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SNAP CRITICAL ASSEMBLY-4B
PHASE III WATER IMMERSION EXPERIMENTS

AEC Research and Development Report

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.

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SNAP CRITICAL ASSEMBLY-4B
PHASE III WATER IMMERSION EXPERIMENTS

By
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ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.
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ABSTRACT

The Phase III SNAP Critical Assembly - 4B (SCA-4B) experiments which were conducted as one of the projects in the SNAP Aerospace Safety Program are described. These experiments demonstrate conclusively the effectiveness of: (1) a void-poison sleeve (shipping sleeve) in maintaining a bare core subcritical when it is internally flooded and fully reflected with water and (2) void-poison filler blocks in maintaining a beryllium-reflected core subcritical when it is submerged in but not internally flooded with water. Experimental data were also provided in support of the subsequent SNAPTRAN 10A/2-3 destruct experiment.

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I. INTRODUCTION

A. SCOPE OF PROGRAM

The safety aspects of SNAP 10A/2 and SNAP 8 reactors must be considered during three phases of operation after the cores are assembled. These are: (1) transportation to the launch site, (2) handling at the site, and (3) launch, during any of which the possibility of core immersion in water exists. The overall objectives of this program were to determine the reactivities of water immersed cores and to establish means of maintaining subcriticality.

The experimental program was conducted in three phases, all of which have been completed. Phase I and II are reported in Reference 1 and Phase III is covered herein. All of the latter experiments were conducted in the SNAP Generalized Critical Facility, which is described in Reference 2.

The objectives during Phase III of the experimental program were as follows:

- 1) To demonstrate the capability of a combination of poison and void sleeves to maintain a bare core subcritical when internally flooded and fully reflected with water
- 2) To demonstrate the capability of filler blocks in maintaining subcriticality when a beryllium-reflected core becomes submerged in water (but not internally flooded)
- 3) To provide experimental data to verify analytical techniques for predicting the effectiveness of poison sleeve designs. (The object is to minimize the amount of testing required in developing new sleeves.)

During the conduct of the Phase III experiments, the SNAPTRAN (SNAP Transient) Experimental Program at the National Reactor Testing Station (NRTS) in Idaho was started. Consequently, the scope of Phase III was expanded to provide experimental data for this program. The object of the tests were to:

- 1) Determine the critical loading and excess reactivity of the SNAPTRAN 10A/2 -3 core for comparison with another SNAP 10A/2 core loading (SCA-4)
- 2) Determine the nuclear feasibility of utilizing the ejection of a surrounding poison sleeve to initiate the destruct transient of the SNAP-TRAN 10A/2 -3 core.

B. CONCLUSIONS

The main conclusions derived from Phase III of the water immersion program are as follows:

- 1) A combination void and poison sleeve (designated shipping sleeve) is capable of maintaining both SNAP 10A/2-and SNAP 8-type bare cores subcritical when they are immersed in and internally flooded by water.
- 2) A combination void and poison filler block is capable of maintaining a beryllium-reflected SNAP 10A/2-type core subcritical when it is immersed in but not internally flooded by water. (This was not studied for a SNAP 8-type core.)

The following conclusions were derived from the tests conducted in support of the SNAPTRAN Program:

- 1) The SNAPTRAN 10A/2 -3 core had somewhat less total reactivity than the SCA-4 loading. This comparison provides a reference for correlating SNAPTRAN experiments with flight system cores.
- 2) It is feasible, from a nuclear standpoint, to initiate the SNAPTRAN 10A/2 -3 destruct transient by ejecting a poison sleeve.

II. DESCRIPTION OF EXPERIMENTAL APPARATUS

A. SNAP CRITICAL ASSEMBLY -4B (SCA-4B)

1. SCA-4B Machine

The SCA-4B machine was the same as that used in the earlier phases of the program and described in References 1 and 3. It was found during Phases I and II that the upper tank provided sufficient reactivity control. Therefore, in Phase III, axial reflection at the bottom of the core was provided by a deeper base tank and the control cap was not used. Photographs of the assembly are shown in Figures 1 and 2. The two stainless-steel drums containing the supply water can be seen behind the assembly in Figure 2. The upper reflector tank and the core vessel are welded together and attached to the upper support structure as shown in Figure 1. The base tank was attached to the passive moveable frame, which is shown in the down position in Figure 1 and in the up position in Figure 2. During operation, this frame was raised a total of 21 in. by the electromagnet (see Figure 1). The base tank was raised by remote control and was observed on remote TV. A scram action could be accomplished by de-energizing the magnet which then allowed the frame and base tank to fall. Free fall occurred during the initial 8 in. of travel and the remainder of the descent was controlled by a hydraulic snubber. The total time lapse between activation of the manual scram button and the frame contacting the snubber was measured to be 0.486 ± 0.014 sec. Normal draining of the water in both top and base tanks could also be initiated on scram.

A schematic representation of the core vessel and water tank assembly is shown in Figure 3. No attempt is made to present true proportions and highly accurate details because several different tank assemblies (see Appendix B) and many components were used during the course of the experiments. The cover plate was bolted to the core vessel, with the indicated O-ring providing a water-tight seal. When the base tank was raised to the position shown in Figure 3 and filled with water, it provided the lower portion of the core with effectively infinite axial and radial reflection. Filling the top tank resulted in a fully reflected core. In the dry-core mode of operation, an aspirator was connected to the core vessel drain pipe to remove continuously any water that might enter the core. In the internally flooded mode, the bottom of the core vessel had perforations

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Figure 1. SCA-4B Critical Assembly Machine
(Base Tank in Lowered Position)

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Figure 2. SCA-4B Critical Assembly Machine
(Base Tank in Raised Position)

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Figure 3. Schematic Diagram of Core Vessel and Water Tank Assembly

which allowed simultaneous filling of the core and base tank. In this case, the base tank wall extended above the top of the core so that complete flooding of the core could be achieved. This configuration is shown in Figure 2 in which it can be seen that the base tank partially encloses the top tank.

2. Water System

A photograph of the water system is shown in Figure 4. The compressed gas bottle provides pressure for operating the pneumatic valves. A schematic diagram of the water system is shown in Figure 5. The system was fail-safe in that drain valves were normally open, fill valves were normally closed, and the water supply drums were well below the level of the assembly tanks (see Figure 2). Either a scram or an electrical or pneumatic power failure would cause draining of the water from the top and base tanks into the supply drums. The time to drain the base tank water to the bottom of the core with the base tank in the up position was 39 sec.

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Figure 4. SCA-4B Water System

B. T. = BASE TANK
 T. T. = TOP TANK
 F. F. = FAST FILL
 S. F. = SLOW FILL

N O = NORMALLY OPEN
 N C = NORMALLY CLOSED
 ☒ = VALVE

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Figure 5. Schematic Diagram of Water System

The base tank fast fill rate was used when filling the region between the bottom of the base tank and the bottom of the core. The slow fill rate was used when filling the remainder of the base tank. The top tank slow fill rate was used when filling the region between the bottom of the top tank and the top of the core and the fast fill rate was used for filling the remainder of the top tank. The fill rates were adjusted to limit the maximum rate of reactivity insertion to $2\text{¢}/\text{sec}$ under all conditions.

The two sight glasses (see Figure 3) permitted remote TV observation of water levels in the top and base tanks. In addition, water detectors were used to interlock various fill rates and to indicate full tank conditions. However, the overflow lines were sized so that even if filling were continued after a full condition was indicated the level could not rise above the overflow pipes.

The aspirator was provided with water flow from two pumps in parallel. The action of the flow created a vacuum in the aspirator which in turn removed continuously any water that might enter the core. An added precaution in the dry core mode of operation (for SNAP 10A/2 only) was the placement of styrofoam inserts in the interstices between fuel rods to preclude water. The aspirator and inserts were not used in the internally-flooded-core mode.

3. Operation Sequence and Interlock System

A sequence of operations and a system of interlocks were established to allow safe and efficient utilization of the assembly. The sequence of operations is illustrated in Figure 6. Initially, the assembly is dry with the base tank in the down position. The base tank is then raised and filled at the rates previously discussed with the core vessel filled simultaneously if so required. The final step is filling the top tank to achieve criticality or a full tank, as the case may be.

The interlock system was designed to ensure the proper sequence of operations. The various interlocks are summarized as follows:

a. Base Drive-up Interlocks

- 1) The top tank must be empty.
- 2) The top tank fill valves must be closed.
- 3) The base tank dump valve must be open.

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Figure 6. Sequence of Critical Assembly Operations

- 4) The base tank fill valves must be closed.
- 5) The base tank must be empty.

b. Base Tank Fast Fill Interlocks

- 1) The base tank must be in the up position.
- 2) The top tank must be empty.
- 3) The top tank fill valve must be closed.
- 4) The water level must be more than 1/2 in. below the bottom of the core vessel.
- 5) The base tank slow fill valve must be closed.

c. Base Tank Slow Fill Interlocks

- 1) The base must be in the up position.
- 2) The top tank must be empty.
- 3) The top tank fill valves must be closed.
- 4) The base tank fast fill valve must be closed.

d. Top Tank Fill Interlocks

- 1) The base tank must be in the up position.
- 2) The base tank must be full.
- 3) Only one fill valve (either fast or slow) can be used at one time.

B. REACTOR CORES

1. SNAP 10A/2 Core

Diagrams of the SNAP 10A/2-type core are shown in Figure 7. A full core loading contained 37 cylindrical fuel-moderator elements. The rods were clad with Hastelloy-N, with end caps and grid plate indexing pins at each end. The fuel-moderator was the γ -phase hydride of Zr + 10 wt % of 93.2% enriched uranium alloy. The fuel-moderator region had a volume of 520 in.³ (8540 cm³).

The fuel elements were arranged on 1.260-in. centers in a triangular array. The overall configuration was a right cylinder with a hexagonal cross section that was 7.80 in. across the flats and 8.81 in. across the corners by 12.225 in. high. The fuel elements were supported and positioned laterally by top and bottom grid plates. Beryllium inserts were placed between the core vessel and the fuel-moderator region.

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Figure 7. SNAP 10A/2 Core Configuration

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During the series of experiments with internal flooding of the core, the interstices were filled with water. During the series with the core voided, the interstices were filled with styrofoam spacers. This material, having a density less than 2 lb/ft³, had a calculated total reactivity effect over the entire core of about 10¢ with respect to void.

2. SNAP 8 Core

Some details of the SNAP 8 fuel element and core assembly are shown in Figure 8. The fuel elements are right-circular cylinders of ZrH-10% U hydrided to an N_H of 6.0×10^{22} atoms/cc and clad with Hastelloy-N tubing and end plugs. The fuel rods are 0.532 in. in diameter and 14 in. long. The Hastelloy-N tubing is 0.540 in. ID and 14.47 in. long, with a wall thickness of 0.010 in. The inside of the tubing is coated with a ceramic hydrogen barrier material. Fuel element end plugs, 0.532 in. in diameter and 0.35 in. thick, are inserted into the tubing and welded. The finished fuel elements are 0.560 in. in diameter and 14.47 in. long (excluding the 9/64-in.-diameter, 1/2-in.-long positioning pins located at each end).

The reactor core is composed of 211 fuel elements, in triangular arrays on 0.570-in. centers, and 18 internal reflector pieces. The outline of the fuel element complex is a regular hexagon with slightly chopped corners. This can be seen in Figure 9 which shows the core lattice numbering. Internal reflectors occupy the space on the periphery of the core between the hexagonal fuel element pattern and the cylindrical reactor vessel. These internal reflectors are made of BeO hot-pressed to 98% of theoretical density and clad with 0.010-in.-thick Hastelloy-N. They extend the full length of the core between the top and bottom grid plates.

C. NUCLEAR INSTRUMENTATION

A block diagram of the nuclear instrumentation is shown in Figure 10. The instrumentation is basically of the following three types:

- 1) Two startup channels, consisting of two fission chambers connected through preamplifiers to log count rate meters and period amplifiers
- 2) An intermediate range channel consisting of an ion chamber whose output is used to activate a period scram

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Figure 8. SNAP 8 Fuel Element and Core Details

Figure 9. SNAP 8 Core
Lattice Numbering

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- 3) Two level scram channels which consist of ion chambers and linear micro-microammeters.

Each of those channels has trip units which provide a normal bias of 12 volts to the logic unit chassis under normal conditions which fall to zero in the trip condition. Any zero in the logic chassis, except in channels 1 and 2 in which case both must be zero, causes the magnet supply to be turned off.

Experimental results are obtained from the dual purpose safety and operations system shown in the block diagram. Approaches to critical are monitored by inverse multiplication plots, using data from the two fission counter channels of Figure 10 (channels 1 and 2 include scalers and recorders). Supercritical reactivities can be measured by recording successive readings of the current from an ion chamber. The latter is accomplished by using the indicated electrometer and the analog-to-digital converter which converts dc voltage into variable frequency ac. The frequency is measured with the electronic counter and the output information is provided by the printer.

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Figure 10. Safety System and Operations Instrumentation Block Diagram

The locations of the fission counters and ion chambers are shown in Figure 11. The ion chambers of channels 3, 4, and 5 are clearly visible while that for period measurements is partially hidden. The channels 1 and 2 fission counters are hidden by the frame.

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Figure 11. Fission Counter and Ion Chamber Locations

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III. CRITICAL ASSEMBLIES OF SNAP 10A/2 AND SNAP 8 CORES

A. WATER-REFLECTED CORES

1. SNAP 10A/2 Configurations

An assembly, with an unshimmed beryllium reflector attached, which was water-reflected (dry core) achieved criticality with a core loading of 26 fuel elements and a water level 9.5 in. above the bottom of the fuel. The remaining 11 lattice positions were filled with dummy lucite rods. The configuration is shown in Figure 12. The shape of the reflector and its location relative to the core is illustrated at the top. The SCA-4 fuel element numbers and their lattice locations are shown below. A description of this fuel is presented in Appendix A.

The SNAPTRAN 10A/2 -3 (dry core, without a beryllium reflector) became critical with 33 fuel elements, 4 lucite rods, and a water level of 14.94 in. above the bottom of the fuel. It was possible to achieve a full core loading by decreasing the amount of water added to the top tank. The various critical loadings are presented in Table 1. The lattice locations of the fuel elements are shown in Figure 13 and a description of the fuel is presented in Appendix A. The lattice locations were filled in order, so that a 33-element loading means that the first 33 lattice positions were filled. No styrofoam inserts were placed between fuel elements in this particular experiment.

2. SNAP 8 Configuration

The fully loaded, bare, SNAP 8 dry core proved to be subcritical when fully water-reflected. The extrapolated critical loading was no less than 260 ± 10 fuel elements. Based upon this value and an estimated average value of 10¢ each for the remaining 49 elements, the subcritical reactivity for the full core was found to be $\$4.90 \pm 1.00$.

A pulsed neutron decay measurement, obtained by the technique described in Appendix C, gave a subcritical reactivity of $\$3.22 \pm 1.00$. The large uncertainty in these values is due to the assembly being far subcritical. However, they are in agreement within the estimated errors.

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Figure 12. SCA-4, 26-Element Loading with Beryllium Reflector

TABLE 1

SNAPTRAN 10A/2 -3 CRITICAL LOADINGS AT VARIOUS
WATER LEVELS IN TOP TANK

Number of Fuel Elements	Number of Lucite Rods	Water Level Above the Bottom of the Fuel at Critical (in.)
33	4	14.94
34	3	12.06
35	2	11.06
36	1	10.31
37	0	9.88

Figure 13. SNAPTRAN 10A/2 -3
Full Core Loading

(X) LATTICE LOCATION
X FUEL ELEMENT NUMBER

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B. WATER-IMMERSED-AND-REFLECTED CORES

1. SNAP 10A/2 Configuration

The SCA-4 fuel was loaded to critical in the internally flooded and reflected mode to determine the reproducibility with Phase I test results. Criticality

was achieved with the 25-element loading shown in Figure 14 and 1.88 in. of axial (top) reflector. The worth of water reflector was found to have the following values:

Axial Water Reflector Height (in.)	Reactivity (ρ)
1.88	0
2.00	+ 4.9
2.25	+ 10.5
2.50	+ 15.5

The critical loading of the Phase I experiment was 25 fuel elements with 1.75 in. of axial water reflector. The present configuration is approximately 22¢ less reactive (based on estimates of the total excess reactivities). The agreement is good, considering that critical machine modifications were made between the two experiments, e.g., elimination of the control cap.

Figure 14. SCA-4, 25-Element Loading

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2. SNAP 8 Configuration

The SNAP 8 core, with beryllium-oxide inserts, internally flooded and water-reflected, was critical with the 173-element loading shown in Figure 15. The 38 lattice positions in the cross-hatched area were filled with lucite rods. There was a small excess reactivity of 8.5¢ with full water reflection.

C. DETERMINATION OF TOTAL EXCESS REACTIVITY OF A FULLY LOADED CORE

Loading to full core from a critical configuration was accomplished by compensating the excess reactivity with poison splines. The splines were right-circular cylinders which fitted the interstices between fuel elements. The SNAP 10A/2 splines were filled with boron powder and those for SNAP 8 contained a mixture of samarium, gadolinium, and other rare earth oxides. The following assemblies were loaded to full cores and the excess reactivities were determined.

1. SCA-4 Fuel Loading

The SCA-4 fuel was loaded to full, dry core, with 12 poison splines located in the positions shown in Figure 16. The loading began with a 32-element

Figure 15. SNAP 8 173-Element Loading

Figure 16. SCA-4 Fuel - Full Core Loading

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core and progressed with additions of single fuel elements. A sufficient number of splines to compensate for the excess reactivity was inserted after each fuel element addition. The calibration curve shown in Figure 17 was determined from period-measurement-versus-water-level data obtained during the stepwise procedure. The fully reflected condition is used as the reactivity reference and the bottom of the fuel is used as the water level reference.

The worth of each of the five added fuel elements is shown in the following table: (The average worth of each element is approximately +77¢. The 32-element loading with full top tank had an excess reactivity of 41¢.)

Fuel Element Number	Reactivity Worth (¢)
84	+ 73
85	+ 65
86	+ 86
98	+ 86
116	+ 76

Boron poison spline worths are given in Table 2. The available excess reactivity in the top tank with 37 fuel rods and 12 poison splines was \$1.77. The average worth of each spline was approximately -21¢.

The cumulative excess reactivity, obtained by summing fuel element worths and the 32-element available excess, was found to be $\$4.27 \pm 0.10$. The estimated error involved in each individual measurement is $\pm 2\text{¢}$.

A somewhat greater cumulative excess reactivity, $\$6.4 \pm 0.6$, was previously measured^{1*} for the identical fuel loading. However, the assembly configuration was not precisely identical, with differences in top cover plate and the bottom of the core vessel. The measured worth of these differences was found to be 38¢. Therefore, the remainder of the difference in results must be

ascribed to the experimental methods used. A pulsed neutron technique measuring subcritical reactivity was used previously. It is felt that the present method of poison addition and period measurement gives more reliable results.

2. SNAPTRAN 10A/2-3 Fuel Loading

This core loading started with the 33-element dry core previously discussed. The remaining four fuel elements were added one at a time and the excess reactivity was compensated by the addition of poison splines in the

Figure 17. SCA-4 Fuel - Top Reflector Calibration

*Superscripts refer to the References listed at the end of the report.

TABLE 2
POISON SPLINE WORTHS IN SCA-4 FUEL LOADING

Boron Poison Spline Number	Reactivity Worth (¢)	Boron Weight (gm)
31	- 19	1.40
32	- 17	1.42
27	- 13	1.60
21,23	- 49	1.11,1.18
22,29	- 43	0.99,1.38
33	- 21	1.21
28	- 18	1.72
26	- 21	1.66
30	- 26	1.44
24	- 22	1.58

positions shown in Figure 18. A total of 15 splines were added to achieve full loading. The water reflector curve is shown in Figure 19 and was used to determine the fuel element worths as follows:

Fuel Element Number	Reactivity Worth (¢)
19	75.7
83	84.3
84	84.5
85	61.4

The average worth of the fifteen poison splines is -22.1¢ per spline. Their boron content is shown in Table 3.

The full core excess reactivity was obtained by summing the fuel element worths and the excess of 46.4¢ available with the 33-element loading. It was found to be $\$3.52 \pm 0.25$. The SNAPTRAN core was therefore approximately 75¢ less reactive than the SCA-4 core.

Radial, axial, and tangential relative-flux mappings were obtained for the full core loading using highly enriched U-235 foils. The foils were attached to the outside of the fuel elements at locations shown in Figure 18. The radial

Figure 18. SNAPTRAN 10A/2-3
Full Core Loading with Poison
Splines

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Figure 19. SNAPTRAN 10A/2-3
Fuel - Top Reflector Calibration

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TABLE 3
POISON SPLINE BORON CONTENT IN
SNAPTRAN 10A/2-3 FUEL LOADING

Poison Spline Number	Boron Weight (gm)
21	1.11
22	0.99
23	1.18
24	1.58
25	1.63
26	1.66
27	1.60
28	1.72
29	1.38
30	1.44
31	1.40
32	1.42
33	1.21
34	1.70
35	1.62

mappings are shown in Figure 20. Although the two curves were obtained for different conditions (as stipulated), they are practically identical throughout both core and reflector regions.

An axial mapping along the core centerline is shown in Figure 21. The point representing the lowest foil is somewhat high. The foil may have slipped partially below the fuel into a region of higher thermal flux.

A partial tangential mapping is shown in Figure 22. It can be seen that the flux is higher at the corners of the hexagon than on the flat.

The flux-traverse data are useful in correlating analytical results with the experiments.

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Figure 20. SNAPTRAN 10A/2-3 Fuel - Radial Flux Mapping

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Figure 21. SNAPTRAN 10A/2-3 Fuel - Axial Flux Mapping

3. SNAP 8 Fuel

The loading to full core began with the 173-element critical loading previously discussed. The core was fully water-reflected and internally flooded. Fuel elements and poison splines were added in reactivity steps of less than \$1.00. The full core and poison spline loading are shown in Figure 23. The cumulative excess reactivity was found to be $\$5.87 \pm 0.32$ with a full core loading of 211 fuel rods. At this point, there were also 72 poison splines in the core and the available excess reactivity with full axial water reflector was 18.6¢. It follows that the 72 splines were worth a total of -\$5.68 for an average of -7.9¢ per spline. The top water reflector was calibrated over a range of approximately

Figure 22. SNAPTRAN 10A/2-3
Fuel - Tangential Flux Mapping

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Figure 23. SNAP 8 Full Core
Element and Poison Spline
Loading

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80¢ during the loading of elements numbers 174 through 177. Additional calibration to over \$1.00 was done during poison spline additions. This calibration curve is shown in Figure 24. Also shown is the calibration curve obtained beginning with full fuel loading and 72 poison splines. A comparison of the two curves indicates the reproducibility of the calibration method.

Radial flux data were obtained using highly enriched U-235 foils for: (1) 174 fuel elements and 37 lucite rods and (2) all fuel elements and 72 poison splines (foil positions are shown in Figure 23). The mappings are shown in Figure 25. The flux is observed to have a peak in the water immediately adjacent to the core. This is due primarily to high thermalization and also low absorption in this region compared to that in the core. It can be seen that the presence of lucite in the core has a similar effect. The flux peak in the fuel region can be seen to shift with the core center.

A circumferential flux mapping for the 174-fuel-element and 37-lucite-rod loading is shown in Figure 26. Again, the flux peaking in the lucite region is evident.

Figure 24. SNAP 8 Top Reflector Calibration

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Figure 25. SNAP 8 Median Plane Radial Flux Mapping

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Figure 26. SNAP 8 Median Plane Circumferential Flux Mapping

IV. EFFECT OF POISON AND VOID SLEEVES

A. POISON SLEEVES

1. Water Reflection Only

This set of experiments was performed to provide data for the SNAP-TRAN Program. Figure 27 shows the geometric configurations in the SCA-4B experiment and the SNAPTRAN-3 destruct core experiment. The fuel elements, beryllium internal reflectors, and grid plates are identical in both configurations. These components are coded to permit exact reassembly in the SNAP-TRAN-3 destruct core vessel. The radial core vessel in both configurations is made of 0.031-in.-thick 316 stainless steel.

The SNAP 10A/2 design vessel head is used in both configurations. The SCA-4B head is the unmachined forging and does not incorporate the NaK outlet pipe. The SNAPTRAN-3 head will incorporate NaK filling and pressure transducer apparatus in the NaK outlet area. This difference is expected to have a minor effect on excess reactivity.

The main differences in configuration occur in the axial regions. The "bottom" axial water is slightly closer to the fuel region in the SCA-4B configuration. It is estimated that the 0.25-in. aluminum plate in SCA-4B (see Figure 27) does not significantly affect the reactivity. The "top" water reflection is 0.28 in. further away from the fuel region in SCA-4B. This additional distance comprises a purely void region and probably represents the greatest reactivity difference between the two configurations. The minimum axial water depth of 3.5 in. in this region is near infinite in reflectivity effects.

Another difference, expected to result in minor reactivity effects, is that the destruct core will be filled with NaK whereas the SCA-4B core was voided.

The full core loading previously discussed and shown in Figure 13 was used throughout this set of experiments. Typical Binal sleeves are shown in Figure 28. They were formed in half-sections to facilitate mounting. With one layer of 1/4-in.-thick Binal (10.2 w/o boron in aluminum) sleeves spaced 1/4 in. from the core vessel and completely surrounding the radial core surface, the assembly was far subcritical when fully water-reflected (water between sleeve

SNAPTRAN - 3
DESTRUCT CONFIGURATION

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SNAPTRAN-3
SCA-4B CONFIGURATION

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Figure 27. SNAPTRAN-3 Configurations

Figure 28. Typical Binal
Sleeves

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and core). With the lower two-thirds of the core surrounded by Binal sleeve, the assembly was still subcritical.

The lower Binal sleeve was then removed and the upper sleeve was placed at the bottom of the core, 1/4 in. away from the core vessel, surrounding the lower 2.7 in. of the fuel. In this configuration, the assembly was critical with a water level of 10.4 in. above the bottom of the fuel. Extrapolating the calibration curve (Figure 19) to this value gives an excess reactivity of approximately \$1.70. Subtracting this value from the full core excess of \$3.52 gives -\$1.80 (approximately) as the reactivity worth of the Binal sleeve surrounding the bottom 2.7 in. of the fuel.

In order to determine a sleeve configuration with a negative reactivity worth equivalent to the full core excess, the lower Binal sleeves were placed at various elevations on the core vessel to simulate sleeves of different height. The sleeve was held 1/4 in. from the core vessel in all cases. A summary of the results is given in Table 4.

TABLE 4
DETERMINATION OF BINAL SLEEVE EFFECTIVENESS

Lower Fuel Element Length Surrounded with Sleeve (in.)	Water Level Above Bottom of Fuel (in.)	Excess Reactivity (ρ)
6.50	18.25	subcritical
4.50	14.13	0 (critical)
	14.75	25.7
4.93	18.25	0 (critical)

The third configuration shows that a Binal sleeve surrounding the bottom 4.93 in. of the fuel element length and spaced 1/4 in. from the core vessel is capable of maintaining a fully water-reflected core just critical.

The kinetic parameter, β/ℓ^* , was measured for the foregoing configuration by means of a noise analysis technique discussed in Appendix D. This parameter will be useful in predicting the dynamic behavior of the core. It was necessary to lower the Binal sleeve 1/2 in. to compensate for the reactivity loss due to displacement of top tank water by the scintillation detectors used in this experiment. The results are shown in Table 5.

TABLE 5
SUMMARY OF β/ℓ^* MEASUREMENTS

Run No.	Water Level Above Bottom of Fuel (in.)	$\beta/\ell^* (\text{sec}^{-1})$	$\frac{\beta - \rho}{\ell^*} (\text{sec}^{-1})$	$\left(\frac{1}{\ell_r}\right)^\dagger (\text{sec}^{-1})$
1 (critical)	16.00	584.7	—	7,215
2 (subcritical)	14.13	—	880	21,396

$^\dagger \ell_r$ = reflected neutron lifetime (sec)

The foregoing results are thought to be quite accurate for the following reasons:

- a) Source effects were minimized by placing the source centrally on the bottom of the base tank while scintillation probes were located in the top tank.
- b) Low-frequency and high-frequency spectral components were recorded on separate tapes.
- c) The reactor power level drifted only 3% over the entire critical run (No. 1).

It should also be noted that the subcritical reactivity, as determined by noise analysis,

$$\frac{\rho}{\beta} \approx 1 - \left(\frac{\beta - \rho}{\ell^*} / \frac{\beta}{\ell^*} \right) \approx -50\text{¢},$$

agrees with the value of -38¢ obtained from Figure 19, which is for full water reflection and no Binal sleeve. It is expected that correction for the Binal sleeve will slightly increase this value.

These data indicate that a Binal sleeve is capable of controlling the full core excess of a completely water-reflected (dry) core and thereby established the nuclear feasibility of using such a sleeve to initiate the destruct transient of SNAPTRAN-3. Additionally, the design criteria of the sleeve were determined, as well as the kinetic parameter β/ℓ^* .

2. Water Immersion and Reflection

a. SCA-4 Fuel Loading

The critical SCA-4 core loading with internal flooding and 3/4-in. - thick Binal sleeves was determined. These sleeves covered the entire core radial surface and fitted against the core vessel wall so that no water gap existed on immersion. The approach to critical began with a loading of 33 fuel elements and 4 lucite rods. The critical condition was achieved with 35 fuel elements and 2 lucite rods with 10.25 in. of water above the bottom of the fuel. The same configuration was found to be critical during the Phase II experiments,¹

except that 13.88 in. of water above the bottom of the fuel were required. This difference in water level cannot be evaluated accurately because the top reflectors were not calibrated with wet cores and the core vessel cover plates were not identical.

b. SNAPTRAN 10A/2-3 Fuel Loading

The internally flooded SNAPTRAN 10A/2-3 core was loaded to critical with both 1/4-in.-thick and 3/4-in.-thick Binal sleeves. The sleeves were close fitting and surrounded the entire core vessel radial surface. With the 1/4-in.-thick sleeves, the critical loading was achieved with 36 fuel elements and 1 lucite rod. The water level above the bottom of the fuel was 10.5 in. With the 3/4-in.-thick sleeves, the core loading was the same, but the water level was 9.13 in. Therefore, the thicker sleeves were less effective by the reactivity worth equivalent to 1.38 in. of axial water reflector. This added reactivity may have been due partly to a poorer fit, allowing water between vessel and sleeve. Another effect indicated is that the additional 1/2 in. of Binal was reflecting neutrons of sufficient energy to pass through the 1/4-in. thickness of Binal back into the core. However, an equivalent thickness of water thermalizes the neutrons so that they are absorbed in the 1/4-in. thickness of Binal on their passage back to the core. An optimum poison thickness is therefore indicated. An opposite effect was observed in Phases I and II, due probably to the fit of the sleeves.

B. VOID REGIONS SURROUNDING THE CORE

Void sleeve studies were performed with the SNAP 8 core only. The core was water-reflected and internally flooded, with the beryllium oxide inserts in the core, in all cases. A typical sleeve configuration is shown in Figure 29.

A loading approach to critical was performed starting with 173 fuel elements and 38 lucite rods in the core and 1-in. void sleeves surrounding the entire core vessel radial surface. A typical sleeve is shown in Figure 30. Criticality was achieved with 192 fuel elements and 19 lucite rods, and the excess reactivity was found to be 6.9¢ with full water reflection.

The top reflector calibration curve of Figure 31 was obtained by adding fuel elements to the foregoing configuration. A total of 7 fuel elements was added during the course of the calibration. The high point at 13.8 in. was caused by

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Figure 29. Typical Void Sleeve Configuration Installed Around
SNAP 8 Core

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Figure 30. Typical 1-Inch
Void Sleeve

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Figure 31. SNAP 8 Top Reflector
Calibration with Void Sleeves
Installed

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retention of water on the core vessel cover plate in that measurement. Comparison of this curve with those in Figure 24 indicates that the addition of the 1-in. void sleeves produced no change in calibration beyond experimental errors.

Loading to full core was accomplished by means of the poison spline method with a total of 29 splines added in the positions shown in Figure 32. The cumulative excess reactivity was determined to be \$2.07. The average worth of poison splines was found to be 6.4¢ per spline.

A flux mapping was obtained with the 1-in. void sleeves in place, 211 fuel elements, and 29 poison splines. The foil locations are shown in Figure 32. The results are shown in Figure 33. The two flux dips near the centerline were caused by the proximity of poison splines to the U-235 foils. This curve and the one in Figure 25 are almost the same when normalized to the same value, indicating that the void sleeves had negligible effect on the core flux distribution.

When the 1-in. void sleeves were replaced with 3-in. void sleeves (see Figure 34), the full core loading of 211 fuel elements (no poison splines) was found to be subcritical. The average value of extrapolated critical loading obtained from four neutron detector channels was 221 fuel elements.

The subcritical reactivities for 3-in. and 4-in. void sleeves (see Figure 35) were measured by using noise analysis and pulsed neutron techniques, which are discussed in Appendixes D and C, respectively. Noise analysis yields the parameter β/ℓ^* , which is used in the pulsed neutron decay equation to calculate the subcritical reactivity.

The results of the noise analysis are presented in Table 6. All values are for full water reflection and internal flooding.

The measured excess reactivities for the various configurations are summarized in Table 7.

The β/ℓ^* value of 520 sec^{-1} measured at critical was corrected for subcriticality by 3%/\$ below critical. This correction was based on analytical studies of reactivity on β/ℓ^* .

The tabulated data are shown graphically in Figure 36. Extrapolation of this curve to large void thickness (about 10 ft) should give a subcritical reactivity

Figure 32. SNAP 8 Poison Spline
and Foil Location

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Figure 33. SNAP 8 Radial Flux
Mapping

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Figure 34. Typical 3-Inch
Void Sleeve

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Figure 35. Typical 4-Inch
Void Sleeve

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TABLE 6

SUMMARY OF NOISE ANALYSIS MEASUREMENTS FOR VARIOUS
VOID SLEEVE CONFIGURATIONS

Number of Fuel Elements	Number of Poison Splines	Void Sleeve	$\frac{1}{\alpha}$ (sec)	$\beta / \ell^* (\text{sec}^{-1})^\dagger$
176	none	none	—	773(±5%)
211	72	none	—	523(±5%)
211	29	1-in.	—	520(±5%)
211	none	3-in.	585 ± 60	551 [§]
211	none	4-in.	394 ± 24	581 [§]

$\dagger \ell^* = \ell_c + \beta_7 \ell_r$, where ℓ_c is the core neutron lifetime, ℓ_r is the reflected neutron lifetime, and β_7 is the fraction of seventh-group delayed neutrons.

[§]These values are corrected for subcriticality.

TABLE 7

SUMMARY OF MEASURED EXCESS REACTIVITIES FOR VARIOUS
VOID SLEEVE CONFIGURATIONS

Number of Fuel Elements	Number of Poison Splines	Void Sleeve	Excess Reactivity (\$)
211	72	none	$+5.87 \pm 0.30$
211	29	1-in.	$+2.07 \pm 0.20$
211	none	3-in.	-2.10 ± 0.45
211	none	4-in.	-3.37 ± 0.53

value for a full, flooded core with no radial reflector. The extrapolation of the lower three points is shown in Figure 37. It can be seen that they fall on a straight line in semilog coordinates and extrapolate to a value of $-\$16$ at 100 in. The measured value using the foregoing procedure was found to be $-\$13 \pm 3$. The method of correcting β / ℓ^* for subcriticality introduces large errors far below critical.

Figure 36. SNAP 8 Void Sleeve Reactivity

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C. COMBINATION VOID-POISON SLEEVE STUDIES

1. SNAP 8 Fuel Loading

The poison sleeves studied in Phase I¹ were not adequate in maintaining subcriticality for a fully water-reflected-and-flooded core. The results of the void sleeve studies presented in the foregoing indicated the promise of a combination void-poison sleeve. Consequently, the void study was extended to include one configuration with a 3-in. void sleeve surrounding the core vessel and a 1-in. boron poison sleeve surrounding the void sleeve. The core contained 211 fuel elements and no poison splines. Based on the measured value of $1/\alpha$ of $176 \pm 13 \mu\text{sec}$ and the measured value of β/ℓ^* of 520 sec^{-1} at critical, values of β/ℓ^* and reactivity for this configuration were calculated to be $675 \mu\text{sec}$ and $-\$7.45 \pm 1.10$, respectively.

2. SNAP 10A/2 Fuel Loadings

a. SCA-4 Fuel

Void-poison sleeve studies were also done using the SCA-4 fuel. The purpose was to determine the effectiveness of this combination sleeve and provide criteria for its design. In practice, the sleeve assembly would be protected

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Figure 37. SNAP 8 Void Sleeve Extrapolation

from puncture and minor distortion by a steel cylinder with poisoned end plates. This container was simulated and the component reactivity worths measured during this series of tests.

With the 3-in. void sleeve surrounded by a 1-in., boron-filled sleeve (see Figure 38), full water reflection, and internal flooding, the assembly became critical with a core loading of 36 fuel elements and 1 lucite rod. The water level above the bottom of the fuel was 14.88 in. In this configuration, the worth of cadmium sheet over the ends of the void sleeves was found to be approximately -15 to -30¢. Similarly, the worth of the 1/4-in.-thick steel sleeve surrounding the poison sleeve was found to be approximately +15 to +30¢.

Figure 38. Typical 3-Inch
Void Sleeve Surrounded by
a 1-Inch, Boron-Filled
Sleeve

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The 37th element was loaded into the resulting configuration with the cadmium sheet removed (void, boron, and steel sleeves attached) and criticality was achieved with 12.41 in. of water above the bottom of the fuel.

The SCA-4 fuel was known to have an excess reactivity exceeding the flight system requirements. Therefore, to demonstrate the effectiveness of the simulated sleeve, the excess reactivity was reduced by the substitution of special fuel elements with lower hydrogen content. These special fuel rods are described in Appendix A. The lattice locations and special element numbers are shown in Figure 39. The remaining locations were filled with SCA-4 fuel elements as shown in Figure 16. In these four elements, the average hydrogen concentration was reduced from 6.44 to 5.97×10^{22} atoms/cc. With this change, the assembly was just critical with full water reflection and flooding. The loading of a fifth special element into location number 21 resulted in a subcritical configuration. These results were instrumental in establishing the following shipping sleeve design criteria:

Figure 39. Special Low Hydrogen
Fuel Elements

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- 1) The construction material should be 304 stainless steel.
- 2) The first layer should be a minimum 3-in. simulated void containing less than 2.75% nonvoid.
- 3) The second layer should consist of 1-in.-thick boron powder packed to 20 to 22% of theoretical density.
- 4) The third layer should be less than 1/2 in. of steel.
- 5) Axial ends of voids are to be covered with a 1/4-in.-thick layer of boron powder followed by 1/4 in. of steel.
- 6) The boron powder container and the surface of the simulated void next to the core vessel shall have a thickness less than 0.031 in.
- 7) The sleeve shall extend the total length of the fuel rods.
- 8) The maximum allowable circumference of the core not covered by the sleeve is to be 1/4 in.

b. SNAP 10 FR-1 Fuel

A shipping sleeve, designed and fabricated according to the experimentally established criteria, was tested using flight system fuel, designated as S10FR-1, which is described in Appendix A. The main purpose of this test was the demonstration of sleeve effectiveness in maintaining a fully water-reflected-and-flooded core subcritical. A cutaway diagram of this sleeve is shown in Figure 40. The void region is compartmented to prevent complete flooding of the void in the event the external walls are punctured during a water immersion incident. The shipping sleeve tested is shown in Figure 41. The core vessel is entirely below the top tank and completely surrounded by the shipping sleeve (see Appendix B).

With the shipping sleeve in place, the full core loading shown in Figure 42 (without the poison splines) was attained with the assembly remaining subcritical with full water reflection and flooding. The average value of extrapolated critical loading obtained from inverse multiplication plots of data from three neutron detector channels was 41.8 ± 1.0 fuel elements. Using a value of 55¢ per rod determined from another test, the subcritical reactivity was determined to be $\$2.64 \pm 0.55$.

In order to check this value, two pulsed neutron runs were made, using the procedure described in Appendix C. The first run was made with the foregoing configuration without poison splines, and the second was made with the seven poison splines in the interstitial positions shown in Figure 42. The worth of these poison splines, in the same locations, had previously been determined to be $-\$1.41$ (see Table 2). The data obtained in these runs are as follows:

<u>Run Number</u>	<u>Decay Constant (μsec)</u>	<u>Reactivity (\$)</u>
1	217	R
2	160	R-1.41

The statistical error in each of the runs was not more than 3% in the initial analyzer channels. Assuming that β/ℓ^* in run no. 2 did not change appreciably from that in run no. 1, its value was calculated to be $1160 \pm 120 \text{ sec}^{-1}$. This agrees well with the $1100 \pm 300 \text{ sec}^{-1}$ obtained in Reference 1 for a flooded-core,

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Figure 40. SNAP 10A Reactor Shipping Sleeve Schematic Diagram

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Figure 41. SNAP 10A Reactor Shipping Sleeve Installed on
Demonstration Reactor

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Figure 42. SNAP 10 FR-1
Fuel - Full Core Loading

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Binal-sleeved configuration. Using the β/ℓ^* of 1160 sec^{-1} and the decay constant of $217 \mu\text{sec}$, the subcritical reactivity, R , of the unpoisoned core was calculated to be $\$3.00 \pm 0.50$. This value agrees with the extrapolated value of $\$2.64 \pm 0.55$ within experimental uncertainties. These results proved the adequacy of the shipping sleeve concept.

D. VOID FILLER BLOCK STUDIES WITH BERYLLIUM REFLECTOR

1. SCA-4 Fuel Loading

Initial studies utilizing SCA-4 fuel were made to establish the design criteria for the filler block concept. The fact that the SCA-4 core had a larger excess reactivity than required in the flight system core was taken into consideration. No reactivities were measured because only a subcritical 37-fuel-element configuration was of interest. Hence, modifications were made to the assembly until this was accomplished.

The unshimmed beryllium reflector and boron filler block assembly used in this experiment are shown schematically in Figure 43. The 1/2-in. thickness

Figure 43. SNAP 10A/2
Reflector and Filler
Block Assembly

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of boron powder was encased in stainless steel with the poison sections covering all four drum voids completely and two of the reflector surfaces partially. The control drums were attached to the reflector for this experiment.

With the foregoing assembly attached, the critical loading with water reflection and dry core was 33 fuel elements and 4 lucite rods and 11.75 in. of water above the bottom of the fuel. Adding the 34th fuel element decreased the critical water height to 9.5 in. Comparison of this critical configuration to that shown in Figure 12 (beryllium reflector, no filler block) which also required 9.5 in. of water, indicates the worth of the boron filler block to be equivalent to that of eight fuel elements (positions 27 through 34). The estimated worth of these fuel elements is approximately \$6.00.

The following modifications to the 34 fuel elements and 3 lucite rods configuration were required to achieve a 37-fuel-element subcritical loading:

- a) Simulation of flight plenum chamber
- b) Substitution of ten special low-hydrogen-content (low N_H) fuel elements into the central core region

- c) Placing styrofoam in the remaining drum voids and behind the poison filler blocks on the reflector faces
- d) Styrofoam mockup of the lower plenum and the neutron shield to exclude water from that volume.

The upper flight plenum chamber was simulated by replacing the top cover plate with a plenum chamber. A sketch of the styrofoam mockups is shown in Figure 44. The locations and element numbers of the special fuel elements are shown in Figure 45. No attempt was made to incorporate the reactivity worth of the shield itself, because it is assumed that final reflector shimming and, thus, system reactivity will be determined with the shield in place. In the event of submergence, water will be introduced everywhere except in the shield. The experiments, therefore, mock up the same amount of water addition.

The estimated critical loadings with full water reflection and approximate incremental loadings for the various modifications are summarized in Table 8. These results were used in establishing the following filler block design criteria:

Figure 44. SNAP 10A/2
Styrofoam Mockups

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Figure 45. SNAP 10A/2 Special Fuel Elements

(X) LATTICE LOCATION
X FUEL ELEMENT NUMBER

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TABLE 8

SUMMARY OF CRITICAL AND INCREMENTAL SCA-4
FUEL LOADINGS FOR VARIOUS CONFIGURATIONS

Modification	Critical Loading	Incremental Loading
Filler blocks	33	—
Plenum chamber	33 +	1/2
10 low N _H elements	36	3
Styrofoam voids	36-1/2	1/2
Styrofoam mockups	37 +	> 1/2

- Filler blocks shall cover at least the reflector area outlined in Figure 43.
- Nuclear poison to be 1/2-in. thickness of boron powder packed to at least 20% theoretical density

- c) Void portion of filler block shall be closer to the core, with poison extending inside the drum voids.
- d) The two remaining reflector faces to be covered with at least 1-1/2 in. of void followed by 1/2 in. of boron powder
- e) Reduce filler block structural material to permit greater area to be covered with boron powder.

2. SNAP 10FR-1 Fuel Loading

The purpose of this experiment was to prove the effectiveness of a filler block designed and fabricated according to the experimentally established criteria.

The S10FR-1 flight system fuel and SCA-4C unshimmed beryllium reflector were used in this test. The four control drums were attached to the reflector and fixed in the full-out position. The combination void-poison filler block was attached to the reflector. A styrofoam mockup of the neutron shield was attached to the bottom of the core to exclude water from that region. This configuration is shown in Figure 46.

The initial fuel loading was 31 fuel elements and 6 lucite rods, following the loading pattern shown in Figure 47. This configuration was found to be subcritical with full water reflection and dry core. The loading to full core was performed by replacing singly the 6 lucite rods with fuel elements. The full core loading of 37 fuel elements (without poison) was achieved with the assembly remaining subcritical. The average value of extrapolated critical loading from three inverse multiplication plots was 39.4 ± 0.4 fuel elements. Using the previously determined average worth of 55¢ per element, the assembly was found to be $\$1.32 \pm 0.22$ subcritical.

This value was checked by measuring pulsed neutron decay. Two runs were made, one with the unpoisoned core and the other with three poison splines in the positions shown in Figure 47. The combined worth of these splines, as positioned, had previously been determined to be approximately -82¢ (see Table 2). The decay constant obtained from each 25-min run is shown in the following table:

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Figure 46. Combination SNAP 10A Void-Poison Filler Blocks
Installed on Demonstration Reactor

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Figure 47. SNAP 10 FR-1 Full Core Loading

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Run Number	Decay Constant, $1/\alpha$ (μ sec)	Subcritical Reactivity (β)
1	572	R
2	430	R-0.82

The estimated statistical error in each run did not exceed 2% in the initial analyzer channels. Using this data and assuming that β/ℓ^* was constant for both runs, its value was calculated to be $702 \pm 70 \text{ sec}^{-1}$. Using this value with the run number 1 decay constant, the unpoisoned core was found to have a subcritical reactivity, R, of $\$1.49 \pm 0.25$. This is in agreement with the extrapolated value.

A third pulsed neutron run was made with the 37-element loading and no water reflector. Using a β/ℓ^* of 1107 sec^{-1} from Reference 4 for a beryllium-reflected reactor and the decay constant of $104 \mu\text{sec}$ obtained in this run, the shutdown margin was calculated to be approximately \$7.70.

This successful demonstration of the filler block concept concluded the water immersion tests.

V. REFERENCES

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APPENDIX A

FUEL ELEMENT DESCRIPTION

1. SCA-4 Fuel

Thirty-seven SCA-4 fuel rods were fabricated, hydrided, and assembled into fuel elements. The assembly procedure is described here.

The fuel rods consisted of 90% zirconium - 10% uranium (93% U-235) alloy hydrided to an N_H of 6.4. The hydrided rods were machined to 1.212 in. in diameter and 12.225 in. long. In some cases, however, two pieces of rod were required to make up this length requirement. Fuel elements No. 81 through 90, 94, 95, 106, 107, 108, 112, 113, and 114 each contained rod segments 10.000 and 2.225 in. long. Fuel elements No. 89, 91, and 92 each contained segments 10.125 and 2.125 in. long.

Hastelloy-N tubing, 1.250 in. in diameter and 0.010 in. in wall thickness, was coated internally according to AI Specification NB0125-100. Both end caps were similar to the cold end caps of the SNAP 10A/2 design; they were not coated.

In order to insert the end caps in place for welding, it was necessary to machine back the coating approximately 0.060 in. from each end of the tube. After one end cap had been welded in place, the weld was leak-checked using a Veeco Helium Leak Detector. The tube was cut to length, the fuel rod and tube were cleaned with acetone, the tube was flushed with helium, and the fuel rod was inserted into the tube. The remaining end cap was placed in position and the cap was welded to the tube. The fuel element was removed, visually inspected, and again leak-checked on the Helium Leak Detector.

Each element was then machined to a length of 12.450 in.; the pins were machined to 0.225 in. diameter by 0.321 in. long. Final inspection was performed and the element identification was etched on each end. The elements were thoroughly cleaned by liquid honing.

Table 9 lists the fuel elements by their components, weights, and percentages of enrich uranium. Table 10 lists values of density and hydrogen concentration. The two values reported for wt % H in Table 10 correspond to the analysis of samples taken from each end of the rod.

TABLE 9
WEIGHTS OF FUEL RODS AND URANIUM
(Sheet 1 of 3)

Element No.	Long Rod No.	Short Rod No.	Weight (gm)	Wt % U	Wt of U	% Wt U ²³⁵	Wt of U ²³⁵	Total Wt of Rod	Total Wt of U	Total Wt of U ²³⁵
81								1398.04	138.53	129.06
	524-8		1141.16	9.91	113.09	93.15	105.34			
		519-5D	256.88	9.91	25.46	93.15	23.72			
82								1394.95	138.71	129.21
	510-3		1137.51	9.96	113.30	93.15	105.54			
		510-1A	256.44	9.91	25.41	93.15	23.67			
83								1388.56	137.72	128.41
	520-4		1131.97	9.92	112.29	93.26	104.72			
		519-5B	256.59	9.91	25.43	93.15	23.69			
84								1399.02	138.74	129.14
	519-3		1142.38	9.91	113.21	93.15	105.45			
		519-5A	256.64	9.91	25.43	93.15	23.69			
85								1398.39	138.50	129.02
	525-7		1142.41	9.91	113.11	93.15	105.37			
		516-1D	255.98	9.92	25.39	93.15	23.65			
86								1397.64	138.53	129.05
	517-2		1140.67	9.91	113.04	93.15	105.30			
		516-1B	256.97	9.92	25.49	93.15	23.75			
87								1398.02	138.70	129.32
	520-6		1142.31	9.92	113.32	93.26	105.68			
		511-1C	255.87	9.92	25.38	93.15	23.64			
88								1397.25	138.57	129.09
	511-2		1140.35	9.89	113.09	93.15	105.35			
		516-1C	256.90	9.92	25.48	93.15	23.74			
89								1397.27	138.45	128.50
	520-1		1155.47	9.82	113.47	93.21	105.76			
		511-1B	241.80	9.92	24.08	93.15	22.44			
90								1396.95	138.57	129.08
	513-2		1140.87	9.92	113.17	93.15	105.42			
		511-1D	256.08	9.92	25.40	93.15	23.66			
91								1398.33	138.72	129.22
	513-4		1142.42	9.92	113.33	93.15	105.57			

TABLE 9
WEIGHTS OF FUEL RODS AND URANIUM
(Sheet 2 of 3)

Element No.	Long Rod No.	Short Rod No.	Weight (gm)	Wt % U	Wt of U	% Wt U ²³⁵	Wt of U ²³⁵	Total Wt of Rod	Total Wt of U	Total Wt of U ²³⁵
92		512-2B	255.91	9.92	25.39	93.15	23.65			
								1399.07	139.03	129.49
	513-7		1156.60	9.94	114.97	93.15	107.09			
93		512-2D	242.47	9.92	24.05	93.15	22.40			
								1399.07	139.01	129.50
	511-5		1156.54	9.94	114.96	93.15	107.09			
94		512-2C	242.53	9.92	24.05		22.41			
								1399.66	138.85	129.41
	516-7		1143.42	9.92	113.43	93.21	105.73			
95		512-2A	256.24	9.92	25.42	93.15	23.68			
								1387.54	137.61	128.23
	516-4		1130.95	9.92	112.19	93.21	104.57			
96		519-5C	256.56	9.91	25.42	93.15	23.68			
								1390.64	138.95	128.57
	515-2		1133.95	9.92	112.49	93.21	104.85			
97		516-1A	256.69	9.92	25.46	93.15	23.72			
								1397.57	137.80	128.38
	529-3	—	1397.57	9.86	137.80	93.17	128.38			
98								1407.76	138.79	129.31
	545-7	—	1407.57	9.86	138.79	93.17	129.31			
								1398.91	137.23	127.86
99										
	528-2	—	1398.91	9.81	137.23	93.17	127.86			
								1389.68	136.19	126.81
100										
	529-2	—	1389.68	9.80	136.19	93.17	126.89			
								1398.68	137.49	128.10
101										
	542-5	—	1398.68	9.83	137.49	93.17	128.10			
								1396.97	137.18	127.81
102										
	542-7	—	1396.97	9.82	137.18	93.17	127.81			
								1404.21	137.47	128.03
103										
	547-2	—	1404.21	9.79	137.47	93.13	128.03			
								1403.04	136.94	127.53
104										
	548-7	—	1403.04	9.76	136.94	93.13	127.53			

TABLE 9
WEIGHTS OF FUEL RODS AND URANIUM
(Sheet 3 of 3)

Element No.	Long Rod No.	Short Rod No.	Weight (gm)	Wt % U	Wt of U	% Wt U ²³⁵	Wt of U ²³⁵	Total Wt of Rod	Total Wt of U	Total Wt of U ²³⁵
105								1388	135.79	127.25
	542-4	—	1388	9.84	135.79	93.17	127.25			
106								1388.93	137.87	128.42
	509-5		1132.49	9.93	112.46	93.15	104.75			
		510-1B	256.44	9.91	25.41	93.15	23.67			
107								1400.70	139.38	129.87
	510-6		1144.26	9.96	113.97	93.15	106.16			
		510-1C	256.44	9.91	25.41	93.15	23.67			
108								1378.93	136.87	127.59
	509-6		1122.49	9.93	111.46	93.15	103.82			
		510-1D	256.44	9.91	25.41	93.15	23.67			
109								1410	138.7	129.2
	552-4	—	1410	9.84	138.74	93.13	129.21			
110								1410	138.7	129.2
	551-3	—	1410	9.82	138.7	93.13	129.2			
111								1393	136.8	127.5
	542-2	—	1393	9.82	136.8	93.17	127.5			
112								1405.74	138.97	129.51
	526-6	—	1149.62	9.88	113.56	93.18	105.84			
		551-1AE	256.12	9.92	25.41	93.15	23.67			
113								1398.81	138.56	129.08
	513-3		1142.37	9.92	113.32	93.15	105.57			
		552-3A	256.50	9.84	25.24	93.15	23.51			
114								1395.30	138.21	128.75
	518-5		1138.90	9.92	112.97	93.15	105.24			
		552-3B	356.50	9.84	25.24	93.15	23.51			
115								1402.82	137.48	128.04
	547-5		1402.82	9.80	137.48	93.13	128.04			
116								1406.29	138.52	129.00
	545-6		1406.29	9.85	138.52	93.17	129.00			
117								1407.75	133.88	129.26
	551-4		1407.75	9.86	133.88	93.13	129.26			

TABLE 10
VALUES OF DENSITY AND HYDROGEN CONCENTRATION
(Sheet 1 of 2)

Element No.	Long Rod No.	Short Rod No.	Wt % H	Density (calculated)	$\frac{N_H}{\text{atoms}} \times 10^{-22}$ cm ³	$\frac{N_H}{\text{Av for Total Rod}}$	Remarks
81	524-8		1.83-1.84	6.04	6.62	6.62	
		519-5D	1.81-1.85	6.04	6.60		
82	510-3		1.76-1.82	6.02	6.44	6.43	Medium Crack
		510-1A	1.79-1.75	6.03	6.38		
83	520-4		1.81-1.85	5.99	6.55	6.56	
		519-5B	1.81-1.85	6.04	6.60		
84	519-3		1.86-1.75	6.04	6.51	6.53	
		519-5A	1.81-1.85	6.04	6.60		
85	525-7		1.80-1.82	6.04	6.53	6.52	
		516-1D	1.82-1.79	6.04	6.52		
86	517-2		1.77-1.84	6.03	6.52	6.52	
		516-1B	1.82-1.79	6.04	6.52		
87	520-6		1.77-1.86	6.04	6.53	6.52	
		511-1C	1.80-1.79	6.02	6.47		
88	511-2		1.81-1.81	6.03	6.52	6.52	
		516-1C	1.82-1.79	6.04	6.52		
89	520-1		1.79-1.80	6.04	6.47	6.47	
		511-1B	1.80-1.79	6.02	6.47		
90	513-2		1.74-1.84	6.03	6.52	6.51	
		511-1D	1.80-1.79	6.02	6.45		
91	513-4		1.78-1.78	6.04	6.42	6.42	
		512-2B	1.77-1.80	6.02	6.40		
92	513-7		1.84-1.70	6.04	6.39	6.39	
		512-2D	1.77-1.80	6.02	6.40		
93	511-5		1.76-1.77	6.04	6.38	6.38	
		512-2C	1.77-1.80	6.02	6.40		
94	516-7		1.78-1.77	6.05	6.43	6.42	
		512-2A	1.77-1.80	6.02	6.40		
95	516-4		1.87-1.81	5.98	6.57	6.58	
		519-5C	1.81-1.85	6.04	6.60		

TABLE 10
VALUES OF DENSITY AND HYDROGEN CONCENTRATION
(Sheet 2 of 2)

Element No.	Long Rod No.	Short Rod No.	Wt % H	Density (calculated)	$\frac{N_H}{cm^3} \times 10^{-22}$	$\frac{N_H}{Av}$ for Total Rod	Remarks
96	515-2		1.75-1.80	6.00	6.38	6.41	
		516-1A	1.82-1.79	6.04	6.52		
97	529-3	—	1.77-1.79	6.04	6.43	6.43	
98	545-7	—	1.78-1.84	6.08	6.57	6.57	
99	528-2	—	1.78-1.80	6.04	6.46	6.46	
100	529-2	—	1.80-1.81	6.00	6.49	6.49	Medium Sized Cracks
101	542-5	—	1.75-1.84	6.04	6.50	6.50	
102	542-7	—	1.71-1.80	6.03	6.34	6.34	
103	547-2	—	1.80-1.83	6.06	6.57	6.57	
104	548-7	—	1.77-1.78	6.06	6.42	6.42	
105	542-4	—	1.79-1.79	6.00	6.42	6.42	Small Cracks
106	509-5	—	1.79-1.81	6.00	6.45	6.45	Medium Cracks
		510-1B	1.79-1.75	6.03	6.38		
107	510-6		1.80-1.81	6.05	6.54	6.51	
		510-1C	1.79-1.75	6.03	6.38		
108	509-6		1.80	5.93	6.37	6.37	Large Cracks
		510-1D	1.79-1.75	6.03	6.38		
109	552-4	—	1.78-1.82	6.09	6.55	6.55	
110	551-3	—	1.75-1.77	6.09	6.40	6.40	
111	542-2	—	1.82-1.90	6.02	6.69	6.69	
112	526-6	—	1.78-1.83	6.08	6.55	6.54	
		511-1A&E	1.80-1.79	6.02	6.47		
113	513-3		1.68-1.82	6.04	6.30	6.31	
		552-3A	1.79-1.71	6.02	6.38		
114	518-5		1.79-1.76	6.03	6.41	6.42	
		552-3B	1.81-1.77	6.02	6.45		
115	547-5	—	1.81-1.80	6.06	6.53	6.53	
116	545-6	—	1.81-1.82	6.07	6.56	6.56	
117	551-4	—	1.75-1.84	6.08	6.54	6.54	

2. SNAPTRAN 10A/2-3 Fuel

The SNAPTRAN 10A/2-3 fuel elements were basically similar to SCA-4 elements. The only outward difference was the size of the end indexing pins, which had no bearing on the nuclear characteristics of the elements. A summary of the SNAPTRAN 10A/2-3 fuel element data is given in Table 11. The N_H values listed in all tables are in $\text{atoms/cm}^3 \times 10^{-22}$. The weight-gain value is considered to be more accurate than the other(s).

3. SNAP 10FR-1 Fuel

The S10FR-1 fuel elements were also basically similar to the SCA-4 elements, differing from both SCA-4 and SNAPTRAN 10A/2-3 in end pin design. A summary of their pertinent data is presented in Table 12.

4. Special Low N_H Fuel

The special low N_H fuel elements had the same basic dimensions as the SCA-4 fuel. A summary of their nuclear data is given in Table 13.

5. SNAP 8 Fuel

The SNAP 8 fuel elements have already been described in the text of the report. A summary of the data pertaining to their nuclear characteristics is presented in Table 14.

TABLE 11
SUMMARY OF U-235, N_H, AND Sm₂O₃ CONTENTS OF SNAPTRAN 10A/2-3
FUEL ELEMENTS

Fuel Element No.	Fuel Rod No.	U-235 Content (gm)			N _H			Sm ₂ O ₃ Content - mg			
		Rod Total	Major Section Homogeneity		Rod End Samples	Corrected End Samples	Weight Gain	Tube End Sample /In.	Element Total	Distribution	
			Max/In.	Min/In.						Max/In.	Min/In.
19	593-2	128.5	10.63	10.32	6.40	~	6.48	10.3	112	12.4	6.3
23	579-6	126.7	10.53	10.20	6.48	~	6.36	9.3	100	11.3	5.2
26	574-1	125.8	10.34	10.19	6.50	~	6.55	8.9	84	10.9	4.9
37	576-4	126.5	10.08	9.98	6.50	~	6.39	7.4	78	9.4	3.3
40	554-4	128.3	10.18	10.04	6.53	~	6.36	7.8	84	9.7	3.6
46	579-1	126.5	10.18	9.86	6.53	~	6.55	8.1	86	10.1	4.0
51	577-1	126.4	10.11	9.86	6.50	~	6.58	7.8	84	9.7	3.6
52	587-6	128.6	10.21	10.09	6.42	~	6.52	7.6	80	9.6	3.5
55	577-5	126.3	10.11	9.86	6.59	~	6.56	8.0	85	10.1	4.0
62	576-5	126.2	10.08	9.82	6.53	~	6.55	8.3	88	10.3	4.3
65	591-5	126.3	10.03	9.92	6.54	~	6.54	9.7	105	11.7	5.7
66	591-2	126.7	10.03	9.92	6.38	~	6.39	8.7	105	9.8	6.5
67	575-4	128.2	10.23	10.06	6.58	~	6.49	7.4	78	9.4	3.3
68	584-2	127.2	10.10	9.90	6.56	~	6.55	9.6	104	11.7	5.6
80	587-3	128.4	10.19	10.08	6.63	~	6.48	8.4	90	10.5	4.4
81	592-4	127.5	10.11	10.03	6.52	~	6.52	8.2	88	10.3	4.2
83	594-3	126.7	10.05	9.94	6.46	~	6.63	7.3	83	8.4	5.1
84	597-6	128.3	10.20	10.04	6.45	~	6.32	9.1	103	10.2	6.9
85	591-3	126.8	10.03	9.92	6.40	~	6.44	9.1	103	10.2	6.9
86	585-4	127.7	10.19	9.94	6.56	~	6.59	8.8	100	9.9	6.6
92	574-6	125.5	9.99	9.78	6.42	~	6.41	8.0	91	9.1	5.8
132	583-5	126.8	10.05	9.93	6.41	~	6.36	8.1	93	9.2	5.9
133	591-6	126.7	10.03	9.92	6.40	~	6.53	8.2	85	9.3	6.0
137	578-1	125.0	10.08	9.82	6.55	~	6.51	8.5	98	9.6	6.3
147	573-1	125.3	9.98	9.83	6.46	~	6.45	7.8	89	8.9	5.6
151	585-1	127.9	10.19	9.94	6.57	~	6.46	8.9	101	10.0	6.7
153	595-3	127.6	10.05	9.94	6.46	~	6.46	7.1	81	8.2	4.9
171	593-6	128.2	10.24	9.97	6.47	~	6.58	7.8	89	8.9	5.6
173	577-2	126.5	10.11	9.86	6.45	~	6.35	7.1	81	8.2	4.9
174	593-1	128.7	10.24	9.97	6.51	~	6.54	7.4	85	8.5	5.2
176	585-2	127.8	10.19	9.94	6.59	~	6.58	7.4	85	8.5	5.2
9	593-4	128.8	10.18	10.12	6.61	~	6.28	9.5	-	-	-
25	554-6	128.3	10.19	10.04	6.45	~	6.35	9.7	-	-	-
47	597-2	128.3	10.20	10.04	6.40	~	6.30	7.8	-	-	-
58	597-1	128.2	10.20	10.04	6.53	~	6.51	7.6	-	-	-
90	573-6	125.5	9.98	9.83	6.41	~	6.38	8.0	-	-	-
135	594-5	126.7	10.05	9.94	6.47	~	6.58	7.6	-	-	-

TABLE 12

SUMMARY OF U-235, N_H, AND Sm₂O₃ CONTENTS OF S10FR-1
FUEL ELEMENTS

Fuel Element No.	Fuel Rod No.	U-235 Content (gm)			N _H			Sm ₂ O ₃ Content (mg)			
		Rod Total	Major Section Homogeneity		Rod End Samples	Corrected End Samples	Weight Gain	Tube End Sample/In.	Element Total	Distribution	
			Max/In.	Min/In.						Max/In.	Min/In.
35	581-2	125.9	10.30	10.25	6.43	6.25	6.33	8.0	85	10.1	4.0
36	581-1	125.8	10.28	10.24	6.52	6.36	6.52	9.3	100	11.4	5.3
136	615-5	127.0	10.41	10.19	6.63	6.45	6.35	9.9	115	11.0	7.7
141	619-4	130.5	10.71	10.59	6.48	6.30	6.28	7.5	85	8.6	5.3
142	619-3	130.4	10.70	10.58	6.50	6.32	6.32	7.7	87	8.8	5.5
143	618-4	130.6	10.67	10.56	6.40	6.22	6.18	8.0	91	9.0	5.8
144	618-3	130.7	10.68	10.56	6.48	6.30	6.05	8.3	95	9.4	6.1
145	618-1	130.4	10.66	10.54	6.61	6.43	6.41	7.7	87	8.8	5.5
146	605-6	128.1	10.48	10.37	6.59	6.41	6.47	7.5	85	8.6	5.3
150	608-6	128.3	10.52	10.44	6.42	6.24	6.26	8.8	100	9.9	6.6
154	607-6	128.3	10.53	10.39	6.63	6.45	6.48	6.7	75	7.8	4.5
155	601-4	129.4	10.72	10.39	6.51	6.33	6.17	9.9	115	11.0	7.7
156	602-4	130.5	10.77	10.34	6.42	6.24	6.19	6.9	77	8.1	4.7
159	614-1	127.3	10.42	10.34	6.58	6.40	6.57	7.6	86	8.8	5.4
160	620-6	130.4	10.69	10.60	6.57	6.39	6.45	6.7	75	7.8	4.5
161	617-4	130.4	10.71	10.63	6.64	6.46	6.20	8.5	97	9.6	6.3
162	617-5	130.4	10.71	10.63	6.60	6.42	6.30	8.2	93	9.3	6.0
163	614-2	127.0	10.42	10.34	6.52	6.34	6.53	8.7	99	9.7	6.5
164	613-6	127.4	10.44	10.28	6.47	6.29	6.25	7.9	91	9.0	5.7
165	613-3	127.0	10.42	10.24	6.42	6.24	6.21	7.5	85	8.6	5.3
166	617-3	130.3	10.69	10.61	6.59	6.41	6.58	7.2	81	8.3	5.5
167	617-1	130.4	10.69	10.61	6.58	6.39	6.52	7.1	80	8.2	4.9
168	617-2	130.3	10.69	10.61	6.48	6.30	6.54	7.8	89	8.9	6.6
175	615-3	126.8	10.38	10.15	6.59	6.41	6.42	7.5	85	8.6	5.3
179	616-1	127.6	10.47	10.34	6.56	6.38	6.22	7.5	85	8.6	5.3
180	581-3	125.8	10.28	10.24	6.38	6.20	6.19	7.1	80	8.2	4.9
181	605-2	128.2	10.49	10.38	6.47	6.29	6.25	9.2	105	10.3	7.0
182	602-2	130.7	10.77	10.34	6.49	6.31	6.11	7.3	83	8.4	5.1
183	604-3	130.6	10.71	10.58	6.39	6.21	6.23	7.3	83	8.4	5.1
184	605-4	128.2	10.47	10.38	6.38	6.20	6.25	7.0	79	8.1	4.8
185	605-5	128.2	10.49	10.38	6.46	6.28	6.27	7.6	86	8.8	5.4
186	616-2	127.6	10.47	10.35	6.47	6.29	6.30	7.1	80	8.2	4.9
187	615-2	126.9	10.39	10.16	6.43	6.25	6.32	6.9	77	8.1	4.7
189	613-1	127.1	10.43	10.26	6.51	6.33	6.57	7.9	91	9.0	5.7
190	613-4	127.5	10.45	10.27	6.50	6.32	6.30	8.4	97	9.5	6.2
193	616-3	127.8	10.48	10.36	6.49	6.31	6.21	6.3	70	7.4	4.1
194	615-1	127.1	10.40	10.18	6.42	6.24	6.23	7.1	80	8.2	4.9
195	619-2	130.6	10.71	10.59	6.45	6.27	6.16	6.8	76	7.9	4.6
196	611-2	127.6	10.47	10.36	6.40	6.22	6.25	7.0	79	8.1	4.8
197	616-4	127.8	10.48	10.36	6.42	6.24	6.23	7.2	81	8.3	5.0

TABLE 13

SUMMARY OF U-235, N_H , AND Sm_2O_3 CONTENTS
OF SPECIAL LOW N_H FUEL ELEMENTS

Fuel Element Serial No.	U-235/Rod (gm)	Fuel Rod Average N_H	Sm_2O_3 /In. (mg)
111	127.7	6.05	8.7
112	129.9	5.93	9.3
113	129.1	5.87	7.9
114	129.7	6.02	7.5
115	126.5	5.91	9.1
116	126.4	5.97	8.3
126	129.1	6.00	7.8
127	129.0	6.10	7.6
128	128.8	5.90	8.2
129	128.7	6.00	8.6

TABLE 14
SUMMARY OF U-235, N_H, AND Sm₂O₃ CONTENTS
OF SNAP 8 FUEL ELEMENTS
(Sheet 1 of 6)

Fuel Element Serial No.	Fuel Rod Serial No.	U-235/Rod (gm)	Maximum U-235/In. (gm)	Minimum U-235/In. (gm)	Isotopic Enrichment	Fuel Rod Average N _H	Sm ₂ O ₃ /In. (mg)
109	550B-2	30.09	2.19	2.08	93.16	5.91	2.8
187	563C-3	30.94	2.24	2.15	93.15	6.00	3.5
191	558D-2	30.15	2.18	2.12	93.15	5.93	3.2
193	570A-1	30.69	2.20	2.18	93.15	6.05	3.1
197	550B-3	30.09	2.19	2.08	93.16	5.90	3.4
201	569D-2	30.07	2.19	2.13	93.15	5.79	3.2
237	570B-1	30.00	2.16	2.12	93.15	5.84	3.2
281	530D-1	30.39	2.19	2.16	93.19	6.19	3.2
293	531C-4	30.25	2.23	2.12	93.17	6.05	3.1
304	571D-4	30.44	2.20	2.14	93.15	6.15	3.3
340	565D-3	30.18	2.18	2.13	93.15	5.84	3.6
341	565C-4	30.62	2.26	2.15	93.15	5.87	3.0
146	570A-3	30.77	2.21	2.19	93.15	6.07	3.1
171	549C-4	30.92	2.25	2.17	93.16	5.84	3.1
177	558A-4	30.72	2.21	2.18	93.15	5.97	2.7
179	555B-3	29.83	2.20	2.10	93.13	5.70	3.6
303	565A-4	29.86	2.16	2.11	93.15	5.90	2.9
309	559C-3	30.64	2.21	2.18	93.15	5.82	2.9
313	550D-1	31.55	2.26	2.23	93.16	5.81	2.8
318	567D-4	30.32	2.21	2.14	93.15	5.83	3.1
346	565A-3	29.88	2.16	2.11	93.15	6.03	2.7
353	569D-3	30.05	2.19	2.12	93.15	5.82	2.5
383	571A-3	30.28	2.17	2.16	93.15	6.15	2.7
410	561D-2	30.77	2.23	2.17	93.15	5.82	2.7
150	564B-3	30.28	2.21	2.14	93.15	5.84	3.3
176	562A-2	30.63	2.24	2.12	93.15	5.81	3.1
192	557C-3	30.50	2.22	2.13	93.13	5.96	2.8
245	558D-4	30.16	2.18	2.12	93.15	5.97	3.8
272	561C-4	31.39	2.28	2.19	93.15	5.95	3.6
288	571C-4	30.88	2.22	2.19	93.15	5.86	3.6
311	564B-2	30.26	2.16	2.14	93.15	6.08	3.0
323	565B-2	30.80	2.21	2.18	93.15	5.85	3.2
332	565C-2	30.66	2.24	2.15	93.15	5.85	2.9
352	569D-4	30.05	2.19	2.12	93.15	5.84	2.9
358	565C-1	30.66	2.24	2.15	93.15	5.85	2.8
372	563A-2	30.25	2.17	2.14	93.15	5.97	2.5

TABLE 14
SUMMARY OF U-235, N_H, AND Sm₂O₃ CONTENTS
OF SNAP 8 FUEL ELEMENTS
(Sheet 2 of 6)

Fuel Element Serial No.	Fuel Rod Serial No.	U-235/Rod (gm)	Maximum U-235/In. (gm)	Minimum U-235/In. (gm)	Isotopic Enrichment	Fuel Rod Average N _H	Sm ₂ O ₃ /In. (mg)
181	558E-4	30.00	2.18	2.09	93.15	5.97	2.9
206	572C-4	30.34	2.26	2.10	93.15	5.93	3.4
262	572C-2	30.34	2.26	2.10	93.15	5.97	3.2
268	572C-1	30.34	2.26	2.10	93.15	5.91	3.9
278	565A-2	29.88	2.16	2.11	93.15	5.86	2.6
302	572B-3	30.36	2.19	2.14	93.15	5.88	3.1
315	560C-1	31.03	2.25	2.19	93.15	5.93	3.6
329	557A-2	30.98	2.22	2.19	93.13	5.94	2.3
330	555B-4	29.76	2.19	2.09	93.13	5.97	3.3
335	559A-1	30.40	2.19	2.14	93.15	5.97	2.8
336	559A-2	30.41	2.19	2.14	93.15	6.09	3.2
357	555A-1	30.45	2.21	2.16	93.13	6.06	2.5
141	563D-2	31.29	2.26	2.21	93.15	6.06	3.4
149	570B-2	30.11	2.17	2.13	93.15	5.84	3.4
154	567B-3	30.39	2.27	2.13	93.15	5.88	3.2
174	572C-3	30.34	2.26	2.10	93.15	6.01	3.7
182	571C-2	30.89	2.22	2.19	93.15	5.91	3.2
231	571C-3	30.87	2.22	2.19	93.15	5.84	3.0
236	569C-3	30.49	2.23	2.16	93.15	5.83	3.4
292	568D-2	30.56	2.22	2.15	93.15	5.84	3.6
319	555C-3	29.78	2.17	2.10	93.13	6.00	3.2
339	562A-3	30.61	2.24	2.12	93.15	5.86	2.6
342	549B-2	30.37	2.19	2.16	93.16	5.92	2.8
356	557A-1	31.02	2.23	2.19	93.13	5.97	2.9
212	535A-1	30.51	2.20	2.15	93.17	6.19	3.4
244	572B-4	30.37	2.19	2.14	93.15	5.93	3.7
249	568C-2	29.78	2.16	2.10	93.15	5.96	3.6
274	557A-3	31.05	2.23	2.19	93.13	5.86	2.4
289	570A-4	30.79	2.21	2.19	93.15	5.92	2.9
316	571C-1	30.92	2.22	2.20	93.15	5.93	*
322	555D-2	29.32	2.11	2.08	93.13	5.92	2.5
350	561C-3	31.36	2.27	2.19	93.15	5.84	2.5
359	555C-4	29.76	2.17	2.10	93.13	6.03	2.7
396	571B-1	30.34	2.24	2.09	93.15	5.80	2.8
400	559A-5	30.43	2.19	2.15	93.15	5.88	2.4
430	562D-2	30.85	2.23	2.19	93.15	5.81	4.4

TABLE 14
SUMMARY OF U-235, N_H, AND Sm₂O₃ CONTENTS
OF SNAP 8 FUEL ELEMENTS
(Sheet 3 of 6)

Fuel Element Serial No.	Fuel Rod Serial No.	U-235/Rod (gm)	Maximum U-235/In. (gm)	Minimum U-235/In. (gm)	Isotopic Enrichment	Fuel Rod Average N _H	Sm ₂ O ₃ /In. (mg)
137	569B-4	30.03	2.16	2.13	93.15	5.92	2.8
153	565B-1	30.77	2.21	2.18	93.15	6.03	2.6
155	569C-4	30.46	2.22	2.15	93.15	5.84	2.8
175	570C-4	30.58	2.20	2.17	93.15	5.91	2.9
241	563D-1	31.20	2.25	2.20	93.15	6.00	3.6
271	559C-4	30.62	2.21	2.18	93.15	5.82	3.0
280	556A-4	30.16	2.21	2.12	93.13	6.12	3.2
283	557A-4	31.02	2.23	2.19	93.13	5.97	2.4
307	559C-1	30.67	2.21	2.18	93.15	6.09	3.3
337	569B-2	30.11	2.17	2.13	93.15	5.97	2.7
347	567C-4	30.25	2.23	2.13	93.15	5.89	2.5
367	560B-4	30.46	2.20	2.15	93.15	5.82	2.5
349	570A-2	30.80	2.21	2.19	93.15	5.85	2.8
370	557D-1	30.91	2.24	2.19	93.13	6.09	3.2
378	549B-1	30.39	2.19	2.16	93.16	5.80	3.4
379	569A-2	30.92	2.27	2.14	93.15	5.80	2.9
384	565B-4	30.76	2.21	2.18	93.15	5.93	2.4
403	558A-1	30.70	2.21	2.18	93.15	5.97	2.9
415	562C-4	30.35	2.19	2.14	93.15	5.86	3.5
427	578A-3	30.97	2.24	2.17	93.15	5.96	3.5
429	568A-1	30.97	2.24	2.17	93.15	5.80	3.3
433	562D-4	30.84	2.23	2.19	93.15	5.87	3.3
450	563C-1	30.99	2.24	2.16	93.15	5.86	3.5
461	560B-2	30.49	2.20	2.15	93.15	5.86	3.5
308	557D-4	30.97	2.24	2.19	93.13	6.09	3.0
348	562C-1	30.31	2.18	2.14	93.15	5.88	2.7
388	565D-4	30.18	2.18	2.13	93.15	5.80	2.6
398	559B-1	30.54	2.23	2.13	93.15	5.80	4.1
443	530A-1	29.79	2.14	2.11	93.19	6.20	3.4
452	571A-2	30.30	2.17	2.16	93.15	5.82	3.3
453	582B-1	30.52	2.22	2.12	93.28	6.15	2.9
456	565A-1	29.86	2.16	2.12	93.15	5.94	3.0
457	568A-4	30.93	2.24	2.17	93.15	6.03	3.0
458	562D-1	30.71	2.22	2.18	93.15	5.85	3.8
463	563B-3	29.93	2.18	2.12	93.15	5.93	3.5
464	569C-1	30.51	2.23	2.16	93.15	6.09	3.7

TABLE 14
SUMMARY OF U-235, N_H, AND Sm₂O₃ CONTENTS
OF SNAP 8 FUEL ELEMENTS
(Sheet 4 of 6)

Fuel Element Serial No.	Fuel Rod Serial No.	U-235/Rod (gm)	Maximum U-235/In. (gm)	Minimum U-235/In. (gm)	Isotopic Enrichment	Fuel Rod Average N _H	Sm ₂ O ₃ /In. (mg)
343	555D-3	29.27	2.11	2.08	93.13	5.81	2.9
369	561A-4	31.14	2.26	2.19	93.15	5.87	2.7
408	557C-4	30.52	2.22	2.14	93.13	6.03	4.5
420	558C-3	30.31	2.20	2.09	93.15	6.03	3.0
423	572D-1	29.77	2.15	2.09	93.15	5.93	3.2
425	571D-3	30.48	2.20	2.14	93.15	6.02	4.0
428	567C-1	30.29	2.26	2.02	93.15	5.83	3.5
437	560B-3	30.50	2.20	2.15	93.15	6.04	3.3
439	567D-2	30.25	2.20	2.13	93.15	6.06	3.6
440	549D-4	31.65	2.27	2.20	93.16	5.90	3.3
442	563B-2	29.99	2.15	2.12	93.15	5.94	4.0
448	549D-3	31.67	2.27	2.20	93.16	5.77	3.2
462	559D-1	30.17	2.17	2.14	93.15	6.04	3.6
445	549C-2	30.93	2.24	2.15	93.16	6.08	2.9
441	550C-2	30.39	2.22	2.11	93.16	6.05	3.1
436	567A-4	30.45	2.23	2.13	93.15	5.85	3.8
432	567B-2	30.41	2.27	2.13	93.15	5.92	3.1
431	558C-4	30.29	2.20	2.09	93.15	5.97	3.5
422	556B-4	30.56	2.27	2.14	93.13	5.96	3.9
413	568B-1	30.41	2.25	2.17	93.15	5.95	4.2
397	549D-2	31.52	2.27	2.20	93.16	5.88	2.6
232	565C-5	30.61	2.26	2.15	93.15	5.81	2.8
470	537A-2	29.77	2.15	2.08	93.17	5.97	3.4
419	549C-1	30.96	2.24	2.15	93.16	6.06	3.5
471	556D-3	30.44	2.22	2.13	93.13	5.94	3.1
489	568C-1	29.80	2.19	2.10	93.15	5.82	2.7
483	571D-2	30.52	2.21	2.15	93.15	5.97	2.9
434	550D-3	31.32	2.25	2.22	93.16	6.06	2.7
485	571B-4	30.34	2.24	2.08	93.15	6.03	3.6
488	534D-4	30.54	2.25	2.14	93.17	6.20	3.3
482	560C-2	31.02	2.25	2.19	93.15	5.89	3.9
472	561C-2	31.32	2.27	2.18	93.15	5.88	3.3
475	568B-4	30.47	2.23	2.14	93.15	5.87	2.8
476	572B-2	30.29	2.18	2.13	93.15	6.03	2.6
491	567A-1	30.18	2.21	2.12	93.15	5.82	3.0
474	572A-3	30.26	2.19	2.13	93.15	6.09	3.1

TABLE 14
SUMMARY OF U-235, N_H, AND Sm₂O₃ CONTENTS
OF SNAP 8 FUEL ELEMENTS
(Sheet 5 of 6)

Fuel Element Serial No.	Fuel Rod Serial No.	U-235/Rod (gm)	Maximum U-235/In. (gm)	Minimum U-235/In. (gm)	Isotopic Enrichment	Fuel Rod Average N _H	Sm ₂ O ₃ /In. (mg)
486	582C-2	30.98	2.23	2.16	93.28	6.15	3.5
361	571B-2	30.32	2.24	2.08	93.15	6.00	2.4
402	563A-3	30.25	2.17	2.14	93.15	6.03	3.7
286	532D-2	30.08	2.20	2.14	93.17	6.09	2.8
394	565B-3	30.77	2.20	2.19	93.15	5.75	2.9
412	558D-3	30.18	2.18	2.12	93.15	5.95	3.5
416	555D-1	30.22	2.11	2.08	93.16	6.01	3.7
446	561D-1	30.73	2.22	2.17	93.15	5.84	4.1
466	567C-3	30.28	2.22	2.13	93.15	5.92	3.4
477	558A-2	30.69	2.20	2.18	93.15	6.03	3.4
479	559C-2	30.66	2.21	2.18	93.15	5.85	3.4
492	558D-1	30.02	2.17	2.12	93.15	5.96	3.4
503	556D-1	30.44	2.22	2.13	93.13	5.98	3.2
392	559A-3	30.42	2.19	2.15	93.15	5.96	3.2
417	562A-1	30.63	2.24	2.12	93.15	5.83	3.7
389	564A-2	30.62	2.21	2.16	93.15	5.94	3.0
454	567A-3	30.41	2.23	2.13	93.15	6.01	2.7
426	567C-2	30.22	2.22	2.12	93.15	5.99	3.9
490	572B-1	30.50	2.19	2.13	93.15	5.97	2.7
487	582A-2	30.76	2.22	2.17	93.28	6.13	3.2
449	582A-3	30.77	2.22	2.17	93.28	6.19	3.1
447	582A-4	30.76	2.22	2.17	93.28	6.11	2.5
360	582B-2	30.56	2.22	2.13	93.28	6.03	3.4
478	582B-3	30.56	2.22	2.13	93.28	6.11	2.7
481	582C-3	30.98	2.23	2.16	93.28	6.15	3.4
396	532C-2	29.91	2.16	2.11	93.17	6.15	2.7
505	555A-4	30.43	2.21	2.16	93.13	6.09	3.6
510	556B-3	30.55	2.27	2.14	93.13	6.09	2.9
511	571D-1	30.44	2.20	2.14	93.15	5.95	3.3
513	582C-4	30.97	2.23	2.16	93.28	6.20	3.3
518	572A-2	30.26	2.19	2.13	93.15	6.01	3.1
519	564B-4	30.28	2.21	2.14	93.15	5.81	3.1
533	531D-4	30.23	2.20	2.12	93.17	6.15	2.9
523	555A-2	30.46	2.21	2.16	93.13	5.84	3.0
526	556A-3	30.15	2.21	2.12	93.13	6.05	3.3
527	564B-1	30.25	2.21	2.14	93.15	6.04	3.5

TABLE 14
SUMMARY OF U-235, N_H, AND Sm₂O₃ CONTENTS
OF SNAP 8 FUEL ELEMENTS
(Sheet 6 of 6)

Fuel Element Serial No.	Fuel Rod Serial No.	U-235/Rod (gm)	Maximum U-235/In. (gm)	Minimum U-235/In. (gm)	Isotopic Enrichment	Fuel Rod Average N _H	Sm ₂ O ₃ /In. (mg)
111	531C-1	30.26	2.23	2.12	93.18	6.20	2.8
287	531C-3	30.25	2.23	2.12	93.18	6.15	2.8
291	555C-2	29.80	2.17	2.10	93.18	6.09	3.2
522	568A-2	30.97	2.24	2.17	93.15	5.85	3.0
451	568B-3	30.43	2.23	2.13	93.15	6.07	2.8
164	569B-3	29.93	2.16	2.12	93.15	6.11	3.3
484	570D-1	30.23	2.19	2.13	93.15	5.87	2.6
517	559B-4	30.60	2.23	2.13	93.15	6.03	3.4
515	560A-2	30.58	2.21	2.15	93.15	5.81	4.0
424	567A-2	30.42	2.23	2.13	93.15	5.98	4.3
521	567D-3	30.33	2.21	2.14	93.15	5.83	4.3
529	570D-2	30.22	2.18	2.13	93.15	6.00	4.1
247	571A-4	30.29	2.17	2.15	93.15	5.89	2.5
411	557B-1	30.22	2.21	2.12	93.13	5.98	3.4
421	536C-4	30.64	2.20	2.17	93.17	6.15	2.7
499	556C-3	29.21	2.13	2.09	93.13	5.97	3.4
506	555B-2	30.64	2.23	2.12	93.13	5.79	4.3
508	563C-2	31.14	2.26	2.17	93.15	5.81	4.1
528	531A-1	30.03	2.16	2.12	93.18	6.15	3.4
530	557D-2	30.97	2.24	2.19	93.13	6.03	4.1
531	540D-4	29.62	2.12	2.11	93.17	6.20	3.5
525	550A-1	30.01	2.24	2.16	93.16	5.85	4.1
500	557B-5	30.05	2.19	2.10	93.13	5.85	3.8
351	582A-1	30.72	2.22	2.17	93.28	6.05	3.9
310	565C-3	30.66	2.24	2.15	93.15	5.85	3.2
493	564A-3	30.68	2.21	2.16	93.15	5.85	3.2
501	538C-4	29.50	2.15	2.07	93.17	6.09	3.5
504	561C-1	31.51	2.29	2.20	93.15	5.86	3.9
509	564A-1	30.61	2.21	2.16	93.15	6.03	3.9
514	569D-1	30.01	2.21	2.12	93.15	5.83	3.6
516	562A-4	30.63	2.24	2.12	93.15	5.74	3.9
520	560A-3	30.41	2.20	2.14	93.15	5.87	3.6
524	555B-1	30.40	2.23	2.11	93.13	5.89	3.7
532	582B-4	30.44	2.21	2.12	93.28	6.18	3.7
188	532D-3	29.96	2.19	2.09	93.17	6.21	3.4

APPENDIX B

CORE VESSEL AND TOP TANK ASSEMBLIES

1. SNAP 10A/2

Two core vessel and top tank assemblies were used in the SNAP 10A/2 configurations. The one shown in Figure 48 had approximately two-thirds of the core vessel below the base of the top tank and the remainder above. Test sleeves for this configuration were therefore made in sections to fit the divided core vessel. The base tank is shown in the up position. The assembly shown in Figure 49 was designed to allow for attaching the radial beryllium reflector. The full-length shipping sleeve was also tested using this assembly. The base tank is shown in the up position as before.

2. SNAP 8

The core vessel and top tank assembly used in all SNAP 8 configurations is shown in Figure 50. All sleeves were made in sections to fit the divided core vessel radial surface. The base tank is shown in the up position.

NOTE: ALL DIMENSIONS IN INCHES
BASE TANK IN UP POSITION

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Figure 48. SNAP 10A/2 Reflector Assembly

NOTE : ALL DIMENSIONS IN INCHES
BASE TANK IN UP POSITION

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Figure 49. SNAP 10A/2 Reflector Assembly (Small Top Tank)

NOTE: ALL DIMENSIONS IN INCHES
BASE TANK IN UP POSITION

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Figure 50. SNAP 8 Reflector Assembly

APPENDIX C

PULSED NEUTRON DECAY IN SUBCRITICAL REACTIVITY MEASUREMENT

1. General Considerations

The subcritical reactivity of a multiplying medium may be determined from the decay of the neutron flux caused by an introduced neutron pulse. The rate of decay increases with greater subcritical reactivity. The exponential decay rate should be measured after the initial transients from the neutron burst have died out and before the appearance of the delayed neutrons. In this way, the prompt decay is determined.

The prompt exponential decay of the neutron flux may be represented by:

$$\phi \sim e^{-\alpha t},$$

in which the decay constant, α , is to be measured. The constant, α , is related to the subcritical reactivity by:

$$\alpha = \frac{\beta}{\ell^*} (1 + R),$$

where

β = effective delayed neutron fraction

ℓ^* = prompt neutron lifetime

R = subcritical reactivity in dollars.

If β/ℓ^* is known (e.g. from noise analysis), then determination of one value of α allows calculation of R . Otherwise β/ℓ^* itself may be obtained from the measurement of two subcritical configurations with a known difference in subcritical reactivity. The following equation may be used:

$$\frac{\beta}{\ell^*} = \frac{\alpha_2 - \alpha_1}{\Delta R}$$

The known difference in subcritical reactivity, ΔR , can be achieved by adding poison of predetermined worth to the desired subcritical configuration. The foregoing equation is valid when the added poison has a negligible effect on β/ℓ^* .

A more detailed discussion of the pulsed neutron technique in determining subcritical reactivity is given in Reference 5.

2. Equipment

The neutron generator used was a Kaman model A-1000 manufactured by Kaman Nuclear, Colorado Springs, Colorado. The generator produces 14.1 Mev neutrons by the bombardment of a tritiated titanium target with a deuteron ion beam. It may be operated in either a continuous or a pulsed mode. In order to measure the decay of neutron flux in a subcritical reactor, the following additional equipment was used:

- a) Four scintillation-type neutron detectors with power supplies and amplifiers
- b) A TMC (Technical Measurements Corp.) time analyzer with a model 212 pulsed neutron logic plug-in unit
- c) A trigger pulse shaper to trigger the TMC analyzer
- d) An oscilloscope to observe the beam pulse striking the target of the neutron generator.

The functional relationships of these items are shown in Figure 51. The broken line separates the items located in the test cell from those in the control room. The neutron generator controls were adjusted to obtain the optimum pulse shape on the oscilloscope. The plutonium-beryllium startup source used in assembling the reactor core was kept as far as possible from the scintillation neutron detectors. It has no function in pulsed neutron decay measurements. The scintillation neutron detectors were placed as close to the core as practicable. The decreasing side of the ion pulse at the target was shaped and used to synchronize the TMC analyzer. The analyzer recorded the counts of the neutron detectors on 256 channels as a function of time. In this manner, the maximum effect of a neutron pulse and the subsequent decay were determined at the location of the neutron detectors. The neutron flux at the detectors was assumed to be proportional to that in the reactor core.

3. Experimental Procedure

After the installation of the equipment, the gain settings on the Hewlett-Packard amplifiers and the discriminator of the TMC plug-in unit were set so

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Figure 51. Pulse Neutron Decay Apparatus Block Diagram

that no gamma or X-ray counts were observed on the display of the TMC analyzer, which was triggered with an external pulser (neutron generator not operating). A very low (background) counting rate of several counts per minute was observed with the proper settings.

The four neutron detectors were then disconnected and checked one at a time. This was done by placing the 5-curie Plutonium Beryllium source near the connected detector and observing the TMC display. A much greater number of counts, randomly distributed, was observed when the detector was functioning properly.

TMC channel widths and the neutron generator pulse rate were chosen so that the 256 channels fitted between pulses. Each of the channels represents a point on the time axis.

The neutron generator was started and checked for normal operation according to the instruction manual.

The deuteron beam pulse width was adjusted to approximately 100 μ sec with a repetition rate of approximately 100/sec. These were found to be suitable values for the SNAP cores tested.

TMC analyzer settings were made. Typical settings were 10- μ sec channels, 2 channels of background, and 2 channels of delay. In this mode of operation, the channels fall at 20- μ sec intervals; i.e., there is a dead time of 10 μ sec between channels. The 10- μ sec dead time occurs regardless of which channel width is chosen.

The reactor was assembled to the desired subcritical configuration, and the neutron generator was started.

Operation was continued for approximately 30 min, then the neutron generator and TMC analyzer were stopped. The accumulated count data in the TMC analyzer were printed out.

4. Analytical Procedure

The last 10 of the 256 channels were used as background counts. The average of these values was subtracted from each of the other channel counts. These corrected data were then plotted on semi-log graph paper with counts-per-channel as the ordinate (log) and time as the abscissa (linear). The time in seconds required for the counts to decrease by a factor of "e" on the straight portion of the plot is equal to the reciprocal of the decay constant; i.e., $1/\alpha$. If the β/ℓ^* value were known, the subcritical reactivity was calculated from:

$$R = \frac{\ell^*}{\beta} (\alpha) - 1.$$

If β/ℓ^* were not known, an additional measurement was made after poison of predetermined worth had been added to the core. This parameter was then calculated from:

$$\frac{\beta}{\ell^*} = \frac{\alpha_2 - \alpha_1}{R_2 - R_1}.$$

Knowing β/ℓ^* , it was then possible to calculate the subcritical reactivity for the desired configuration.

APPENDIX D

NOISE SPECTRA ANALYSIS

1. Introduction

A highly sensitive neutron detection system, capable of handling neutron count rates up to 10^7 /sec, was used in conjunction with a fast count rate circuit to study the fluctuations in the neutron level in the SNAP critical assemblies at frequencies from 10 to 80,000 cps. The detection system consisted of four scintillation detectors operating in parallel and installed at the top of the reactor. Pulses from the scintillators were combined using cathode follower mixers, amplified with four cascaded 100-Mc linear amplifiers, and coupled to a fast count rate circuit. The output of the count rate circuit is the equivalent of ion chamber current, but the useful frequency range of the count rate circuit extends beyond 100,000 cps. The fluctuating portion of the signal from the fast count rate circuit was amplified and recorded on magnetic tape. A modified Technical Products wave analyzer system was used to analyze the recorded random signals. The system contains a band pass filter which breaks the random signal into narrow frequency bands. The signal from the filter is electronically squared and averaged. A series of these measurements at various frequencies represents the power spectral density function of the reactor noise signal and is proportional to the square modulus of the reactor transfer function.

The power spectral density of the reactor noise over the frequency range from 10 to 80,000 cps is dependent on three reactor system time constants. The interpretation of the three break frequencies is based on the kinetics model described later. One time constant is approximately the average duration of the prompt neutron chains ℓ^*/β , where $1/\beta$ is the average number of events in the chains (β is the effective delayed neutron fraction) and ℓ^* is the average time between the events (average prompt neutron lifetime). Since this is a two-region reactor, ℓ^* depends on ℓ_c (the average core lifetime), ℓ_r (the average length of time neutrons spend in the reflector), and β_7 (the fraction of the neutrons appearing in the core which came from the reflector).

$$\ell^* = \ell_c + \beta_7 \ell_r$$

The second time constant is simply ℓ_r . The third is approximately the average duration of prompt neutron chains, $\ell_c/\beta + \beta_7$, exclusive of reflector neutrons, where $1/\beta + \beta_7$ is the average number of events in these subchains and ℓ_c is the average time between events. This time constant would dominate the high-frequency kinetics behavior of the reactor even if the reflector were completely removed.

A highly efficient detector can detect short bursts of leakage neutrons, corresponding to the occurrence of individual prompt neutron chains. These chains occur at random and generate a random square wave signal. The spectral density of such a signal is proportional to $B/\omega^2 + \alpha_1^2$, where $1/\alpha_1$ is approximately ℓ^*/β , the average duration of the short bursts described previously, B is proportional to the average rate at which the bursts occur, and ω is equal to $2\pi f$.

Many of the detected neutron pairs will be separated by an average time interval related to the reflector lifetime. This effect also tends to generate a random square wave signal. The spectral density of such a signal is $C/\omega^2 + \alpha_2^2$, where $1/\alpha_2$ is the average reflector lifetime, ℓ_r , and C is proportional to the average rate at which the pairs are detected. Short bursts of neutrons, within the prompt chains, may also be detected if the detector has an extremely fast response. These subchains correspond to prompt chains within the core, with an average duration of approximately $\ell_c/\beta + \beta_7$. This effect also tends to generate a random square wave signal with a power spectral density proportional to $D/\omega^2 + \alpha_3^2$, where $1/\alpha_3$ is approximately $\ell_c/\beta + \beta_7$.

Since the neutrons arrive at the detector at a random rate, another random square wave signal is generated by this detection process. However, the power spectral density of such a signal will appear as a constant, proportional to the average detection rate, at the frequencies of interest for the experiment because of the extremely short average duration of the pulses. Thus, for this experiment, it will be assumed that the power spectral density of the detector output is simply the sum of these four effects:

$$S_y(\omega) = A + \frac{B}{\omega^2 + \alpha_1^2} + \frac{C}{\omega^2 + \alpha_2^2} + \frac{D}{\omega^2 + \alpha_3^2}$$

This equation can be written in the form:

$$S_y(\omega) = A + \frac{K(\omega^2 + \alpha_4^2)(\omega^2 + \alpha_5^2)}{(\omega^2 + \alpha_1^2)(\omega^2 + \alpha_2^2)(\omega^2 + \alpha_3^2)}$$

The same equation is mathematically derived from a two-region reactor kinetics model. A digital curve fitting code is used to determine the best values of α_1 , α_2 , and α_3 from the measured power spectral density data.

2. Data Acquisition System

Measurements of the high-frequency portion of the reactor transfer function of small metal reactors from the power spectral density of the reactor neutron level fluctuations require that the device used to detect these fluctuations has a very fast response and a high detection efficiency. The high frequency response of ion chambers is severely limited by the R-C time constant of the chamber circuit and the ion collection times. A successful alternate approach has been the use of a pulse-type detector and a fast count rate circuit to simulate ion chamber current. The frequency response of such a system is limited only by the averaging characteristics of the count rate circuit and can be easily measured using a signal generator and an oscilloscope. There is no need to expose the detector to a strong random gamma or neutron source to measure the system frequency response.

Most pulse chambers have severe limitations at high count rates (over 100,000 c/sec) because of pulse pile up and the dead time of the standard type of linear amplifiers (DD2) commonly used to process the chamber pulses.

However, scintillation detectors (with zinc sulfide, silver-activated, boron-loaded crystals) meet the requirements for observing the decay of prompt neutron chains in a small metal reactor. These neutron detectors provide extremely short output pulses with a rise time which is less than 50 nanosec and can be used at detection rates up to about 10^7 /sec.

The scintillator outputs are coupled to cathode follower mixers through coax cables less than 8 in. long to avoid capacitive effects which attenuate and lengthen the output pulses. Two detectors are connected to each preamp-mixer and the preamp outputs are coupled directly together at the input of the first of four

cascaded Hewlett-Packard 100-Mc linear amplifiers. The schematic of the preamp-mixer and the block diagram of the complete detector system are shown in Figures 52 and 53, respectively. The use of short cables to the preamps and the use of special preamps and amplifiers with a high frequency response was necessary to preserve the fast rise time of the pulses from the detectors, which permits useable count rates up to about 10^7 /sec.

Pulses from the wide band linear amplifiers are coupled directly to a fast count rate circuit. The output of the count rate circuit is the equivalent of ion chamber current, but the useful frequency range of the count rate circuit shown in Figure 54 extends beyond 100,000 cps. The frequency response is limited only by the averaging time of the circuit and can be easily measured with a signal generator and ac voltmeter or oscilloscope, as shown in Figure 55. Details for the design of a basic count rate circuit are given in Reference 6. As shown in Figure 54, a high pass filter is included at the output of the count rate circuit. The high pass filter eliminates slow frequency drift problems but still has a flat frequency response above 1 cps. The count rate signal is amplified and recorded on magnetic tape with both FM and direct-recording systems. The three recordings indicated in Figure 56 cover the frequency range from 1 to 10^5 cps. The two FM recording systems provide high recording accuracy (large dynamic range, large signal-to-noise ratio) in the important lower frequency ranges.

3. Data Reduction System

A random signal may be analyzed by passing it through a narrow band pass filter, which can be tuned to a number of different frequencies, and measuring the mean-square value of the resultant signal at each frequency. A series of measurements, of the mean-squared amplitude per unit of frequency of the random input signal, is called the power spectral density (because the square of the amplitude of a signal is proportional to the power of the signal).

The power spectral density of the recorded reactor noise signal is measured, using a modified Technical Products noise analyzer system shown in Figure 57. The narrow band filters, which have a center frequency of 97,000 cps are electronically tuned to various frequencies in the range from 2 to 10,000 cps by means of a heterodyne signal from a variable high-frequency oscillator. A precise measurement of the electronic position of the filter is determined by electronically "subtracting" a stable 97,000-cps signal from the frequency of the

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Figure 52. Cathode Follower - Mixer Preamplifier

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Figure 53. Detection System Block Diagram

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Figure 54. Count Rate Circuit and High Pass Filter

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Figure 55. System for Measuring the Frequency Response of the Count Rate Circuit

heterodyne signal and measuring the relatively low difference frequency. Slow drift in the heterodyne signal frequency is detected by comparing the phase relation between the difference frequency and a stable low-frequency signal generator with a lissajous pattern on an oscilloscope. The slow drift can then be corrected by changing the frequency of the variable high-frequency oscillator with the manual control.

The signal from the output of the band pass filter is electronically squared and averaged. The average value of the squared signal from the band pass filter is obtained by integrating the signal for 300 sec and measuring the final value with a digital voltmeter. A series of these measurements at different frequencies can be used to make a graph of the power spectral density of the input

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Figure 56. Data Acquisition System for High-Frequency Noise Measurements

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Figure 57. Heterodyne Analyzer System for Measuring the Power Spectral Density of a Random Signal

signal. A block diagram of the data reduction system is given in Figure 58. A more detailed description of the analyzer system is included in Reference 7.

4. Installation of the Data Acquisition System

The optimum detector location for this type of experiment is in the core itself. However, in addition to the fact that this would have been very difficult to arrange, it is felt that the presence of a neutron absorber (boron-loaded detector) in the core might have seriously affected the measured lifetime. The next best detector location was at the edge of the core. The most accessible edge of the core in the small SNAP reactors is the top, in the region above the upper grid plate. The small physical size (about 2 in. diameter) permitted four of the detectors to be grouped together and placed on top of the reactor. The combination of four detectors operating in parallel and located at the upper edge of the core provided a reasonably high detection efficiency. The mixers were assembled in small miniboxes and used a single remote power supply to permit their installation within about 6 in. of the detectors. The preamp power supply, the high voltage supply for the detectors, and the first 100-Mc linear amplifier were placed at a convenient spot in the reactor test cell. Coax cables were used for all power supply and signal circuits. Line voltage was made available for all three remote units. The second linear amplifier was located in an intermediate station between the reactor test cell and the control room.

The rest of the amplifiers and recording apparatus were located in the control room. A block diagram of the detection system showing the manner in which the components were connected is given in Figure 53. The gain controls of the first three 100-Mc amplifiers were set at a maximum and the fourth was set on linear position. The recording system shown in Figure 56 included an FM portable tape recorder system furnished by the Data Acquisition and Reduction center at Atomics International and a direct-record recording system. The only additional equipment, necessary for making the recordings, was a dual-channel oscilloscope, which monitored both the input and output of the recording system.

5. Experimental Procedure

After the detector system had been installed, the scintillation detectors were tested one at a time with the neutron source, to verify that they were

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Figure 58. Data Reduction System for High-Frequency Reactor Noise Measurements

operating properly. The appropriate reactor system and facility check list was completed by the reactor operator who then brought the reactor up to a low critical level, noting the critical top tank water level.

The operator of the recording system adjusted the gain of the differential amplifier such that the peak value of the noise signal at the input to the tape recorders was about ± 1.5 v (3 v peak-to-peak) on the oscilloscope. The gain positions of the amplifiers, power level, and reactor configuration were recorded.

During the recording session, the reactor operator was required to hold the power to within 10% of its value at the start of the recording. This was accomplished by occasionally adding top tank water to compensate for the loss of power.

Two recordings were made, using FM record networks, one at 60 ips for 12 min and the other at 7-1/2 ips for 50 min. A third recording was made at 60 ips for 12 min, using direct record networks. The tape and channel number of each recording was noted.

The recording made at 60 ips with direct record networks was played back at 7-1/2 ips and re-recorded at 60 ips, using FM record networks (this compressed the 80-kc bandwidth of the original direct recording to a 10-kc bandwidth which could be analyzed with the existing equipment).

A frequency response measurement of the count rate circuit and amplifier was performed over the frequency range from 10 cps to 80 kc, using the arrangement shown in Figure 55. The squared response was plotted as a function of frequency.

6. Analysis Procedure

The power spectral density function of the detected neutron fluctuations were obtained by passing the recorded signal through a narrow band pass filter and measuring the mean-square value at 64 different frequencies in the range from 10 to 80,000 cps. The recordings of the reactor "noise" signals were analyzed in three different frequency regions.

The recording made at 7-1/2 ips was analyzed at 60 ips over the effective frequency range from 10 cps to 1.2 kc. The recording made at 60 ips was analyzed in the frequency region from 500 cps to 10 kc, and the recording which was re-recorded after reducing the tape speed by a factor of eight was analyzed

over the effective frequency range from 5 to 80 kc. All of these signals appeared between 80 cps and 10 kc at the analyzer because of the time base changes with the tape recorders. A 20-cps filter and a 300-sec averaging time were used for all measurements. The data taken over the frequencies from 10 cps to 1.2 kc and 5 to 80 kc were normalized to the data in the midfrequency range from 500 cps to 10 kc. The power spectral density at the overlapping frequencies were used to obtain the normalization factors.

The power spectral density curve of the detected neutron fluctuations was corrected for the frequency response of the count rate circuit and differential amplifier by dividing the power spectral density data by the squared frequency response at each frequency.

7. High-Frequency Reactor Noise Studies

A simplified theoretical derivation of the spectral density, function of high-frequency reactor noise, is presented for a source-free, two-region (core and reflector), critical reactor with a detector in the reflector. The symbols used in the basic kinetic equations from Reference 8 are as follows:

k = neutron multiplication constant for infinite core.

β = total fraction of delayed neutrons.

P_c = probability of a neutron staying in the core.

P_r = probability that a disappearing reflector neutron will reappear in the core.

n_c = neutron density in the core.

n_r = neutron density in the reflector.

ℓ_r = total length of time that neutrons spend in the reflector.

ℓ_c = total length of time that neutrons spend in the core.

λ_i = decay constant for the i^{th} delayed neutron group.

C_i = concentration of the i^{th} delayed neutron group.

β_i = fraction of neutrons emitted by the i^{th} delayed neutron group.

The basic kinetic equations from Reference 8 are:

$$\frac{dn_c}{dt} = \frac{(1 - \beta) k P_c n_c}{\ell_c} + \frac{P_r n_r}{\ell_r} + P_c \sum \lambda_i C_i - \frac{n_c}{\ell_c}, \quad \dots (1)$$

$$\frac{dn_r}{dt} = \frac{(1 - \beta) k (1 - P_c) n_c}{\ell_c} + (1 - P_c) \sum \lambda_i C_i - \frac{n_r}{\ell_r}, \quad \dots (2)$$

$$\frac{dC_i}{dt} = \frac{\beta_i k n_c}{\ell_c} - \lambda_i C_i. \quad \dots (3)$$

The critical equation, found by setting all derivatives to zero, is:

$$k \left[P_c + (1 - P_c) P_r \right] = 1. \quad \dots (4)$$

8. Derivation of the High Frequency Transfer Function for a Critical Assembly Perturbed Only by Statistical Fluctuations in the Neutron Multiplication Constant (Reactor Noise)

Linearize Equations (1), (2), and (3) by replacing the time-varying quantities, n and k , with their equilibrium values plus a small time-dependent variation,

$$n = n_c + \Delta n, \quad k = k_o + \Delta k$$

Assume ΔP_r , ΔP_c , ΔC_i are equal to zero, neglect products of differentials, subtract steady components, and Laplace transform to obtain:

$$s \Delta n_c = - \frac{\Delta n_c}{\ell_c} + \frac{(1 - \beta)}{\ell_c} \left[P_c k_o \Delta n_c + P_{co} n_{co} \Delta k \right] + \frac{P_r \Delta n_r}{\ell_r} \quad \dots (5)$$

$$s \Delta n_r = - \frac{\Delta n_r}{\ell_r} + \frac{(1 - \beta)}{\ell_c} \left[(1 - P_{co}) (n_{co} \Delta k + k_o \Delta n_c) \right] \quad \dots (6)$$

$$\Delta n_r = \frac{\frac{(1 - \beta)}{l_c} \left[(1 - P_{co}) \left(\frac{\Delta k}{k_o} + \frac{\Delta n_c}{n_{co}} \right) \right] k_o n_{co}}{S + \frac{1}{l_r}} \quad \dots (7)$$

Then, substituting Equation (7) in (5), solving for $\Delta n_c / n_{co}$, applying the zero reference critical equation,

$$k_o P_{co} + k_o (1 - P_{co}) P_{ro} = 1,$$

and defining $\beta_7 \equiv k_o (1 - P_{co}) P_{ro}$ as the fraction of the neutrons appearing in the core which came from the reflector, obtain:

$$\frac{\Delta n_c}{n_{co}} = \frac{\frac{\beta}{l_c l_r} (1 - \beta) \left[(1 - \beta_7) l_r S + 1 \right] \frac{\Delta k}{\beta k_o}}{S^2 + S \left(\frac{\beta + \beta_7 - \beta \beta_7}{l_c} + \frac{1}{l_r} \right) + \frac{\beta}{l_r l_c}} \quad \dots (8)$$

Note that from Equations (2) and (3),

$$\frac{n_{ro}}{l_r} = \frac{n_{co} k_o (1 - P_{co})}{l_c} \quad \dots (9)$$

From Equations (7), (8) and (9) with $(1 - \beta) \cong 1$ and $\Delta k / \beta k_o = \Delta \rho$ in dollars,

$$\frac{\frac{\Delta n_r}{n_{ro}}}{\Delta \rho} = \frac{\beta}{l_r S + 1} + \frac{\beta \left[(1 - \beta_7) l_r S + 1 \right]}{l_r l_c (l_r S + 1) \left[S^2 + S \left(\frac{\beta + \beta_7 - \beta \beta_7}{l_c} + \frac{1}{l_r} \right) + \frac{\beta}{l_r l_c} \right]} \quad \dots (10)$$

Solving for the roots of the quadratic in the denominator to obtain $(S + \alpha_1)(S + \alpha_3)$, let

$$b = \frac{\beta + \beta_7 - \beta \beta_7}{l_c} + \frac{1}{l_r}$$

and

$$c = \frac{\beta}{l_r l_c}$$

and

$$S = -\frac{b}{2} \left[1 \pm \left(1 - \frac{4c}{b^2} \right)^{1/2} \right].$$

If $4c/b^2 \ll 1$, which is usually true, then:

$$S \cong \frac{b}{2} \left[1 \pm \left(1 - \frac{2c}{b^2} \right) \right] = -\frac{c}{b} \text{ and } -b + \frac{c}{b},$$

$$\alpha_1 \equiv \frac{c}{b}, \quad \alpha_3 \equiv b - \frac{c}{b}.$$

Therefore,

$$\alpha_1 = \frac{\beta}{(l_c + l_r \beta_7) + l_r \beta (1 - \beta_7)}. \quad \dots (11)$$

It is shown in Reference 8 that in order for the bare reactor kinetics equation to give the same results as the reflector reactor kinetics equation at low frequencies

$$l^* = l_c + l_r \beta_7.$$

Thus,

$$\alpha_1 = \frac{\beta}{l^* \left[1 + \frac{\beta l^*}{l_r} (1 - \beta_7) \right]}. \quad \dots (12)$$

The other root of the quadratic is:

$$\alpha_3 = \frac{\beta + \beta_7}{\ell_c} + \frac{1}{\ell_r} - \frac{\beta\beta_7}{\ell_c} - \alpha_1. \quad \dots (13)$$

Therefore, Equation (10) can be rewritten as:

$$\frac{\frac{\Delta n_r}{n_{ro}}}{\Delta \rho} = G(S) = \frac{\frac{\beta}{\ell_r}(S + \alpha_1)(S + \alpha_3) + \frac{\beta}{\ell_r \ell_c} \left[(1 - \beta_7)S + \frac{1}{\ell_r} \right]}{\left(S + \frac{1}{\ell_r} \right) (S + \alpha_1) (S + \alpha_3)}. \quad \dots (14)$$

Thus,

$$G(S) \sim \frac{(S + \alpha_4)(S + \alpha_5)}{(S + 1/\ell_r)(S + \alpha_1)(S + \alpha_3)}.$$

The equation for the high-frequency reflected reactor kinetics equations is obtained by converting from the Laplace to the Fourier transform to obtain:

$$G^2(\omega) \sim \frac{(\omega^2 + \alpha_4^2)(\omega^2 + \alpha_5^2)}{(\omega^2 + 1/\ell_r^2)(\omega^2 + \alpha_1^2)(\omega^2 + \alpha_3^2)}. \quad \dots (15)$$

The power spectral density of observed reactor noise, $S_y(\omega)$, is then (for a detector in the reflector):

$$S_y(\omega) \sim G^2(\omega) + A,$$

where A is a constant resulting from the detection process.

Thus,

$$S_y(\omega) = A + \frac{K(\omega^2 + \alpha_4^2)(\omega^2 + \alpha_5^2)}{(\omega^2 + \alpha_2^2)(\omega^2 + \alpha_1^2)(\omega^2 + \alpha_3^2)}, \quad \dots (16)$$

where:

$$\alpha_2 = \frac{1}{l_r},$$

$$\alpha_1 = \frac{\beta/l^*}{1 + \frac{\beta/l^*}{1/l_r}(1 - \beta_7)},$$

$$\alpha_3 = \frac{\beta + \beta_7 - \beta\beta_7}{l_c} + \frac{1}{l_r} - \alpha_1.$$