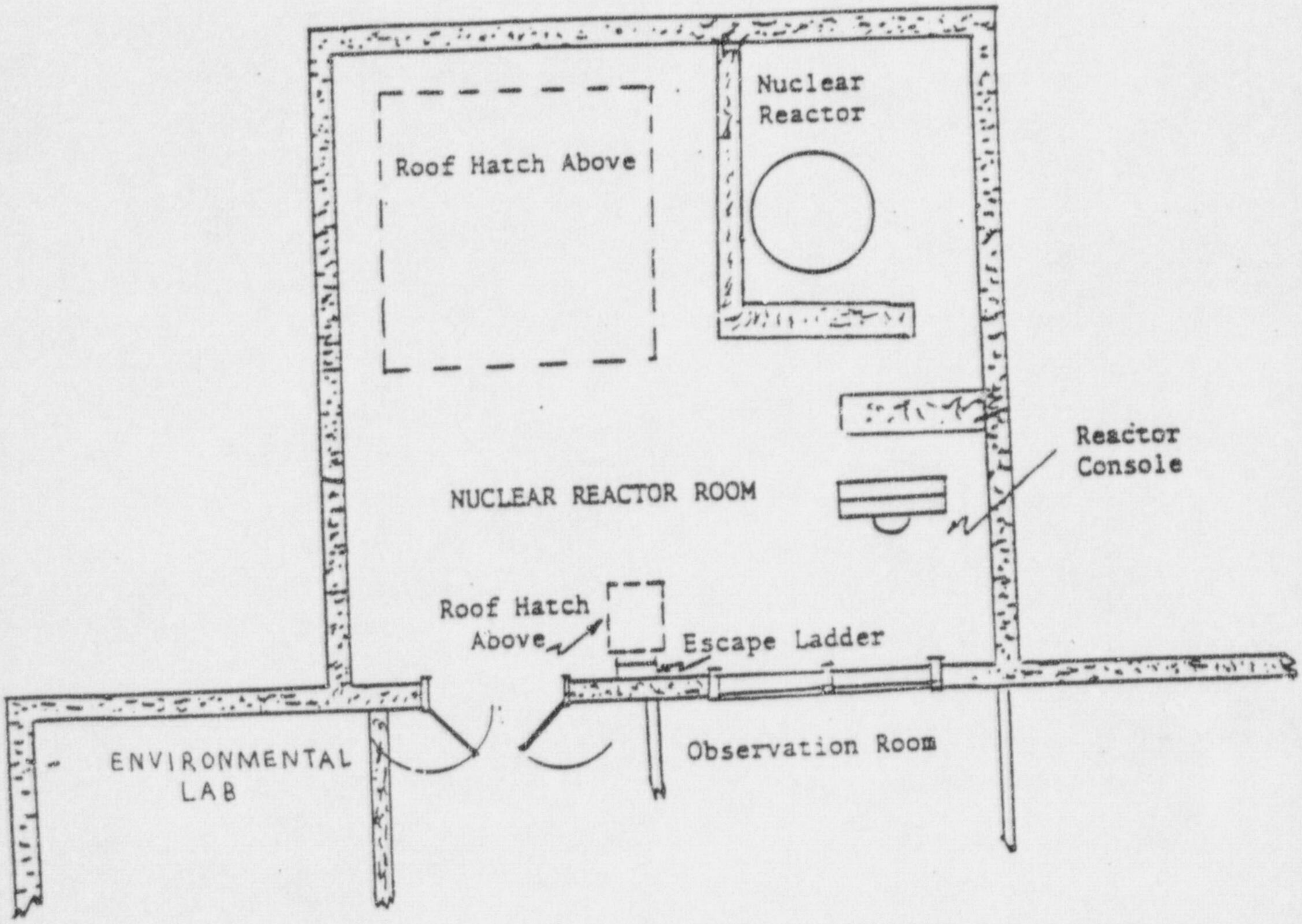
Lillibridge Building First Level Floor Plan (Basement)

Figure 1



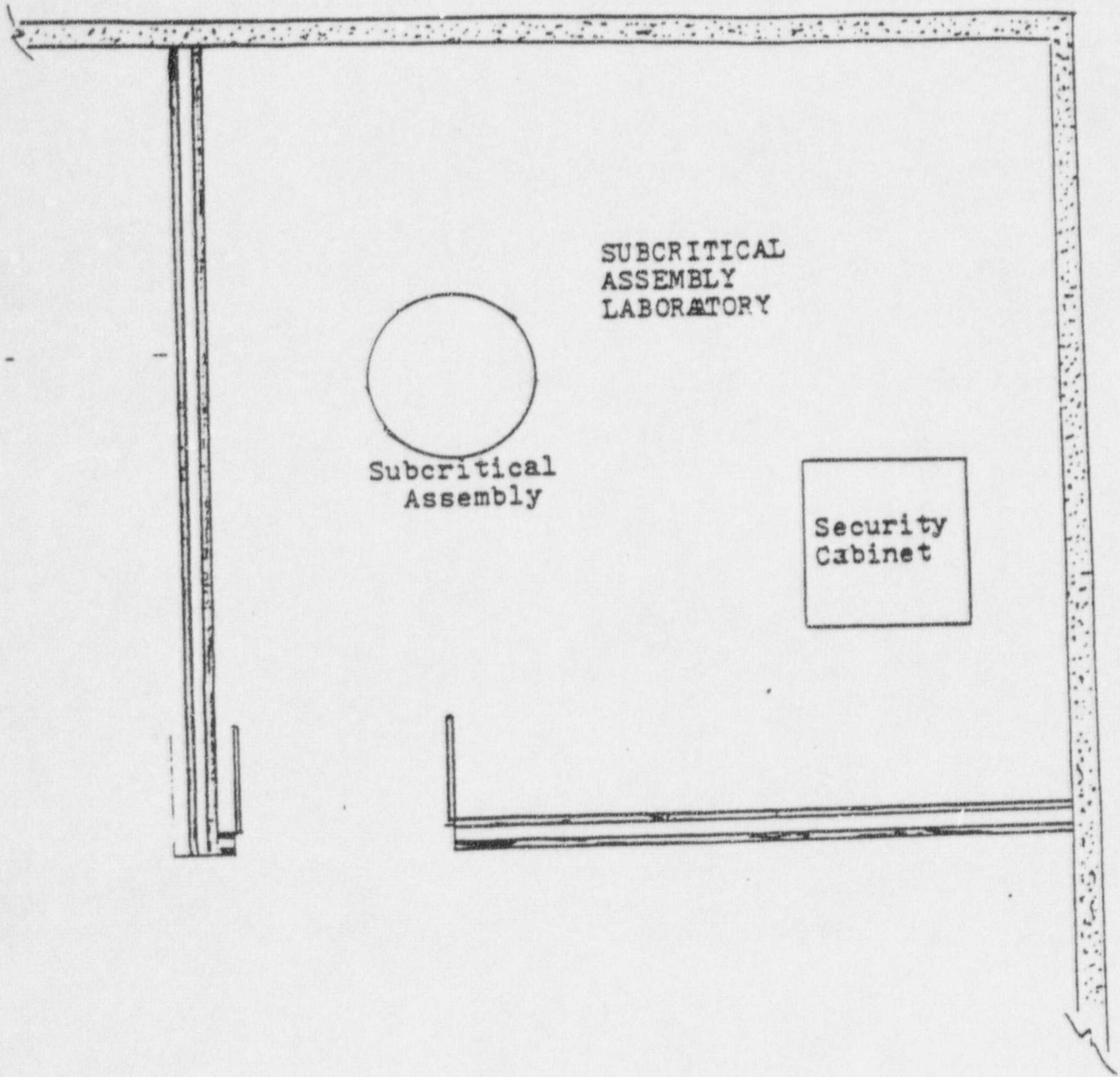
*Indicates fire-evacuation alarm actuators.

Figure 2
Second Level Building Floor Plan (Ground Level)



Floor Plan of Nuclear Reactor Laboratory

Figure 3



Floor Plan of Subcritical Assembly Lab
Figure 4

APPENDIX 2

NOTIFICATION ROSTER

REACTOR ADMINISTRATOR	J. BENNION	HOME: 208-233-3239 WORK: 208-236-3351
REACTOR SUPERVISOR	T. GANSAUGE	HOME: 208-234-0862 WORK: 208-236-3637
RADIATION SAFETY OFFICER	T. GESELL	HOME: 208-237-1076 WORK: 208-236-3669
ISU CAMPUS SECURITY		208-236-2515
POCATELLO POLICE DEPARTMENT		911
POCATELLO FIRE DEPARTMENT		911

SUPPORT NOTIFICATION ROSTER

IDAHO STATE POLICE	208-236-6066
BANNOCK REGIONAL MEDICAL CENTER	208-239-1800
POCATELLO REGIONAL MEDICAL CENTER	208-234-0777
ISU ADMINISTRATION	208-236-3440
NUCLEAR REGULATORY COMMISSION	301-816-5100

UPDATED 5/6/98

APPENDIX 3.

EMERGENCY EVACUATION PLAN

TO BE FOLLOWED IN THE EVENT OF A NUCLEAR EMERGENCY

WHICH HAS POTENTIAL OF CAUSING INJURY

1. The licensed reactor operator is cognizant of the detailed emergency plan. HE/SHE WILL BE IN CHARGE OF EVACUATION.

2. Use the normal room exit and building exits if possible. The escape hatch located in the roof is to be used only if normal exits are blocked by fire or radiation. Make sure exits to lab are closed after all persons are out.

3. The radiological monitoring instrument on the reactor console and the reactor log book will be brought from the laboratory room by the reactor operator.

4. If the radiation levels are above 10 mR/hr outside the operations area of the Nuclear Reactor Laboratory, the reactor operator will order building evacuation.

5. The first person to reach the Emergency Ventilation Cutout Switch (located on the south wall, across from the health physicist's office) will trip all ventilation off the line.

6. The reactor operator shall initiate building evacuation by tripping one of the building fire alarms located at the bottom of the staircase on the south side (or on the way to the staircase on the north side) of the building.

7. The reactor operator shall notify the Reactor Supervisor and/or the Reactor Administrator immediately.

8. The Reactor Supervisor and/or the Reactor Administrator shall be in charge of all building reentry.

EMERGENCY PROCEDURES (continued)

e. Reactor Administrator — Work: 208-524-0905; Home: 208-526-4907

f. Reactor Supervisor — Work: 208-236-3637; Home: 208-233-1173

4. The security officer will record the name and location of the person receiving the threat.

5. The removal or transfer of any radioactive material will be the responsibility of the Reactor Administrator and/or the Reactor Supervisor.

6. Engineering Department Staff will assist with any subsequent searches of the Lillibridge Engineering Laboratory.

B. Fire or Explosion at the Lillibridge Engineering Laboratory

1. Notify the Pocatello Fire Department by the quickest available means, i.e., radio, fire alarm, telephone 911.

2. The Fire Department will:

   a. Proceed to the area, but be sure that the fire fighters are wearing protective clothing and breathing devices.

   b. Notify Pocatello Police Department, Bannock County Sheriff, and/or Idaho State University Security to post guards around the area, and to keep out any unauthorized vehicles and persons.

PROPRIETARY INFORMATION
NOT FOR PUBLIC DISCLOSURE

CONTAINS 10 CRF 2.790 (D)
INFORMATION WITHHELD FROM
PUBLIC DISCLOSURE

ENCLOSURE 1

OPERATING ORGANIZATION

POSITION	NAME	TELEPHONE NUMBER
Reactor Administrator	A. Stephens	Work: 208-524-0905 Home: 208-526-4907
Reactor Supervisor	R. Clovis	Work: 208-236-3637 Home: 208-233-1173
Radiation Safety Officer	T. Gesell	Work: 208-236-3669 Home: 208-237-1076

REVISION 3
FEBRUARY 23, 1990

Enclosure A4

Figure A1. Locations of ISU Accelerator Center sites relative to the Nuclear Facility.

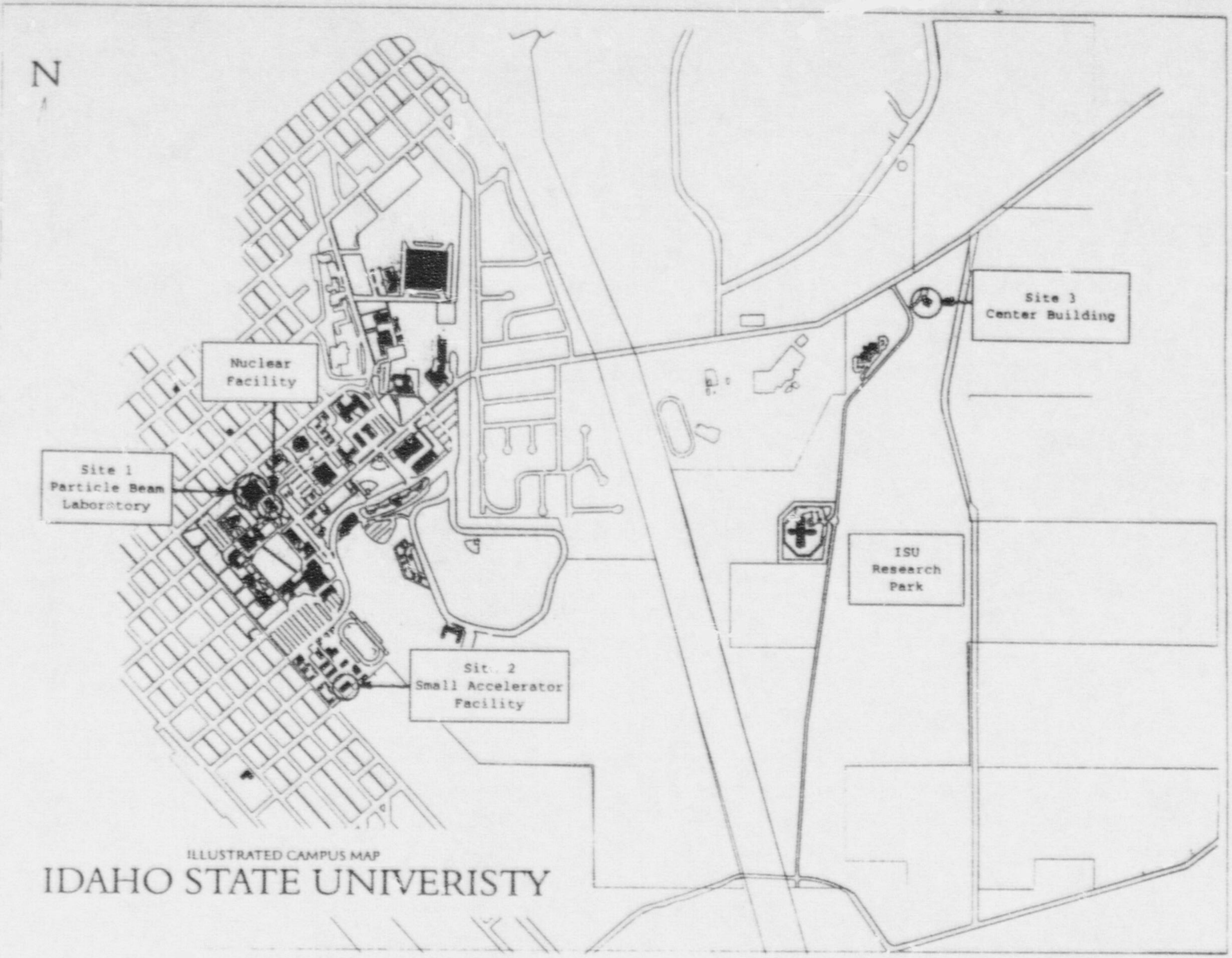


Figure A1. Locations of ISU Accelerator Center sites relative to the Nuclear Facility.

Enclosure A5

Figure A2. Map of the Pocatello area surrounding the ISU campus.

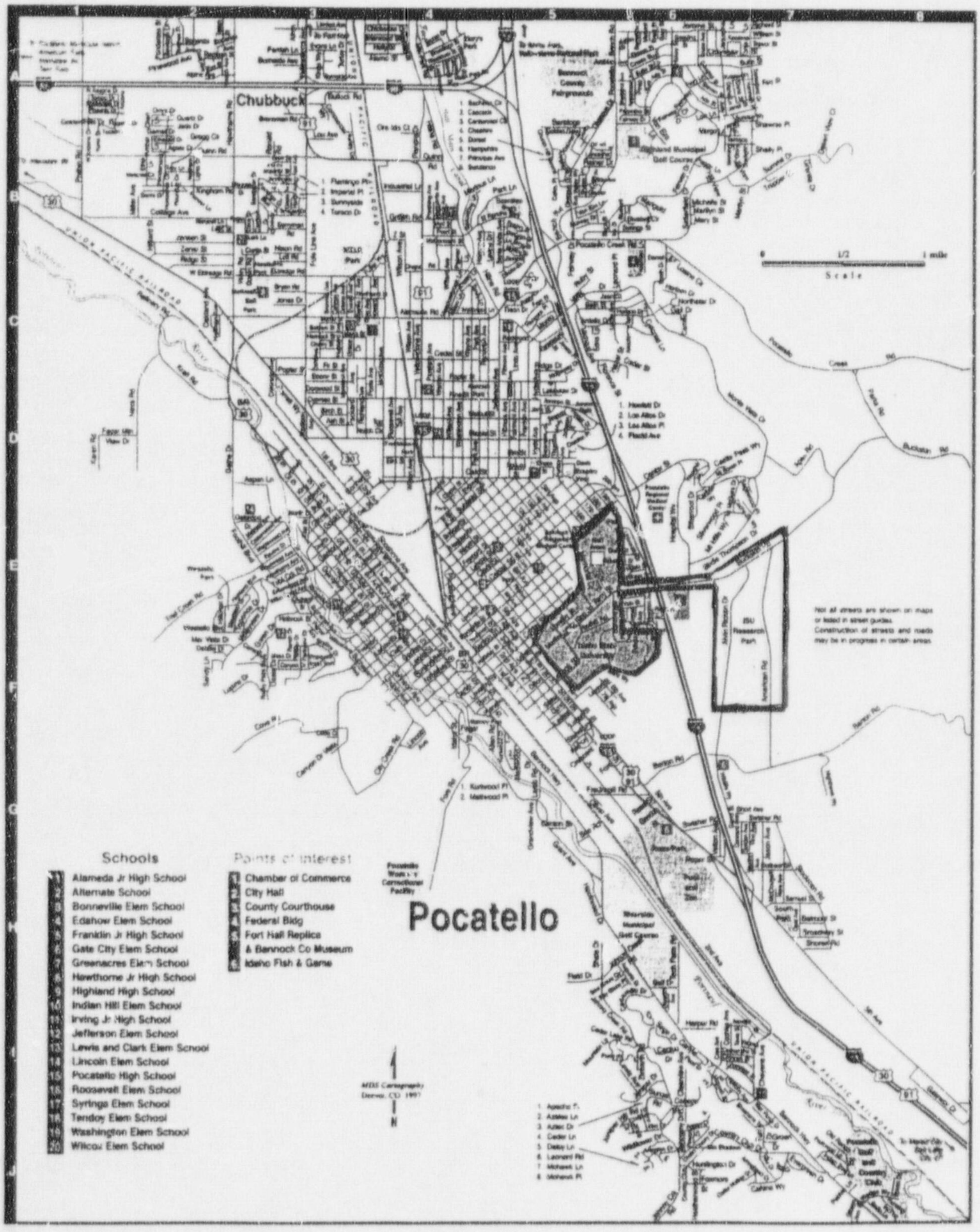


Figure A2. Map of the Pocatello area surrounding the ISU campus.