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REACTIVITY EFFECTS OF GADOLINIUM-155
ON A SNAP REACTOR

AEC Research and Development Report

Atoms International
North American Rockwell

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Canoga Park, California 91304

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**REACTIVITY EFFECTS OF GADOLINIUM-155
ON A SNAP REACTOR**

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SUMMARY

The effects of gadolinium-155 on both the reactivity and the temperature coefficient of reactivity of a SNAP reactor were measured. This study is reported here in three parts.

OPERATIONAL-ENVIRONMENT EXPERIMENTS

The SNAPTRAN-1 Reactor, a SNAP 10A reactor specially adapted for high-temperature and transient experiments, was used in the operational reactor phase of the gadolinium-155 experiments program. This reactor is capable of operating at temperatures up to 1400°F and of step reactivity insertions of more than \$1.50; accordingly it was well suited to this application. The reactor is fueled with 37 uranium/zirconium-hydride elements and is reflected with beryllium. The gadolinium-155 was mixed with a nickel powder to serve as a substrate, and rolled and formed into sleeves that were slipped over the individual fuel elements in the core.

The experimental program consisted of reactivity measurements at room temperature and elevated temperatures followed by a series of power transients producing significant temperature change to determine the prompt temperature coefficient. These measurements were repeated (with the exception of one prompt coefficient) for each of three concentrations of gadolinium-155 in the core. The reactivity loss with gadolinium addition was found to be an almost linear relationship with no observed saturation effect. The expected change in temperature coefficient was found, extrapolating to a value of zero or slightly positive prompt coefficient at the highest concentration of gadolinium ($\Sigma_a t = 0.23$). Under the operating limits of the reactor this configuration was not used in the transient temperature-coefficient measurements.

The transient experiments also showed that gadolinium-155 made the temperature coefficient less negative, and the amount of the effect increased with increasing temperature. Since the highest Gd concentration was not tested in the transient testing program, the results were inferred by extrapolation. In such case the prompt coefficient would become positive at about 1050°F.

The data are in good agreement with the SNAPTRAN-1 experiments at NRTS. They cannot, however, be compared directly with the isothermal coefficients, since isothermal measurements include grid and reflector effects, while prompt measurements exclude these but include spatial effects. Even so, the qualitative effects observed are in agreement.

WATER-IMMERSION TESTS

The SCA-4B critical assembly, used in all previous studies of water-immersed SNAP reactors, was used also in this water-immersion program. The same fuel, gridplates, and gadolinium sleeves as were used for Part I of this study were used in these tests. Reactivity measurements were made as a function of Gd concentration in bare water-immersed and beryllium-reflected water-immersed reactors.

The method of determining the reactivity of an assembly that is potentially 10 to 11 supercritical is a critical aspect of these measurements. In these experiments boron-filled aluminum tubes, referred to as "boron splines" were used to offset the excess reactivity. Their reactivity effect was determined by measuring the worth of one or a few splines at each characteristic location and summing these worths over the core. This method assumes no interaction between the splines. The reactivity was also deduced by measuring the reactivity effect of removing one or more of the Gd sleeves from each typical location and summing the worths over the whole core. The two methods gave consistent results for the reactivity effect of the gadolinium. It was found that a $\sum_a t$ of Gd^{155} equal to 0.217 was required to achieve a reactor that was just critical in water.

INTRODUCTION

The reactivity effect of gadolinium-155 in a SNAP reactor was investigated under the Aerospace Safety Program. One of the objectives of this program is the development of engineered safeguards that will assist in preventing inadvertent criticality upon water-immersion of a SNAP reactor after launch. Gadolinium-155 has a large resonance in its cross-section that preferentially poisons a water-immersed SNAP reactor more than one in its normal operating configuration. This program was planned to measure the critical factors affecting the use of gadolinium-155 in a SNAP reactor.

The program included measurements of the reactivity effect of Gd in water-immersed cores as well as in the operational core configuration, and measurements of the effect of Gd on the temperature coefficient. Because the Gd resonance lies below the median fission energy of a SNAP reactor, as the reactor temperature and hence the neutron energy increase, the effective cross-section of Gd decreases. This effect leads to a positive contribution to the reactor temperature coefficient and potential problems in the reactor-control characteristics.

The data developed through this program, although obtained on a SNAP 10A reactor, will be used to establish the accuracy of analytical models that will be used in the design of other SNAP reactors. In addition to the interest in this material as a means of achieving water-immersion subcriticality, it is also of interest as a long-term burnable poison. The reactivity data will be used to infer cross-section information which will be used to predict its burnup characteristics.

PART I: OPERATIONAL ENVIRONMENT EXPERIMENTS

I. DESCRIPTION OF THE TEST ENVIRONMENT

A. FACILITY

The operational environment tests were done in Test Cell No. 1 of the SNAP Environmental Test Facility (SETF), Bldg 024, located in the Atomic International Nuclear Field Laboratory (AINFL).

A detailed description of the SETF is given in Reference 1. The important feature of this facility, used in this portion of the test program, was the capability of purging the cell atmosphere with nitrogen from the existing LN_2 system and of performing all thermal and transient tests with less than 2% oxygen in the cell. Except for installation of a control console, instrumentation, and heater wiring the facility was operated without modification.

B. SNAPTRAN ASSEMBLY

The SNAPTRAN reactor is basically a SNAP 10A/2 reactor with certain ancillary features to permit operation at high temperatures and provide for rapid insertion of reactivity. It was used successfully in an extensive transient testing program at the National Reactor Testing Station (NRTS).^{(2)*} The core consists of 37 uranium/zirconium-hydride fuel elements fully enriched in uranium-235, mounted on gridplates with a lattice spacing of 1.27 in. The lattice spacing was increased from the standard SNAP core spacing of 1.26 in. to accommodate the gadolinium sleeves. It is reflected by beryllium in the radial direction, with control provided by four beryllium drums which vary the neutron leakage from the core. This arrangement is shown in Figure 1. Reactor characteristics are summarized in Table 1.

To provide for high-temperature operation the core and reflector are enclosed in an insulated box. The shafts of the control drums extend through this box to position the drive mechanisms in a low-temperature environment. Four 12.5-kw electrical heaters are mounted on the inside walls of the box to heat the reactor. During all high-temperature operations an atmosphere of argon was maintained in the thermal box by means of a feed line and supply bottle located in the access area. This was necessary to minimize nitriding of the beryllium. Figure 2 is a photograph of the SNAPTRAN machine.

*Superscript numbers in parentheses are for references at end of this report.

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Figure 1. SNAP 10A Reactor

TABLE 1
SNAPTRAN REACTOR CORE COMPOSITION

Fuel Elements	
Number	37
Lattice spacing, in.	1.27
Fuel diameter, in.	1.212
Cladding material	Hastelloy-N
Cladding diameter OD, in.	1.25
Cladding thickness, in.	0.015
Fuel alloy, wt % uranium in zirconium	10
Degree of hydriding, N_H , 10^{22} at./cc	6.5
Active fuel length, in.	12.25
Internal Reflectors	
Composition	Beryllium
Cladding	None
External Reflectors	
Composition	Beryllium
Cladding	None
Nominal thickness, in.	1.875
Reflector Control Elements	
Number	4
Material	Beryllium
Nominal thickness, in.	1.875 minimum 3.629 maximum
Gridplates	
Material	304 stainless steel
Thickness, in.	0.188
Core Vessel	
Material	316 stainless steel
Internal diameter, in.	8.875
Thickness, in.	0.032
Maximum Uranium Loading, kg uranium-235	4.75
Median Fission Energy, ev	0.18

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Figure 2. SNAPTRAN Assembly

Since the normal control-drum-drive speeds are not adequate to start prompt transients an additional drive mechanism is provided to permit rapid insertion of one or two control drums. This motion, from 45° out to the full-in position, is provided by a rack-and-pinion drive from a pneumatic cylinder. This drive mechanism is separate from the normal speed drive and is controlled by a separate portion of the reactor control system. Actual transient operations were begun by a sequence timer which removed the tooth clutch clamps, fired two drums in, vented the gas pressure, and scrambled all four drums to end the transient. Each of these operations was controlled by a preset timer to maintain flexibility of operation for the various transients. Typical drum step-in times of 220 msec (including relay closures) were measured with drum scram times of 250 msec.

C. DATA ACQUISITION

1. Steady-State Nuclear Instrumentation

In measurements near delayed critical (from critical to approximately 30¢ supercritical), data on reactor power level and power-level rate of change were obtained from conventional boron-lined ionization chamber and fission-counter detectors. Five channels of nuclear-safety instrumentation were used as listed in Table 2.

TABLE 2
NUCLEAR-SAFETY INSTRUMENTATION

Quantity	Detector	Function
2	Fission counter	Log countrate/period
1	Ion chamber	Log -N/period
2	Ion chamber	Power level

In addition to these channels the reactor operator had one consisting of an ionization chamber and a range-switching picoammeter which was used to monitor the flux level and establish steady-state conditions.

Reactivity measurements were made by a separate channel in which an ion-chamber signal was amplified by a picoammeter and converted to a variable

frequency by a voltage-to-frequency converter. This frequency was counted on an electronic counter to provide a digital signal which was proportional to the reactor flux. A small digital computer (Computer Automation 816 with a 4k memory) was programmed to solve the inverse kinetics equation using the Code RHO.^(3,4) The calculation was made in real time so that a printout of the reactivity was obtained every 4 sec. This equipment was used to perform control-drum calibrations in the course of the program.

2. Reactor Transient Measurements

The reactor power level during the transient tests was measured by means of fast-neutron scintillation detectors. These detectors are basically identical to those used previously in the SNAPTRAN Program.⁽⁵⁾ The scintillator was a cylinder of NE-102 plastic, 2-3/4-in. diam by 2-in. long. Two types of phototubes were used to detect the scintillations. For the more sensitive ranges (less than 100-kw reactor power) a 6324-A photomultiplier was used, while for the high-power levels an ITT Type-FW-114 photodiode was used. The photomultipliers were operated with 600 volts across the dynode string, and the photodiodes were operated with a 90-volt bias battery to collect the photoelectrons. The detectors were housed in a lead cave, 4-in. thick, covered with boral and located about 14 ft from the center of the reactor. The photomultiplier detectors had a sensitivity of about 2×10^{-9} amp/watt and the photodiodes had a sensitivity of 1.5×10^{-12} amp/watt.

The currents from these detectors were amplified by field-effect-transistor (FET) input solid-state amplifiers (Analog Devices 142B) to a standard level of 1 volt. The sensitivity of the different channels was adjusted by means of plug-in feedback resistors to achieve overlapping coverage of the power transient from about 1 kw full-scale on the most sensitive range to 1000 Mw full-scale on the least sensitive range. A total of 12 channels of power data were used in this system.

3. Temperature Measurements

Temperatures during the transients were measured by means of special thermocouple assemblies called energy probes (EP). These consist of a very small piece of uranium/zirconium-hydride fuel material (~0.062 by 0.062 by 0.157 in.) to which 0.5-mil chromel and alumel leads are attached. The assembly is insulated with magnesium oxide and placed in a sheath. The leads are

extended and the signal connected to a differential input amplifier (Analog Devices 142B). The outputs of up to 12 EP's were used to obtain data on the power distribution in the reactor. Since the material in the EP's is identical to the reactor fuel, the temperature rise in the probe on a fast transient corresponds to that of the adjacent fuel.

The amplified signals from the nuclear detectors and the EP's along with drum position and timing signals were recorded as analog signals on magnetic tape. Two 14-channel instrumentation recorders (CEL Model 3300) were used in the FM mode at 60 ips with a 54-kHz center frequency. This provided a 10-kHz bandwidth which was quite adequate for the transients. The timing signal was a 100-kHz signal which was switched on concurrently with the "fire" signal from the transient timer. This signal was used as both a reference mark for synchronizing the channels and a time base in digitizing the data.

An eight-channel oscillograph recorder (Brush Mark 200) was used for on-line monitoring of the transients. All the data on the magnetic tapes were played back for recording on the oscillograph so that they could be examined before proceeding with the test series.

The analysis of the data from the experiment was made using the STAC Code on an IBM 360-50 computer. Data from the transient were input to the code by means of digital magnetic tape. These tapes were prepared by a digitizing computer. The block diagram of this system is shown in Figure 3. The timing signal from the analog tape is sent to the digitizing control unit which controls a gate on the signal channel carrying the FM data from the tape. The digitizing control permits selection of a delay time between the fire command and the start of the digitizing as well as the time interval between data points. The FM signal is converted into a pulse train of twice the repetition rate of the incoming frequency and passed through the gate of the digitizing control. Pulse trains of 108 to 151 kHz arrive at the frequency counter (General Radio Model 1191) which is operated in the period-measurement mode. By counting the pulse output of its internal 10-MHz clock during the time required for the arrival of 10, 100, or 1000 input pulses, the counter is able to digitize the incoming frequency to an accuracy always better than 0.5%. Data points as closely spaced as 120 μ sec are used routinely. The accuracy and the data rate are inversely related so that better accuracy is achieved with lower digitizing rates.

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Figure 3. Data-Acquisition System

The digital data is input to a small computer (Computer Automation 816) which stores the data in its memory. The operation is repeated to obtain a total of 3,000 data points on a single channel. The data, together with header information which identifies the transient, the detector, the amplification, and the time-base parameters are placed in a suitable format and reloaded on digital magnetic tape by the computer. This process is repeated for each of the data channels on the analog tape, producing a digital record of the transient that can be processed using the STAC code.

II. EXPERIMENTAL PROGRAM

A. FUEL LOADING

The SNAPTRAN core vessel was loaded initially with 37 lucite dummy elements. These elements were then replaced with fuel elements, without nickel sleeves, in a stepwise manner using the usual inverse multiplication technique until a critical configuration was achieved. The remaining lucite elements were replaced with fuel elements one at a time until all 37 were loaded. No Be shims were on the reflector or drums. The SNAPTRAN fuel elements were then removed, the 1.27-in. (lattice-spacing) gridplates replaced with the 1.26-in. gridplates, and the core reloaded with 36 elements. The last element was not loaded since the indications from the $1/M$ curve were that criticality might be achieved on the second drum. The core was again unloaded and reloaded with 37 SCA-4C fuel elements on the 1.26-in. gridplates. After determining the critical drum configuration these elements were removed and the SNAPTRAN fuel elements with Ni sleeves were reloaded in the core vessel using the 1.27-in. gridplates. Subsequent loadings with Gd sleeves on the elements were completed in one step after calculation of the amount of drum and/or reflector shim required to provide the necessary excess reactivity.

B. DRUM CALIBRATIONS

Drums B and C were used for all excess-reactivity measurements and transient-reactivity insertions; therefore they were calibrated for each sleeve and/or shim configuration. Drums A and D acted as "safety" elements and were calibrated with the Ni sleeve configuration only. The calibration curves for drums A and D were substantially the same as those for B and C.

The calibration procedure consisted of positioning Drum B all the way in, Drum C full-out, with A and D banked at the critical position. Drum C was then inserted to a position which would result in a 30 to 60-sec period. When sufficient data had been taken, B was moved out to establish a negative 30-sec period. This process was repeated until Drum B was full-out and C was inserted to the full-in position. Both the positive and negative period data were fed to an on-line computer which continuously computed the reactivity changes taking place. (See Section I-C for details.) These reactivity increments were then compiled to produce an integral worth curve for each of the 2 drums.

C. FLUX-DISTRIBUTION MEASUREMENTS

The relative radial- and axial-flux distributions were measured for the Type-I sleeve configuration using 1/4-in.-diam by 0.005-in.-thick gold foils. The foils were attached to 1/4-in.-wide by 0.020-in.-thick aluminum strips which were inserted in the interstices of the core. The axial traverses were measured adjacent to the central element (interstices 5, 10, and 14). The radial traverses were made by placing foils at the midplane of the core at several radial positions. The foils were counted on standard scintillation-type β -counters.

By suitable weighting and graphical integration the average relative flux in the core was determined. The average absolute flux was obtained by comparing the activities of two Al - 18% U (93.11% U^{235}) foils irradiated in the SNAPTRAN core with two identical foils irradiated in a known thermal flux for the same length of time in the AE-6 reactor. The flux determined in this manner is an equivalent 2200 meter/sec thermal flux and is not the true flux in the core. However, the number of fissions per second occurring in the core can be calculated from this equivalent flux, the 2200 meter/sec macroscopic fission cross-section of U^{235} , and the volume of the core. The power level at which the flux mapping was done can then be obtained from the conversion factor, 3.3×10^{10} fissions/watt-sec. This power determination was used to calibrate all the neutron detectors. The calibration factors were later refined by applying the results of the energy-probe measurements made during the prompt transient tests.

D. CONFIGURATIONS

Gadolinium-155 in the form of Gd_2O_3 was incorporated in a Ni matrix and rolled into foils of 2-mil thickness. These were cut, formed into sleeves, and spot-welded so that they could be slipped over each fuel element. Sleeves were restrained from vertical movement in either direction by the upper and lower gridplates. The four types of sleeves fabricated for use in this program are given in Table 3.

Six test configurations were constructed using these sleeves: four contained 37 sleeves of one gadolinium concentration, one was a mixed loading, and the

TABLE 3
GADOLINIUM SLEEVES

Type	Gadolinium Concentration, $\Sigma_a t^*$
I	Nickel only
II	0.0730 ± 0.0058 (2σ deviation)
III	0.1499 ± 0.0016
IV	0.2298 ± 0.0209

other a central-void configuration. The mixed loading consisted of Type-IV sleeves on the inner 19 elements with Type-I sleeves on the outer 18 elements. The central-void configuration used Type-I sleeves on 36 elements with the central element removed and a Type-IV sleeve installed in that location.

During the central-void configuration tests, measurements of the worths of aluminum and steel placed in the central void were conducted for the purpose of checking streaming calculations. Rods of aluminum and cold-rolled steel, 12-in. long and 1/2, 7/8, and 1-1/4 in. in diameter, were placed in the central position in the Type-IV sleeve. The worth of each rod with respect to a void was then determined from the change in critical drum position.

As the amount of Gd in the core increased, additional Be reflection was required to provide the necessary excess reactivity for the transient and thermal tests. This was accomplished by adding 1/8-in. shim-plates to the flat surfaces of the reflector block and to the flat surface of each drum. The following procedure was then used for changing sleeve configurations:

- 1) The shim configuration was adjusted so that sufficient excess reactivity would be available to obtain a critical configuration with both the present and succeeding sleeve configurations.
- 2) Drums B and C were calibrated with each sleeve and shim configuration.
- 3) Excess reactivities of each configuration were obtained from critical drum positions.
- 4) Sleeve worths were then calculated from the difference in excess reactivities.

*Based on $\sigma_a \text{ Gd}^{155}$ (2200 m/s) = 58,000 b

E. THERMAL TESTS

Isothermal temperature-coefficient measurements were made with each sleeve configuration. In each case the following procedure was used: (1) The reactor was brought to a source-free power level of approximately 10 watts at room temperature and the critical drum configuration noted; (2) The electrical heaters were then activated and the system temperature raised to 300°F; (3) Heater power was reduced to achieve a uniform steady-state system temperature at this point. This process was repeated at nominal system temperatures of 600, 900, and 1200°F. The reactor was maintained critical throughout the experiment and drum configurations noted periodically. Chromel-Alumel thermocouples were placed throughout the assembly as indicated in Table 4. Temperatures were printed out on a data-logger typewriter and multipoint recorder.

TABLE 4
THERMAL-TEST THERMOCOUPLE SCHEDULE

Location	Number of Thermocouples
Reflector	8
Door heaters	4
Core interstices	28
Upper and lower gridplates	4 each
Internal beryllium	2
Control-drum towers and bases	1 each
Core-vessel/reflector interface	12
Argon environment	4

F. TRANSIENT TESTS

Following the isothermal temperature coefficient measurements, a series of transient tests was made with each sleeve configuration. A typical series started with a 60¢ transient from 75°F followed the next day by an 80¢ transient from ~100°F. The third day a series of cascaded \$1.20 transients was made as follows: a \$1.20 transient was performed at a system temperature of 75 to 100°F. Transient scram times were so selected that an average core-temperature rise

of at least 300°F was obtained. The heaters were then activated and the entire system brought to an isothermal temperature of 300°F. The next \$1.20 transient was made at this temperature. This procedure was repeated at 600 and 900°F for a total of four \$1.20 transients.

All transients used both B and C drums in the coupled mode (simultaneous insertion) in the following manner:

- 1) B and C drums were set to a position where subsequent insertion to 0° would cause the desired excess reactivity.
- 2) The reactor was brought to a source-free critical level using drums A and D.
- 3) Drums B and C were withdrawn to 45° and the flux level allowed to drop several decades.
- 4) The console was switched to pulse operation, tooth clutch clamps were applied, and B and C drums charged.
- 5) Tape recorders were started.
- 6) The sequence timer was started.

The timer would then automatically release the clamps, fire B and C into 0°, vent, and scram all drums on the preset timing schedule.

Figure 4. Axial-Flux Distribution

III. RESULTS

A. FUEL LOADING vs REACTIVITY

On the initial loading of SNAPTRAN elements, criticality was achieved on the 32nd element with an excess reactivity of approximately 53¢. With 37 elements loaded, the excess was \$3.95 with the 1.27-in. lattice-spacing gridplates. Reloading these fuel elements on the 1.26-in. gridplates produced an excess reactivity of \$3.80 with 36 elements. Since the 37th element would be worth approximately \$0.69 as determined from the average of the last five elements loaded, a total excess of \$4.49 is indicated for a fully loaded core. Thus, decreasing the lattice spacing from 1.27 to 1.26 in. is worth \$0.54. An excess reactivity of \$3.98 was measured for the SCA-4C elements on the 1.26-in. gridplates. This \$0.54 difference in reactivity was attributed in part to the difference in hydrogen content of the two fuel loadings (N_H of 6.54 for SNAPTRAN vs 6.40 for SCA-4C).

Excess reactivities for the gadolinium-sleeve configurations with the shim configurations used for the thermal and transient test are given in Table 5a and 5b, respectively.

B. FLUX DISTRIBUTIONS

The results of the gold-foil flux-distribution measurements are given in Figures 4 and 5. No Be shims were installed on the reactor during these measurements, and Ni sleeves without Gd were in place. Data points were normalized to the maximum value for the axial measurements and to the central foil for the radial measurements. The actual measurements are probably accurate within $\pm 1\%$, but hexagonal array and drum effects are probably larger effects than that.

C. DRUM CALIBRATIONS

A total of 34 drum calibrations were made during the course of this program. All four drums were calibrated at ambient temperature in the Ni-sleeve configuration. In addition, in this configuration Drums B and C were calibrated

TABLE 5
TEST CONFIGURATIONS

a. Thermal			
Sleeve	Shim Thickness (in.)		Excess Reactivity (\$)
	Drum	Reflector	
I	None	None	3.95
II	1/4	1/4 on 2 flats	2.05
III	1-7/8	1/4 on 2 flats	1.22
IV	1-7/8	1-1/4 on 4 flats	1.43
Central void Mixed loading	1-7/8	1/4 on 2 flats	3.15

b. Transient			
I	None	None	3.83
II	None	1-1/4 on 4 flats	2.92
III	1/2	1-1/4 on 4 flats	2.07
Central void	None	1-1/4 on 4 flats	2.50
Mixed loading	1/2	1-1/4 on 4 flats	2.93

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Figure 5. Radial-Flux Distribution
(Type-I Sleeve Configuration, SNAPTRAN Assembly)

at 280, 500, 800, and 1200°F average system temperatures. No change in the shape of the integral worth curve or variation in total worth was detected over this temperature region.

Drums B and C were also calibrated for each shim and sleeve configuration. The addition of Be to the reflector block and drum noticeably altered the shape of the integral-worth curve and the total worth of the drum. Three Drum-C calibration curves, representing the extreme variation from the no-shim case, are shown in Figure 6.

Calibration curves for Drums B, A, and D are shown in Figure 7. They were taken during the Ni-sleeve configuration with no shims on the reflector or drums. During the course of the thermal testing with the Type-II sleeves the drive shaft on Drum B twisted about 8°; full-in is thus indicated as 8° instead of 0°. Since, for transient tests, only that portion of the drum from "step" to "full-in" is of interest, the calibration curve is plotted as having zero worth at 8°. The curves showed a repeatability of about $\pm 2\%$ per drum.

D. WORTH MEASUREMENTS

The total worths of the various concentrations and combinations of Gd-Ni sleeves were measured in the SNAPTRAN assembly. The results of these measurements are given in a. of Table 6 and are all relative to the Ni-sleeve configuration. The worth of each of the aluminum and steel rods in the central position in the Type-IV sleeve is presented in b. of Table 6. These worths are all with respect to a void.

The reactivity effect as a function of $\Sigma_a t$ for Gd is shown in Figure 8. Again the worths shown are relative to $\Sigma_a t = 0$ (Ni only). The value of $\Sigma_a t = 0.118$ for the mixed loading was obtained from an arithmetical average of the $\Sigma_a t$ for 19 Type-IV and 18 Type-I sleeves. No weighting factors were used.

E. THERMAL-TEST RESULTS

Isothermal temperature coefficients were measured for each of the configurations shown in Table 5a up to system temperatures of 1150°F. The reactivity loss at nominal system temperatures of 400, 600, 900, and 1150°F relative to ambient temperature operation is shown graphically in Figure 9. The variation of temperature defect at 1200°F with gadolinium concentration is shown in Figure 10.

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Figure 6. Drum-C Integral-Worth Curves

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Figure 7. Drums A, B, and D Integral-Worth Curves
(Type-I Sleeves, No Shims)

TABLE 6
REACTIVITY WORTHS

a. Sleeve		
Sleeve Type	Worth Relative to Nickel (\$)	
II	-2.71	
III	-5.04	
IV	-7.29	
Mixed Loading	-4.20	
Central Void	-3.13*	

b. Aluminum and Steel Worthy		
Rod Diameter (in.)	Worth (ϕ)	
	Aluminum	Steel
1/2	+6.2	+9
7/8	+18.5	+17.5
1-1/4	+43.5	+24

*From measurements reported in Reference 7, void worth is -\$2.90. Therefore, the Type-IV central sleeve is worth ~ -\$0.23.

The value of the temperature coefficient for each configuration at various system temperatures is given in Table 7. This value was obtained by calculating the slope of the reactivity-loss curve (Figure 9) at each specified temperature.

The temperature uniformity was about $\pm 20^{\circ}\text{F}$ but the error in the mean temperature is probably smaller, perhaps $\pm 15^{\circ}\text{F}$. The error in reactivity differentials is largely due to drum-calibration uncertainty, which we might estimate at 2% maximum. These indicate that a slope taken over a 300°F span and involving a reactivity change of 60ϕ would be in error by $\pm 0.014\phi/^{\circ}\text{F}$. We might estimate that all the values are of about that accuracy. A plot of coefficient vs Gd loading, Figure 11, shows that accuracy of $\pm 0.014\phi/^{\circ}\text{F}$ is quite credible, insofar as consistency is concerned.

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Figure 8. Reactivity Effect of Gadolinium Sleeves

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Figure 9. Isothermal Temperature Effect on Reactivity

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Figure 10. Variation of Temperature Defect at 1200°F with Gadolinium Concentration

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Figure 11. Isothermal Temperature Coefficient for Various Gadolinium Loadings

TABLE 7
VARIATION OF TEMPERATURE COEFFICIENT WITH TEMPERATURE

Temperature (°F)	Gadolinium Concentration, Σ_a^t (ϕ /°F)					
	I Nickel	Central Void	II 0.073	Mixed 0.118*	III 0.15	IV 0.23
150	-0.12	-0.12	-0.11	-0.10	-0.10	-0.04
300	-0.17	-0.16	-0.15	-0.12	-0.10	-0.07
600	-0.24	-0.22	-0.20	-0.16	-0.12	-0.08
900	-0.28	-0.24	-0.21	-0.17	-0.16	-0.11
1200	-0.29	-0.25	-0.22	-0.21	-0.17	-0.14

*Arithmetical average

F. TRANSIENT-TEST RESULTS

A summary of the initial conditions for each of the transient tests made is shown in Table 8. A detailed tabulation and analysis of the results are given later in this report.

TABLE 8
TRANSIENT-TEST SUMMARY

Test No.	Date	Sleeve Configuration	Shim Configuration		Planned Reactivity Insertion	Initial Temperature (°F)	Comments
			Drum	Fixed			
1	10-29-68	Type I	None	None	60¢	80	Scrammed before peak.
2	30-68				66¢	90	
3	31-68				80¢	100	
4	11-1-68				\$1.05	120	
5	4-68	Aborted: Failure in B-drum charging system.					
6	6-68	Type I	None	None	\$1.20	80	
7	7-68				\$1.30	100	
8	3-13-69				18¢	-	Drum-calibration check.
9	13-69				60¢	73	
10	14-69				80¢	98	Detector amplifiers not operating properly.
11	17-69				\$1.20	73	Timing signals not recorded on FM tape.
12	17-69					366	
13	17-69					609	
14	17-69					956	
15	20-69					82	High background noise on tape recording.
16	20-69					350	
17	20-69					560	
18	20-69					860	
19	31-69	Type I + Void		1-1/4 in. on 4 flats	60¢	74	
20	31-69				80¢	210	
21	4-1-69				\$1.20	127	
22	1-69					320	
23	1-69					601	
24	11-69	Type II	None	None	60¢	72	Detector calibration run. Scrammed before peak.
25	15-69				60¢	77	
26	15-69				60¢	230	
27	16-69				80¢	153	
28	17-69				\$1.20	118	
29	18-69				\$1.20	263	Scrammed before peak.
30	18-69				\$1.20	617	
31	18-69				\$1.15	930	
32	21-69				\$1.20	78	Scrammed before peak.
33	21-69				\$1.20	125	
34	29-69	I, IV	1/2 in.	1-1/4 in. on 4 flats	60¢	73	
35	29-69				60¢	370	
36	30-69				80¢	215	
37	5-1-69				\$1.20	138	
38	1-69				\$1.20	350	
39	1-69				\$1.20	620	
40	13-69	III	1/2 in.	1-1/4 in. on 4 flats	60¢	72	
41	14-69				60¢	103	
42	14-69				60¢	270	Scrammed before peak.
43	19-69				50¢	75	
44	19-69					350	
45	23-69					520	Scrammed to limit average temperature rise to 250°F.
46	23-69					870	
47	6-3-69	I	None	None		70	
48	4-69				60¢	85	
49	5-69				80¢	95	Scrammed before peak.
50	6-69				\$1.20	85	
51	6-69					350	
52	6-69					620	
53	6-69					860	

IV. ANALYSIS OF THE CRITICAL EXPERIMENTS

A. PREDICTIONS OF GADOLINIUM-155 WORTH

The initial predictions of Gd^{155} worth in the SNAPTRAN environment were made in 1967. Between the time of these early calculations and the firm definition of the experimental program several geometric and material composition modifications were made. The SNAPTRAN gridplate lattice was changed from 1.26 to 1.27 in. to permit the ready insertion of the Ni-Gd sleeves. This led to a reduction in reactivity of the SNAPTRAN assembly. (The calculated worth of this modification was $-\$0.80$. Subsequent measurement gave a value of $-\$0.54$.) The concentrations of the foil poisons were reduced from $\Sigma_a t^*$ values of 0.0, 0.1, 0.2, 0.3 to 0.0, 0.067, 0.133, 0.200. This was done to assure that measurements of Gd^{155} could be made in a critical configuration (available excess was $\sim \$8$) as well as to minimize the possibility of experiencing a positive prompt temperature coefficient for any of the foil sets.

With these changes made to the program, revised predictions of Gd^{155} worth in the SNAPTRAN-2 assembly were made. The then current 26-group library was used to make the calculations. Table 9 summarizes the calculations.

TABLE 9
PREDICTED WORTH OF GADOLINIUM-155
FOILS IN SNAPTRAN-1

Foil Type	$\Sigma_a t$	Worth (\$)
I	0.0	Reference
II	0.067	-3.00
III	0.133	-5.05
IV	0.200	-6.92

*Based on a Gd^{155} (2200 m/s) = 58,000 b

B. POST-EXPERIMENT ANALYSIS

The measured worths of the Gd sleeves showed good agreement with the predicted worth. However, it was found in analyzing the SCA-4B experiments (see Part II) that the equivalent $\Sigma_a t$ values of the fabricated foils were higher than those used in the experiment. When the actual $\Sigma_a t$ was used the agreement between prediction and measurement was not as close.

Since the initial predictions were made a new cross-section library had been developed for the SNAP reactor program. As with the SCA-4B program, the calculations were repeated using this library. Independent checking of the library with the SNAP critical assemblies had shown good agreement between calculation and measurement. The foil $\Sigma_a t$ was not changed, so that differences between libraries could be observed in both the SNAPTRAN and SCA-4B environments. (The old 26-group cross-section library, although good in determining worths, was poor in predicting criticality. Normalization of several dollars was required to obtain agreement between calculation and experiment.)

The reactivity of the SNAPTRAN machine was calculated using the new library as \$1.84. This value did not include the worth of the insulated heater box, which was taken to be +\$0.50 based on earlier measurements. As there is a significant gap between the box and the critical machine, difficulty was experienced when attempts were made to calculate its worth. Previous calculations had indicated a normalization of +\$1.00 for dry critical SNAP assemblies. Adding these factors, an excess reactivity of \$3.34 was obtained for the SNAPTRAN-1 configuration with Ni sleeves about the fuel. The measured excess of this configuration was \$3.95. The small difference between normalized calculation and measurement (~\$0.60) is most probably due to difficulty associated with the cylindrical representation of the octagonal reflector. In addition, streaming associated with the dry-reactor configuration adds a margin of uncertainty in the buckling representation.

The foil worths calculated with the 28-group cross-section library are given in Table 10. As indicated in the table no significant change occurred from the pre-experiment calculations.

TABLE 10
REVISED GADOLINIUM-155 WORTHS
IN SNAPTRAN

Foil Type	$\Sigma_a t$	Worth (\$)
I	0.0	Reference
II	0.067	3.05
III	0.133	5.35
IV	0.200	7.35

Figure 12 gives the experimental results and calculated worths. It was determined that the experimental data could be revised \$0.10 upward in worth because of the normalization used in referencing the sleeve worth to the Type-I Ni sleeves. The modified experimental results with calculated results are shown in Figure 13. Although the agreement has improved it is not as good as that obtained with the SCA-4B data.

The reason for the discrepancy may be in the modeling of the void region in the core for the cell calculations of the Gd sleeves. Voids are difficult to accommodate with the present computational techniques. Streaming may not have been properly treated, resulting in the observed deviations.

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Figure 12. Calculated and Measured Worth of Gadolinium-155 in SNAPTRAN

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Figure 13. Calculated and Measured Worth of Gadolinium-155 in SCA-4B and SNAPTRAN

V. ANALYSIS OF TRANSIENT EXPERIMENTS

A. DATA-REDUCTION SYSTEM

1. Processing Description

Data describing power, temperature, and control-drum position during each transient were recorded on FM magnetic tape and then converted to a digital magnetic tape. This process was described in Section I-C. The tapes were subsequently processed by the STAC kinetics-analysis program. The results were computer-plotted graphs of normalized power, mean temperature, and temperature coefficient of reactivity. The results for the various configurations were then compared to show the effect of gadolinium-155 on the kinetics of the reactor.

2. The STAC Processing Code

The STAC code was designed to accomplish these objectives:

- 1) Provide a facility for managing, translating, plotting, and listing raw-data tapes from the Analog-to-Digital Converter (ADC).
- 2) Combine and analyze the data to determine kinetic performance.
- 3) Determine a temperature coefficient of reactivity.

The code consists of four independent sections which can be applied independently. Intermediate data are stored in re-usable interface data sets.

Section 0 provides a means of copying the 7-track raw-data tapes produced by the ADC onto a 9-track master data tape. A listing can be made at the same time. Data channels having unreadable records are excluded from the new copy. An alternate routine allows a synthetic-data tape to be constructed from punch-card input. This feature is valuable in checking the code. Noise and digitizing effects are simulated in the synthetic-tape routine.

Section 1 converts each channel of data from the symbolic format used on the master data tape to the appropriate binary representation in engineering units, identifies the data from an input deck of ID cards, provides edits and plots as requested, and stores the converted data on a storage-data set.

Words containing illegal characters are replaced with 0. If an illegal character occurs within the heading information section which identifies the data, the entire channel is rejected.

Section 2 scans the storage-data set, storing neutron-detector (power) data in the magnetic core memory. "Wild" points are replaced with data based on a polynomial fit to neighboring data. Then the data are converted to log space. Each channel of data has a "maximum valid indication" associated with it, above which the data are considered to be invalid. Each channel except the last has a factor, k_n , calculated for it which makes its log-overlap area match the overlap area of the channel above it:

$$\int_{T_{0.8,n}}^{T_{1.0,n}} \ln [k_n P_n(t)] dt = \int_{T_{0.8,n}}^{T_{1.0,n}} \ln [P_{n+1}(t)] dt$$

where

P_n = power indication, n^{th} channel,

$T_{0.8,n}$ = time when P_n reaches $0.8 \times P_n$ maximum,

$T_{1.0,n}$ = time when P_n reaches $1.0 \times P_n$ maximum, and

P_n^{max} = maximum valid indication for channel n .

These factors are calculated and applied to each channel in turn. A composite is then made using data from the first channel up to $T_{1.0,1}$ and then from the second channel from $T_{1.0,1}$ to $T_{1.0,2}$, etc, until a continuous composite power curve is made. The noise and small joining roughness are removed by a smoothing technique, in which a smooth curve, continuous in its first derivative, is passed through the data.⁽⁸⁾ The logarithmic derivative of the data is obtained as a byproduct. The smooth curve is then converted back to linear space and integrated to give energy deposition.

Reactivity is calculated using the integral kinetics equations⁽⁸⁾ and the point-wise logarithmic derivative obtained from the smoothing process. A reactivity value is also calculated for each channel of data by fitting a straight line to it in log space, and then using the slope of this line in the familiar "inhour equation." This is useful in obtaining very accurate values of inserted reactivity from the unsmoothed data.

The power, logarithmic derivative (inverse period), integrated power, and reactivity are written on an interface data set for later use. Edits and plots are given if requested.

The storage data set is scanned again, and temperature data from the energy probe are stored in memory. "Wild" points are replaced with data based on a polynomial fit to neighboring points, and then the initial temperature, T_0 , is determined by averaging the first few data points of all channels. Next the energy deposition in each energy probe is calculated using the integral of the heat capacity for the zirconium/uranium-hydride fuel material. The flux depression in each probe is accounted for at this point by dividing each energy deposition by the calculated ratio of power density in the probe to that in the fuel. Finally, a mean energy is calculated at each point by a least-squares fitting technique described in NAA-SR-11850.⁽⁸⁾ In this technique a precalculated power profile is fitted to all the probe data, i.e., $\bar{E}_1$ is so chosen as to minimize:

$$R_i = \sum_n (E_{n,i} - \bar{E}_i \psi_{r,n} \psi_{z,n})^2$$

where

R_i = fitting residual at i^{th} time point,

$\bar{E}_i$ = mean energy at i^{th} time point,

$E_{n,i}$ = n^{th} probe response at i^{th} time point,

$\psi_{r,n}$ = radial power profile at n^{th} position, normalized to 1.0 average, and

$\psi_{z,n}$ = axial power profile at n^{th} position, normalized to 1.0 average.

The resulting curve of $\bar{E}$ -vs- t is smoothed, and the mean temperature is calculated from the mean energy using the heat-capacity integral. The smoothing method is the same as that used for power. These data are then stored in the interface-data set. Edits and plots are given if requested.

During the least-square fitting a consistency check is made. After $\bar{E}_1$ is calculated the relationship of each channel of data to the mean is examined by calculating C_n to minimize:

$$R_n = \sum_i \left(\frac{E_{n,i}}{\psi_{r,n} \psi_{z,n}} - C_n \bar{E}_i \right)^2$$

and

$$S_n = \sqrt{R_n / (n - 1)}$$

where

R_n = fitting residual for n^{th} channel,

C_n = calibration factor for n^{th} channel, and

S_n = standard deviation for n^{th} channel.

The C_n 's ran typically between 0.9 and 1.1. A value of C_n not in the range 0.8 to 1.2 indicated a misplaced or defective probe and was cause for rejection of the data. Values of S_n were usually 2% or less until a probe became defective.

Section 3 reads the interface data set into memory. Then a power-calibration factor, P_{cal} , is calculated which minimizes

$$R_P = \sum_i (I_i - P_{\text{cal}} \bar{E}_i)^2$$

where

R_P = fitting residual,

I_i = integrated power, i^{th} time point, and

P_{cal} = power-calibration factor.

The factor is applied to both the power and integrated power data. The temperature which would result from the deposition of the calibrated integrated power is calculated from the heat-capacity integral, and all of the results obtained up to that point are edited together.

The temperature coefficient is then evaluated by calculating $d\rho/dT$. The temperature from $\bar{E}_i$ is used first, and then the calculations are repeated using the temperature calculated from I_i/P_{cal} .

Three methods of evaluation are used. In the first, a numerical differentiation is performed by fitting a straight line to the ten nearest neighbors of

each point. Then a single straight line is fitted over the entire range of usable data. Finally, a straight line is fitted to the data falling within the time interval over which the power is above half its peak value. Edits and plots are given.

Examples of the plots produced are shown in a later section.

B. ANALYSIS OF THE DATA

1. Neutronics Parameters

To apply the "backward kinetics" calculation the kinetics parameters Λ (prompt-neutron generation time) and β_{eff} (effective delayed-neutron fraction) were required for each configuration. These were calculated as described in Reference 8 except that AI's 28-group cross-section library was used with the APC code. The 28-group library consists of fast and epithermal sections derived from Brookhaven's ENDF/B cross-section source tape and from AI's AIENDF data, together with a specially prepared set of thermal cross-sections. The APC code is a multigroup discrete-ordinates ($P_L S_N$) neutron-transport code based on ANISN. APC alternates between radial and axial cases until consistent eigenvalues and leakages are obtained.

This method was also used to calculate f , the ratio of the power density in an energy probe to the density in the surrounding fuel; and ψ_r and ψ_z , the radial and axial power-density distributions.

The calculation started with an equivalent cylindrical fuel cell as illustrated in Figure 14. Dimensions and compositions are given in Tables 11 and 12. Fluxes and power density were calculated using APC in the noniterative radial mode. The probe-to-fuel power factor, f , was obtained from these data. Then cross-sections were flux-weighted to give spatially averaged cross-sections. These were used as the core material in the R-Z APC calculation, using the model illustrated in Figure 15. The material compositions are described in Table 13. The mixed Type I and IV sleeves were simulated by filling the inner (Core-I) zone with Type-IV cell material, while the outer (Core-II) zone was filled with Type-I material.

The configuration with an empty Type-IV sleeve in the center required a slightly different model shown in Figure 16. The void-streaming option of APC was used, the effective height adjusted to give an aluminum rod worth

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Figure 14. Actual and Cylindrical Cell Geometry

TABLE 11
CELL DIMENSIONS

Zone	Contents	Outer Radius (cm)
1	Fuel	1.5368
2	Gap	1.5446
3	Ceramic barrier	1.5528
4	Cladding	1.5908
5	Sleeve	1.5968
6	Coolant channel and probe	1.6936

TABLE 12
CELL NUCLIDE DENSITIES
(Units = nuclei/ 10^{-24} cm³)

Nuclide	Material					
	Fuel	Gap	Barrier	Cladding	Sleeve	Channel (x 10^{-24})
Hydrogen	0.06533	10 ⁻⁷				653.3
Zirconium	0.03522					352.2
Uranium-235	0.001446					14.46
Uranium-238	0.0001069					1.069
Oxygen			0.0435			
Sodium			0.0078			
Aluminum			0.0165			
Samarium-149			0.00001988			
Carbon				0.000262		
Chromium				0.0072		
Iron				0.00478		
Nickel				0.0648	0.0823	
Molybdenum				0.00913		
Gadolinium-155					*	

*Gadolinium-155 densities:

Sleeve Type

Density

I

0

II

0.0002100

III

0.0004098

IV

0.0006463

REFLECTOR OUTER RADIUS

NO SHIMS	16.551 CM
1-1/4 IN. ON 4 FLATS	18.420 CM
1-1/4 IN. ON 4 FLATS	
+1/2 IN. ON 4 DRUMS	19.200 CM

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Figure 15. Geometry for Types I, II, III, and Mixed
Types I and IV Sleeves

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1. TRANSVERSE BUCKLING FROM TYPE-I SLEEVE ANALYSIS
USED TO SIMULATE AXIAL LEAKAGE.

2. EFFECTIVE HEIGHT OF 54 CM USED IN VOID ZONES.

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Figure 16. Geometry for Type-I Sleeves with Empty
Type-IV Sleeve in the Center

TABLE 13
REACTOR NUCLIDE DENSITIES
Units = nuclei/ 10^{-24} cm³

Nuclide	Material			
	Reflectors	Vessel	Grids	Empty Sleeve
Beryllium	0.12015			
Oxygen	0.00088			
Chromium		0.01705	0.0128	
Iron		0.0562	0.0345	
Nickel		0.01037	0.0329	0.0823
Molybdenum		0.00302	0.0041	
Gadolinium				0.000575

Notes: 1. Cross-sections from homogenized cell calculations used in core zones.
2. Empty sleeve in center used with Type-I sleeves.

corresponding to the measurements reported earlier in this report. This height was then used with APC to provide the correct streaming leakage from the void enclosed by the center Type-IV sleeve. Figure 17 shows the resulting power depression at the center.

The values of Λ , β_{eff} , and f are tabulated in Tables 14 and 15. The value of Λ for the unpoisoned case agrees precisely with the calculation reported in NAA-SR-11850, even though the latter value was based on different cross-section data. The spatial power distribution, ψ_r , and ψ_z are shown in Figures 17 through 21.

2. Thermal Data

The heat-capacity model was taken from an empirical fit to measured data from several sources, as reported in NAA-SR-11850. For transients started up by a drum insertion of more than \$1 the following model was used:

TABLE 14
BASIC KINETICS PARAMETERS

Sleeves	Shims	Λ	β_{eff}
Type I	None	$6.5 \pm 0.2 \mu\text{sec}$	0.0079 ± 0.0003
Type I + Empty Type IV in Center Position	1-1/4 in. on 4 flats	$7.6 \pm 0.2 \mu\text{sec}$	0.0077 ± 0.0003
Type II	1-1/4 in. on 4 flats	$7.2 \pm 0.2 \mu\text{sec}$	0.0076 ± 0.0003
Types I and IV Mixed	1-1/4 in. on 4 flats + 1/2 in. on 4 drums	$8.0 \pm 0.2 \mu\text{sec}$	0.0076 ± 0.0003
Type III	1-1/4 in. on 4 flats + 1/2 in. on 4 drums	$7.5 \pm 0.2 \mu\text{sec}$	0.0077 ± 0.0003

TABLE 15
FLUX-DEPRESSION FACTORS

Sleeve	Probe/Fuel Power Factor* (f)
Type I	0.964
Type II	0.906
Type III	0.868
Type IV	0.835

*Probes having nearest nearest-neighbors of different types use a weighted mean, i. e. for a probe located adjacent to two Type-IV sleeves and one Type-I sleeve:

$$f = 2/3 \times 0.835 + 1/2 \times 0.964 = 0.878$$

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Figure 17. Power Profile for SNAPTRAN Machine, Type-I Sleeves with Empty Type-IV Sleeve in Center Position

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Figure 18. Power Profile for SNAPTRAN Machine, No Shims, Type-I Sleeves

1-29-70 UNCL

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Figure 19. Power Profile for SNAPTRAN Machine, Type-II Sleeves

1-29-70 UNCL

7700-4540

Figure 20. Power Profile for SNAPTRAN Machine, Mixed Type-I and Type-IV Sleeves

1-29-70 UNCL

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Figure 21. Power Profile for SNAPTRAN Machine, Type-III Sleeves

$$C_p = C_D + C_E \left(\frac{\theta_E}{T} \right)^2 \frac{e^{\theta_E/T}}{\left(e^{\theta_E/T} - 1 \right)^2} + C_L T$$

where

C_p = heat capacity, joules/°R,

C_D = 6842 joules/°R,

C_E = 12590 joules/°R,

θ_E = 3463°R,

C_L = 2.5 joules/(°R)²

T = fuel temperature, °R.

The integral of this gives the energy-temperature relationship, for the case of uniform temperature. The effect of spatial distribution was studied using an auxiliary code which considers a given amount of energy to be distributed over the reactor volume in proportion to the power profile. The mean

temperature is calculated and compared with the temperature corresponding to that energy in the absence of spatial effects. The resulting correction was less than 2% and was ignored in these analyses.

A more important matter is the effect of heat conduction to the cladding, which has a heat capacity of about 1110 joules/°R. The time constant for this condition is roughly 1 sec, decreasing with temperature. Transients started by reactivity of more than \$1 have most of their energy deposition confined to an interval of about 0.03 sec, and thus take place too fast for heating of the cladding to interfere. Transients conducted at 60 to 80¢, however, have temperature-deposition times of 8 to 15 sec, and the cladding is heated almost as much as the fuel. For transients conducted at reactivity of less than \$1, then, the cladding heat capacity was added to that of the fuel, increasing it by an amount ranging from 11% to 200°F to 6% at 1000°F.

It may be important to note that the power level resulting from this analysis refers to power absorbed in the fuel, not to total fission power. Fission in a SNAPTRAN reactor releases 203.9 Mev of energy, of which only 180.4 Mev is absorbed in the fuel during the transient.⁽⁸⁾ The remainder of the energy escapes with neutrons, photons, or neutrinos, or is released in the form of delayed beta or gamma decay after the transient.

3. Indirect Power Normalization

For transients initiated by reactivity insertions of less than \$1 the energy probes did not give reliable data.⁽⁸⁾ The small fuel sample lost energy by conduction to the thermocouple wires and to the sheath which enclosed it, so an indirect process was required for those cases. The neutron-detector calibrations were determined from transients conducted above \$1 in the usual fashion. These calibrations were used in the transients below \$1. The temperature was obtained by integrating the calibrated power data, and then applying the heat-capacity model to deduce the temperature corresponding to that energy deposition.

4. Data Selection for Differentiating

Not all of the data observed could be used in the temperature-coefficient determination. The first and last few degrees of temperature rise occurred

slowly, giving rise to numerical error in calculating $d\rho/dT$. Also, delayed-gamma background response made the neutron detectors unreliable after the power had dropped far below the peak. The data chosen for the $d\rho/dT$ calculation began when the mean temperature had risen by 30°F, and ended when it came within 30° of the final temperature recorded, or after the power fell below 40% of the peak value, whichever came first.

5. Sample Analysis Results

The STAC Code produced a large volume of output, both tabular and graphical, for each transient. Figures 22 through 33 are selected from the output of Transient 39. They are typical of all the transient results.

Figures 22 through 25 show some of the raw data after translation to binary format and engineering units. The curve in Figure 26 shows the mean energy determined by correlating all the energy-probe data, both before and after smoothing. The Figure 27 curve shows the composite power channel obtained by splitting the individual channels, both before and after smoothing. Figure 28 is the reactivity calculated from the power shape, followed in Figure 29 by a comparison between smoothed mean energy from the electronic processing (EP) data and temperature deduced from the normalized integrated power.

Figures 30 through 33 show temperature coefficient determined first by numerical differentiation and then by the slope (labeled "BFIT" on the graph) of a linear fit to the graph of reactivity vs temperature. EP temperature is used first then that deduced from the calibrated integrated-power data. The results of the last four graphs are in satisfactory agreement. The value of BFIT from the linear fit to the EP data would be considered the most reliable result in this case. The slight slope in the results of numerical differentiation is more likely due to the vagrancies inherent in data acquired on submillisecond time scales than to the actual dependence of coefficient on temperature. Only the value of BFIT would be used in the experimental intercomparisons.

C. TRANSIENT-ANALYSIS RESULTS

1. Summary of Transients

A total of 31 transients covering five different reactor configurations were analyzed successfully. Table 16 lists the transients nominal reactivity and

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Figure 22. Raw Data from Power Detector PD-3

7700-4542-2

Figure 23. Raw Data from Power Detector PD-5

7700-4542-3

Figure 24. Raw Data from Energy Probe EP-2-1

TRANSIENT 39 * TAPE 008 * SECTION 1

701279
0012 0000

7700-4542-4

Figure 25. Raw Position Data for Drum B

TRANSIENT 39 * TAPE 008 * SECTION 2

7.5
(1), " m"

7700-4542-5

Figure 26. Mean Energy Data from All Probes
(Unsmoothed and Smoothed)

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Figure 27. Composite Power Data from All Detectors
(Unsmoothed and Smoothed)

TRANSIENT 39 * TAPE 008 * SECTION 2

Z.00279
0029 0000

7700-4542-7

Figure 28. Reactivity Calculated from Power Shape

TRANSIENT 39 * TAPE 008 * SECTION 2

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Figure 29. Temperature Comparison
(EP Data vs Integrated Power Data)

TRANSIENT 39 * TAPE 008 * SECTION 2

7700-4542-9

7700-4542-9

Figure 30. Temperature Coefficient by Differentiation of EP Data

TRANSIENT 39 * TAPE 008 * SECTION 2

7700-4542-10

7700-4542-10

Figure 31. A Linear Fit to EP Data

TRANSIENT 39 * TAPE 008 * SECTION 2

7700-4542-11

7700-4542-11

Figure 32. Temperature Coefficient by Differentiation
of Integrated Power Data

TRANSIENT 39 * TAPE 008 * SECTION 2

700279
0036 0000

7700-4542-12

Figure 33. A Linear Fit to Integrated Power Data

TABLE 16
SUMMARY OF TRANSIENT RESULTS

Transient Number	Sleeve Type	Nominal Reactivity (\$)	Nominal Temperature (°F)	Observed Reactivity (\$)	Observed Temperature (°F)	Peak-Power Channel	Power Calibration	Peak Power	Mean Temperature (°F)	Temperature Coefficient from	
										Energy Probe (C/°F)	Calibration Power (C/°F)
004	I (Nickel) ↓	1.05	120	1.05	105	PD-1	0.215	13.8 Mw	155	-0.118	-0.125
006		1.20	80	1.18	71	PD-2	0.225	168 Mw	212	-0.113	-0.113
007		1.30	100	1.30	91	PD-2	0.207	386 Mw	292	-0.126	-0.120
021	I (Nickel) (Center Void)* ↓	1.20	127	1.20	118	PD-4	0.379	160 Mw	264	-0.128	-0.128
022		1.20	320	1.21	306	PD-4	0.386	177 Mw	439	-0.152	-0.153
023		1.20	601	1.21	579	PD-4	0.377	206 Mw	700	-0.172	-0.168
026	II (Nickel + gadolinium) ↓	0.60	230	0.61	-	PD-6	(1.02)	506 kw	487	-	-0.083
027		0.80	153	0.79	-	PD-6	(1.02)	1.07 Mw	363	-	-0.090
028		1.20	118	1.22	82	PD-5	0.905	209 Mw	263	-0.108	-0.108
030		1.20	617	1.19	630	PD-5	0.872	266 Mw	780	-0.111	-0.111
031		1.15	930	1.15	908	PD-5	0.931	168 Mw	1029	-0.115	-0.115
033		1.20	125	1.21	105	PD-5	0.813	259 Mw	310	-0.096	-0.092
034	I + IV (Nickel + gadolinium) ↓	0.60	73	0.58	-	PD-4	(0.920)	462 kw	331	-	-0.069
035		0.60	370	0.60	-	PD-4	(0.920)	615 kw	599	-	-0.069
036		0.80	215	-	-	PD-6	(1.02)	1.24 Mw	524	-	-0.078
037		1.20	138	1.20	128	PD-5	0.868	196 Mw	322	-0.092	-0.095
038		1.20	350	1.21	347	PD-5	0.826	284 Mw	558	-0.088	-0.089
039		1.20	620	1.20	609	PD-5	0.814	324 Mw	822	-0.083	-0.083
041	III (Nickel + gadolinium) ↓	0.60	103	0.60	-	PD-4	(0.920)	548 kw	406	-	-0.060
042		0.60	270	0.59	-	PD-4	(0.920)	-	394	-	-0.063
043		0.50	75	0.49	-	PD-4	(0.920)	301 kw	351	-	-0.059
044		0.50	350	0.50	-	PD-4	(0.920)	454 kw	569	-	-0.053
045		0.50	520	0.51	-	PD-4	(0.920)	-	648	-	-0.054
046		0.50	870	0.51	-	PD-4	(0.920)	-	1008	-	-0.056
047	I (Nickel) ↓	0.50	70	0.48	-	PD-4	(1.22)	150 kw	223	-	-0.117
048		0.60	85	0.58	-	PD-4	(1.22)	249 kw	247	-	-0.121
049		0.80	95	0.77	-	PD-6	(1.41)	583 kw	194	-	-0.119
050		1.20	85	1.19	104	PD-5	1.22	192 Mw	260	-0.110	-0.109
051		1.20	350	1.21	388	PD-5	1.23	219 Mw	521	-0.135	-0.134
052		1.20	620	1.21	647	PD-5	1.19	249 Mw	769	-0.149	-0.149
053		1.20	860	1.19	907	PD-5	1.27	218 Mw	1014	-0.152	-0.156

*"Center Void" refers to the configuration having a Type-IV sleeve replacing the center element.

initial temperature reported by the experimenters, as well as the reactivity, power, and temperature data from the analysis. The mean temperature, defined as the middle of the analyzable temperature range, was used as an abscissa in the coefficient correlations. Temperature coefficients determined from both energy-probe data and normalized integrated-power data are shown. The former are preferable but only the latter are available for transients conducted at reactivity less than \$1, i.e. below prompt critical. Also shown are the identities of the detectors used for peak-power determination and the calibration factor applied. Factors in parentheses were deduced indirectly due to initial reactivity of less than \$1.

2. Correlations

The most important correlation of these data is probably the plot shown in Figure 34. This shows the mean prompt-temperature coefficient for each transient plotted vs mean temperature. Table 17 gives the slopes and intercepts. The data from each reactor configuration have been plotted with a least-squares-fit straight line of the form $\alpha = A + BT$. These show a consistent variation of coefficient with temperature for a given configuration, a progressive reduction of the coefficient magnitude with increasing gadolinium content, and a tendency to slope toward a positive-temperature effect at high loadings. The line for Type IV was located by extrapolating the linear fits to the preceding data at 0 and 1000°F, as shown in Figure 35.

D. EVALUATION OF RESULTS

1. Relationship to Other Data

First it should be recognized that the results reported in this section do not, and should not, agree with data reported elsewhere in this report. These data are transient prompt-temperature coefficient-of-reactivity data. This type of data is typically more positive than static isothermal data by roughly 0.1¢/°F since it does not involve gridplate or reflector temperature effects. It should be larger in magnitude by a factor of about 1.2 than prompt-isothermal data deduced from static measurements, because in prompt transients the energy is deposited preferentially in the center of the core. Since the deposition is more effective than average in the center, the result is a larger coefficient.

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Figure 34. Prompt-Temperature Coefficient of Reactivity

TABLE 17
PROMPT-TEMPERATURE COEFFICIENT RESULTS

Sleeve Configuration	Temperature Intercept ($^{\circ}\text{F}$)	Coefficient Slope [$^{\circ}\text{F}^{-2}$]
Type I	-0.107	-4.8×10^{-5}
Type I with Type IV + Void in Center	-0.104	-9.9×10^{-5}
Type II	-0.087	-2.5×10^{-5}
Mixed Types I and IV	-0.080	-2.4×10^{-7}
Type III	-0.062	$+8.7 \times 10^{-6}$

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Figure 35. Temperature Coefficient vs Gadolinium Loading

In actual practice, attempts to extract prompt data from isothermal-static data involve considerable inaccuracy. The corrections for grid and reflector temperature effects tend to be large, temperature-variant, and difficult to measure or calculate independently.

The comparison with SNAPTRAN-1 is quite valid, however. The comparison of data is shown in Table 18. This comparison uses data taken without gadolinium and ignores the slight effect of the nickel sleeves in this experiment. The agreement appears to be quite satisfactory in all respects.

TABLE 18
COMPARISON OF KINETICS RESULTS FOR THE
UNPOISONED SNAPTRAN REACTOR

	SNAPTRAN-1	This Study
Measured temperature coefficient at 0°F, $\beta/^\circ\text{F}$	-0.121	$-0.107 \pm 14\%$
Calculated temperature coefficient at 0°F, $\beta/^\circ\text{F}$	-0.134	-
Measured Λ (using calculated β_{eff}), μsec	6.6 ± 0.2	-
Calculated Λ , μsec	6.5	6.5 ± 0.2
Calculated β_{eff}	0.00794	0.0079 ± 0.003
Measured variation of thermocouple with temperature, $(\beta/^\circ\text{F})^2$	-6.2×10^{-5}	-4.8×10^{-5}
Calculated variation of thermocouple with temperature, $(\beta/^\circ\text{F})^2$	-8.75×10^{-5}	-

2. Experimental Errors

Instrument calibration is an important error source. The energy-probe calibration depends upon the temperature of the probe relative to that in the reactor as a whole. The voltage is easy to measure and is accurately known. The power density in the energy probe, however, is affected by flux depression resulting when poison shims surround the fuel elements and from flux distortion resulting from neutron leakage near the control drums.

Also, the energy-absorption rate is not quite proportional to power density. About 12.4 of the 180.4 Mev per fission absorbed promptly in the fuel is due to neutron and photon interactions. These particles travel over mean-path lengths large in comparison to the probe size. Thus the absorption from these effects is more representative of the reactor as a whole than it is of the lower power density. The rest of the prompt absorption is due to fission-fragment recoils, which travel only about 0.001 cm in fuel of probe material. The transport of these particles is not an important problem.

The probes also tend to degrade from use at high temperatures. Hydrogen loss lowers their heat capacity, and prolonged exposure to a hot reducing environment tends to degrade the junction quality. Junction resistance checks and comparison between various probes were used to conserve and control these effects. Some probes failed and were replaced during the course of the experiments, and the new probes did not give noticeably different performance when compared to the replaced probes. Comparison between probes gave agreement usually within $\pm 10\%$.

Neutron detectors were affected by nonlinearity, gamma sensitivity, and calibration uncertainty. Really bad nonlinearity was detectable by comparison with other detectors. Gamma sensitivity was a problem only after the power peak, and this was eliminated by rejecting data occurring after the power fell below half its peak value.

Neutron-detector calibration was not a problem for transients run above prompt critical, since the energy-probe response was used. For the delayed-critical transients the energy probes lost heat too rapidly by conduction to their nonfuel components to be usable. The indirect calibration discussed earlier involved additional error sources in calculating the ratios of detector responses. There may also have been some error due to the effects of detector nonlinearity on the indirect calibration, but it was not possible to observe it independently.

The experiment itself allowed certain complications. In the prompt-critical transients a drum bounce of about $\pm \$0.02$ maximum amplitude prevented establishment of a true asymptotic period before thermal feedback, and probably affected the linearity of the reactivity-vs-temperature curves. Axial conduction, especially to the gridplate, was not a problem. Calculations have shown that

SNAP experiments conducted at reactivity over \$0.40 are not affected by this error. Thermocouples on the gridplates confirmed this. Heat loss to the cladding is not negligible, however. As stated earlier, the heat capacity was increased by 6 to 11% to account for this effect. The time constant for this conduction is roughly 1 sec, although it varies with temperature. Considering the uncertainty in the time constant for conduction, we estimate a potential error of about 3% due to this effect on transients below prompt critical.

3. Data-Acquisition and Analysis Problems

The results were very sensitive to time shifts in the recording-digitizing process, and precise synchronization was required. Errors in this class were detectable by comparison between detectors, and were well controlled. Detector and recording noise was a problem only when usable neutron detectors were too widely spaced in sensitivity.

The software on both the digitizing and analysis computers gave little trouble. Several selections of the smoothing interval were used on a set of four transients, with negligible effect on the results. Numerous hardware errors occurred in the area of tape writing or reading errors. These resulted in replacement or rejection of the erroneous data, however.

The accuracy of all the analysis depended on the calculation of the spatial power profiles (ψ_r and ψ_z), flux-depression factors (f), prompt-neutron generation time (Λ), and the effective delayed-neutron fraction (β_{eff}). The uncertainties in Λ and β_{eff} due to error in the calculation itself were 3 and 4%, respectively. It was not possible to evaluate the uncertainty in these parameters due to input-data error. In the SNAPTRAN experiments β/Λ calculated in the same way as described herein agreed with β/Λ from the experiments within 2%, and Table 18 shows that the value also agrees with the calculation made for this experiment. The number of transients in this series was not sufficient to conclude a new value of Λ . It was not possible to estimate the error in f , except to note that $0.8 < f < 1.0$. Thus, if the amount of depression were in error by 20%, the error in f would be $0 < \text{error} < 4\%$.

The results also depended, as expected, on the accuracy of the drum calibration. In the SNAPTRAN experiments the uncertainty in the 45° to 0° worth

of a single drum was estimated as about 1.5ϕ .⁽⁸⁾ Then, for an insertion of \$1.20, the reactivity above \$1 would be in error by about 8%. This experiment probably had similar accuracy. The use of more reactivity would have made the fuel too radioactive for safe sleeve changing.

4. Overall Accuracy Estimates

The standard deviation of the final prompt-coefficient correlation in SNAP-TRAN-1 was $\pm 0.015\phi/^{\circ}\text{F}$. Due to the large number of transients correlated, the error in the correlation itself may have been less. In this study, only a few experiments were done per configuration. From Figure 34, we might estimate the linear fit of the Type-I data to be in error at 0°F by $\pm 0.01\phi/^{\circ}\text{F}$ (9%) due to random error and delayed-critical normalization problems. Combining this error with the estimated drum calibration, Λ , β_{eff} , and f errors by the sum-of-squares method gives an overall uncertainty of 14% in temperature coefficient. The agreement shown in Table 18 is of this same order. The fits for the other configurations are probably of comparable absolute accuracy, i.e. $\pm 0.015\phi/^{\circ}\text{F}$.

It might also be noted that the reflector configuration can affect the temperature coefficient. The slight difference between the Type-I results and the original SNAPTRAN data, as well as the difference between the Type-I and the Type-I + center-void results may be partly due to the change in reflector effect. Past studies have shown that a thicker reflector enhances the coefficient, as this example would indicate. Consequently it must be kept in mind that one is measuring the effect of both poison addition and reflector configuration change.

PART II: WATER-IMMERSION EXPERIMENTS

I. DESCRIPTION OF THE TEST ENVIRONMENT

A. FACILITY

Measurements of the reactivity effects of gadolinium-155 on a water-immersed SNAP core were made in the SNAP Critical Facility (Bldg 012) which is located near the north boundary of the Atomics International Nuclear Field Laboratory. An artist's drawing of the facility is shown in Figure 36. A detailed description is given in Reference 6.

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

The SCA-4B machine was used for all of the water-immersion experiments. Figure 37 is a photograph of the machine. The assembly consists of an upper reflector tank and core vessel welded together and attached to the upper support structure and a base tank which is attached to a movable frame. Raising the base tank to the operating level is done by means of a magnet and motor-drive system. A dimensional drawing of these components is in Figure 38. This configuration also allows for attachment of a radial beryllium reflector when required.

C. OPERATION

The operation of the SCA-4B machine consists of the following steps:

- 1) Raising the base tank to the upper limit.
- 2) Filling the base tank with water to a level just above the top of the fuel. Since the bottom of the core vessel is perforated, this also floods the core vessel.
- 3) Adding water to the top tank to achieve a critical configuration. Reactor control is maintained by raising or lowering the water level in the top tank. With 3 in. of water in the top tank the reactor core is fully reflected.

All operations are done remotely from the control console. Water levels are monitored by television observation of sight glasses mounted on each tank.

Adjustment of the core excess reactivity is provided by the use of boron-carbide-filled aluminum tubes (boron splines) inserted in the interstitial spaces in the core vessel as required. A measure of the excess reactivity is then obtained from the measurements of the water level in the top tank.

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Figure 36. Building 012, General Critical Building

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Figure 37. SCA-4B Critical Assembly Machine

NOTE ALL DIMENSIONS IN INCHES
BASE TANK IN UP POSITION

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Figure 38. SCA-4B Small Top-Tank Configuration

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Figure 39. Spline Configuration, Type-I Sleeves

II. PROGRAM

A. CORE LOADING

Fuel elements with nickel sleeves were loaded onto the SNAPTRAN grid-plates in the SCA-4B assembly in the manner described in Part I until a critical configuration was achieved. Then, with the addition of each element to the minimum critical configuration, boron splines were added to adjust the excess reactivity so that criticality was always achieved on the top-tank water.

Upon completion of the experiments with this configuration, Type-II sleeves were installed and all 37 elements reloaded in one step with the spline configuration of the preceding series. Consequently the reactor was subcritical by the worth of the gadolinium. Criticality was then achieved by removing the poison splines in a stepwise manner using the inverse multiplication technique. This procedure was followed for each subsequent configuration.

B. CONFIGURATIONS

The configurations studied during the water-immersion phase of the program are summarized in Table 19. Configurations 1 through 5 were fully water-reflected except for the 6 internal beryllium reflectors which were used throughout the program. (See Figure 39.) For Configurations 6 and 7, an external beryllium reflector was installed around the core vessel. The reflector used was the basic SNAP 10A reflector block with control drums and accessory hardware removed. This entire assembly was fully immersed in water.

C. REACTIVITY-MEASUREMENT TECHNIQUES

Immediately following the establishment of a new configuration, the worth of the water in the top tank was calibrated. This was accomplished by adding (or removing) boron splines to change the critical water level. Excess reactivity was then added by raising the water level a known amount and measuring the resulting period. In this fashion an incremental water worth was measured over the useful range of from 1/8 to 3-1/2 in. of water in the top tank and an integral-worth curve constructed.

TABLE 19
WATER-IMMERSION CONFIGURATIONS

Fuel-Configuration No.	Sleeve Configuration	Gadolinium Concentration (Σ_{at})
1	I	0 (nickel only)
2	II	0.073
3	III	0.150
4	31 IV, 6 III	0.217 calculated average
5	Central void with I, II, III, or IV	*
6	III with external beryllium reflector	0.150
7	Mixed loading [†]	0.118

*The central-void configuration consisted of 36 Type-I sleeves with the central element removed and the designated sleeves installed in the central-element location.

†The mixed loading consisted of Type-IV sleeves on the inner 19 elements and Type-I sleeves on the outer ring of 18.

Spline and sleeve worths were then determined by removing a spline or sleeve and noting the change in critical water level. However, when sleeves were removed from the core they were not replaced with Ni sleeves so the worths obtained are with respect to the no-sleeve condition. Total spline and sleeve worths were obtained by summing the individual values.

D. FLUX-DISTRIBUTION MEASUREMENTS

Axial and radial flux distributions were measured for each configuration as described in Part I of this report.

E. WATER-TEMPERATURE COEFFICIENT

In the course of a day's operation the water pump would operate continuously for periods of up to 8 hr. During this time the water temperature would rise to

as high as 95°F as a result of the pumping action and would cool to ~72°F during the subsequent overnight shutdown period. It was then determined that increases in water temperature of this magnitude contributed a rather large positive reactivity effect to each spline and sleeve measurement. To correct the data that had been taken, the water-temperature coefficient and rate-of-rise as a function of pumping time were measured. To extend the temperature range over which data could be obtained, electrical immersion heaters were placed in the water-storage drums and water temperature raised to 110°F. The reactor was periodically brought to critical during the temperature rise and the excess reactivity from the reference water height measured. This test was held with each of the four sleeve configurations (No. 1 through 4, Table 19). No water-temperature measurements were made with the beryllium-reflected configurations.

TABLE 20
SPLINE AND SLEEVE CLASSIFICATION

a. Boron Splines		
Type	Typical Location	Radius (in.)
1	18-11-10	0.733
2A	18-19-11	1.466
2B	18-19-26	1.940
3A	26-19-27	2.643
3B	27-20-19	2.933
3C	34 27-26	3.196
4A	27-20-Be	3.666
4B	27-34-Be	3.880

b. Gadolinium Sleeves			
Type	Number in Core	Typical Location	Radius of Center of Sleeve (in.)
1	1	10	0
2	6	11	1.27
3A	6	19	2.18
3B	6	12	2.54
4A	12	20	3.35
4B	6	13	3.81

III. RESULTS

A. REACTIVITY MEASUREMENTS

The water-immersion phase of the experiment was begun with the fuel-loading procedure. Fuel was loaded with Type-I sleeves installed and a critical configuration achieved with 27 elements. Loading was continued to the full core using boron splines to offset the excess reactivity. The loading was completed with 37 fuel elements and 35 splines. This core, using SNAPTRAN fuel elements on a 1.27-in. lattice, was approximately 20¢ more reactive than the SCA-4B fuel on a 1.26-in. lattice for the same spline configuration.

The spline configuration was then rearranged to provide a more uniform distribution. The final configuration, 36 splines, is shown in the core-loading diagram, Figure 39. Spline and sleeve (fuel) locations were classified by radius as shown in Table 20. Numbers given in the column headed "Typical Location" refer to the element numbers shown in Figure 39. Spline configurations with GD sleeves installed are shown in Figures 40 through 43.

The reactivity effects of removing the splines one at a time are listed in Table 21 for each sleeve configuration. The summation of these worths plus the excess reactivity controlled by the top-tank water level was calculated as shown. As gadolinium was added to the core in increasing concentrations the number of boron splines required to maintain a critical configuration was correspondingly reduced. All of the splines were removed with a configuration of 31 Type-IV and 6 Type-III sleeves. By measuring the difference in worth between a Type-IV and Type-III sleeve in each of the six locations containing the Type-III sleeves it was calculated that a core fully loaded with Type-IV sleeves would be 61¢ subcritical when immersed in water. Sleeve-worth results are also shown in Table 21.

The reactivity lost by the addition of gadolinium to the core was calculated from the spline and sleeve-worth measurements and is plotted as a function of concentration in Figure 44.

B. CENTRAL-VOID CONFIGURATION

Table 22 gives the central fuel-element worth and the worths of the various sleeves installed in the central location with the fuel element removed. Sleeve worths are all with respect to the nickel sleeve.

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Figure 40. Spline Configuration, Type-II Sleeves

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Figure 41. Spline Configuration, Type-III Sleeves

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Figure 42. Spline Configuration, Type-III Sleeves with
External Beryllium Reflector

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Figure 43. Spline Configuration, Mixed Loading with External Beryllium Reflector

TABLE 21
WATER-REFLECTED REACTIVITIES

a. Spline Worths						
Position Type	Type-I Sleeves		Type-II Sleeves		Type-III Sleeves	
	Number	Reactivity (¢)	Number	Reactivity (¢)	Number	Reactivity (¢)
1	3	49.6	2	44.9	1	40.5
2A	3	44.8	2	41.0	0	-
2B	6	40.3	3	35.8	3	32.8
3A	6	29.0	3	27.2	2	25.0
3B	3	28.5	0	-	0	-
3C	6	23.3	6	22.0	3	18.8
4A	3	20.4	3	18.4	0	-
4B	6	19.8	3	17.8	0	-
Total	36	1104.3	22	601.4	9	245.3
Top-Tank Excess Reactivity		40.0		21.0		5.0
Total Reactivity		1144.3		622.4		250.3

b. Sleeve Worths				
Type	Number in Core	Reactivity (¢)		
		Type II	Type III	Type IV
1	1	13.7	27.8	46.0
2	6	15.9	25.3	41.0
3A	6	9.1	20.3	28.5
3B	6	10.2	18.0	22.0
4A	12	8.8	21.8	30.5
4B	6	13.1	22.4	31.5
Total Worth		409.1	805.4	1070.4

Figure 44. Reactivity Effect of Gadolinium

TABLE 22
CENTRAL-VOID AND
SLEEVE WORTHS

Item	Worth (¢)
Central element	+152
Type-II sleeve	- 47
Type-III sleeve	- 83
Type-IV sleeve	-112

C. BERYLLIUM-REFLECTED CONFIGURATION

Critical configurations were achieved with Type-III-sleeve and mixed-sleeve (IV and I) loadings with the external beryllium reflector block installed around the core vessel. Spline worths and excess-reactivity measurements are given in Table 23. Since no water-temperature-coefficient measurements were made with these configurations the data reported in Table 23 are uncorrected for water-temperature variations. A review of the corrections applied to the data from the unreflected configurations indicates a probable error of about 50¢ in the total reactivities reported in Table 23.

Comparing the total worth of the splines in the Type-III water-reflected configuration, \$2.45, with the same spline worth in the beryllium-reflected configuration, \$8.65, indicates that to a first approximation the reflector is worth about \$6.20 with respect to water when fully immersed.

D. TOP-TANK CALIBRATION

An integral-worth curve for the water in the SCA-4B top tank was determined for each configuration. Very little change in the shape of the curve or the total worth of the water in the top tank was noted. A typical integral-worth curve is shown in Figure 45.

E. FLUX DISTRIBUTIONS

Relative axial-flux distributions obtained from the gold-foil activation data are given in Figure 46 for sleeve configurations with no external-beryllium reflector installed. The Type-IV sleeve configuration consisted of 31 Type-IV and 6 Type-III sleeves as discussed earlier in this report. Figure 47 shows axial-flux distributions for two sleeve configurations with the external-beryllium reflector installed.

TABLE 23
BERYLLIUM-REFLECTED REACTIVITIES

a. Spline Worth				
Position Type	Type-III Sleeves		Mixed Loading (IV+I)	
	Number	Reactivity (¢)	Number	Reactivity (¢)
1	3	44.6	6	30.9
2A	3	39.2	6	32.9
2B	6	35.95	8	32.7
3A	3	28.0	3	28.7
3B	3	25.8	4	26.7
3C	6	25.5	6	25.2
4B	6	19.8	6	29.6
Total	30	900	39	1166
Top-tank excess		13		20
Total reactivity		913		1186
b. Sleeve Worth				
1	1	27.5	1	20.5
2	6	25.9	6	21.8
3A	6	18.0	6	23.2
3B	6	16.1	6	26.8
4A	12	25.0	0	
4B	6	29.6	0	
Total Worth	37	865	19	451

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Figure 45. Top-Tank Integral Worth (SCA-4B Assembly)

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Figure 46. Axial-Flux Distributions
(No External Beryllium Reflector)

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Figure 47. Axial-Flux Distributions
(External Beryllium Reflector Installed)

All axial distributions have been normalized to the maximum value in each case. No significant effects on the axial-flux distribution were observed as a result of changing sleeves or adding the external reflector.

Radial-flux distributions are given in Figure 48 for sleeve configurations without the external-beryllium reflector. Again, the Type-IV configuration contained 6 Type-III sleeves. Figure 49 gives the radial distributions obtained with the external-beryllium reflector installed. All radial-flux data were normalized to the central foil in each case. A significant change in radial-flux distribution occurred in the mixed-loading configuration with the external reflector installed. Flux peaking in the outer ring of elements is quite evident.

F. WATER TEMPERATURE-COEFFICIENT RESULTS

From the excess-reactivity measurements obtained during the water-heating tests, a water temperature coefficient was calculated for the operational range of temperatures, i.e. 70 to 100°F. These results are given in Table 24.

TABLE 24
SCA-4B WATER TEMPERATURE COEFFICIENT

Sleeve Type	Temperature Coefficient ($\phi/^\circ\text{F}$)
I	+0.39
II	+0.53
III	+0.53
IV	+0.57

Spline- and sleeve-worth data were corrected using these values and the rate-of-rise heating curve. Due to its long time-constant the fuel temperature changed very little during the short time (~5 min/run) that the core was immersed in water. The abrupt change in coefficient between the nickel and gadolinium sleeve configurations is significant, but a quantitative explanation is not now available.

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Figure 48. Radial-Flux Distributions
(No External Beryllium Reflector)

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Figure 49. Radial-Flux Distributions (External
Beryllium Reflector Installed)

IV. ANALYSIS OF THE CRITICAL EXPERIMENTS

A. PREDICTIONS OF GADOLINIUM-155 WORTH

The initial predictions of Gd^{155} worth in the SCA-4B environment, as in SNAPTRAN, were made in 1967. From the time of these early calculations until the experiments were carried out, changes, as mentioned earlier, were made in the program.

The final predictions of the worth of Gd^{155} in water was made using the then extant 26-cross-section library and the APC buckling-iteration code. The results of this study are given in Table 25 along with the initial experimental results.

TABLE 25
GADOLINIUM-155 WORTH IN SCA-4B

Foil Type	$\Sigma_a t$	Predicted Worth (\$)	Experimental Worth (\$)*
I	0.0	Reference	Reference
II	0.067	4.30	5.22
III	0.133	7.15	8.94
IV	0.200	9.41	>11.44

*Boron spline results

B. POST-EXPERIMENT ANALYSIS

When the initial results of the SCA-4B experiments were obtained and the deviation from prediction noted, a revised analysis of the experiments was made. A new 28-group cross-section library had been developed for the SNAP program with success in predicting critical assemblies. It was decided that this library be used to do the re-analysis.* Table 26 lists the summary of the revised predictions.

*At that time the foil-inspection report was not thoroughly scrutinized and the deviation in gadolinium concentration from specifications was not noted. The re-analysis, therefore, used the same $\Sigma_a t$ as the predictions.

TABLE 26
CALCULATED WORTHS FOR DIFFERENT
FOIL TYPES

Foil Type	$\Sigma_a t$	β_{eff}	Excess (\$)*	Worth (\$)
I	0.0	1.0794	9.42	-
II	0.067	1.0437	5.37	4.20
III	0.133	0.0180	2.27	7.40
IV	0.200	0.0085	-0.19	9.86

* $\beta_{\text{eff}} = 0.0078$

In an analysis of the boron splines it was found that a fluctuation existed in boron concentration. Several measurements were made of the poison content of the splines. The first consisted of weighing a spline with and without the boron. This gave a value of 1.5885 gm/spline. The second measurement consisted of obtaining a weighted average of 20 splines. In this case the average boron content was taken to be 1.34 gm/spline. Finally a set of measurements was made after termination of the experiments in which each spline was weighed, and five weighed empty. A deviation of 0.37 gm was found between the maximum and minimum weights.

The variation in spline composition could account for the deviation between measured sleeve worth and that inferred from the splines, if several high-boron-content splines were used to obtain representative worths. It should be noted that the splines used in the calibration were not developed as precision devices for measuring reactivity. They were devised to provide a means for suppressing reactivity. On the other hand the Gd-Ni sleeves were fabricated to a tight specification and were, hopefully, more uniform. The sleeve worths were assumed to be more credible when derived by sleeve-water substitution than when inferred by spline substitutions.

The receiving-inspection reports for the Gd sleeves were reviewed at this time, and significant deviation was observed. The $\Sigma_a t$ value assumed for each foil type was revised to reflect the increased thickness of the actual foil as opposed to the theoretical value. These values are given in Table 27 for convenience of the reader.

TABLE 27
ACTUAL vs SPECIFIED CONCENTRATIONS
($\Sigma_a t$) OF THE FOILS

Foil Type	$\Sigma_a t$ (theoretical)	$\Sigma_a t$ (fabricated)
I	0.0	0.0
II	0.067	0.0730
III	0.133	0.1499
IV	0.200	0.2298

Figure 50 shows the calculated and measured worths of Gd^{155} in water using the actual $\Sigma_a t$ values for the measured-foil data. These were plotted against foil worth as determined by foil-water substitutions. The experimental and calculated data show excellent agreement. The excess reactivity of the unpoisoned sleeveless assembly as determined by sleeve substitutions (~10.20 with no foils) agrees favorably with the Type-I calculated reactivity when corrected by the Ni-to-water worth ($\$9.42 + \$0.25 = 9.67$).

On an absolute basis a normalization of approximately \$0.50 is needed to equate calculation and experiment. This represents a marked increase in analytical capability over previous cross-section sets and analytic tools for calculating SNAP systems in a water environment.

C. RELATED STUDIES

The determination of β_{eff} for the SCA-4B geometry was made using the new library. It was found to be 0.0078 for the water environment. This value was slightly different from that previously used (0.008) for SNAP assemblies.

A spline-centered cell was configured to provide a heterogeneous treatment of the boron spline. The results of this study gave a worth of \$9.60 for 37 splines with a mean boron content of 1.34 gm, and a value of \$10.40 if the mean content were 1.6 gm. This study substantiated the assumptions that the boron splines were not black to neutrons and hence the measurements could be influenced by variation of the poison content of the splines.

Figure 50. Calculated and Measured Worth of Gadolinium-155 in SCA-4B

PART III: INTRINSIC WATER SUBCRITICALITY POTENTIAL

I. THE INTRINSIC WATER SUBCRITICALITY CONCEPT

As a consequence of their small size and the heavy fuel loading required for criticality, SNAP reactors are undermoderated. Due to this the addition of water to a SNAP core results in a significant increase in reactivity. The Intrinsic Water Subcriticality (IWS) concept is a means whereby a SNAP reactor may be designed to be safely subcritical when subjected to a water environment without grossly affecting core performance under normal conditions. The hazard associated with the potential launch abort of a SNAP system into water is thereby minimized.

Gd^{155} has a large resonance cross-section in the low-thermal-energy region. The relatively unique shape of this cross-section produces more absorption in a water-immersed SNAP reactor than in the normal operation mode, since the water softens the thermal spectrum. In addition the effective Gd^{155} cross-section in the operating condition provides the optimum burnout rate as a burnable poison for the advanced SNAP systems currently under investigation. The prepoisoning of SNAP systems with Gd^{155} would, therefore, produce two desired results; an engineered safeguard against launch aborts, and an ideal burnable poison for normal operation.

Figure 51. Calculated and Measured Worth of Gadolinium-155 in SCA-4B and SNAPTRAN

II. SUMMARY OF EXPERIMENTAL EVIDENCE

A. PREVIOUS EXPERIMENTS (Natural Gadolinium)

Previous experiments⁽⁹⁾ were done to investigate the effect of natural gadolinium on the IWS potential of a SNAP reactor. Although not identical, natural Gd has a thermal cross-section similar in shape to Gd^{155} .

The previous experiments indicated that natural gadolinium did have IWS potential, although the dominating Gd^{157} isotope had a very large cross-section which made it unsuitable for use in a SNAP reactor due to its rapid burnout rate. Hence the present experimental investigation of Gd^{155} which has the appropriate magnitude cross-section.

It should be noted that the previous experiments were static and therefore did not offer verification of the calculated effect of Gd on the prompt temperature coefficient.

B. PRESENT EXPERIMENTS (Gadolinium-155)

The effect of Gd^{155} on reactivity of a SNAP system has been determined in both a conventional (NaK) and a water environment. The worth of Gd^{155} on the SNAP system is enhanced by the presence of water in the core. However, the enhancement of Gd^{155} worth due to water worth between the water and the dry medium is not as large as supposed, based on earlier experiments on natural Gd. Further, the mechanism which enhances the worth of Gd^{155} in water also produces an adverse effect on the prompt temperature coefficient (it becomes less negative). The concentration of Gd^{155} necessary to assure subcriticality in a water environment, yet permit operation under normal conditions following launch, would deplete the negative prompt temperature coefficient by an unacceptable amount. In fact, a positive coefficient would be found above about 1100°F. In addition the reactivity penalty associated with large concentrations of Gd^{155} is such that IWS applications to SNAP systems other than the SNAP 10A/2 type is virtually impossible. Figure 51 is a summary graph of the worth of Gd^{155} in the SNAPTRAN and SCA-4B environments. A significant curvature exists in the SCA-4B worth curve. Earlier experiments on natural gadolinium using boron splines for calibration did not show this effect. The differential-worth curve also shown demonstrates a slowly varying worth for Gd^{155} in water relative to the operating environment.

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III. CONCLUSIONS

The detailed experimental program used in determining the potential of Gd^{155} for IWS purposes has demonstrated that the in-core application of Gd^{155} does not achieve the desired effect. The concentrations of Gd^{155} necessary to produce a marginal IWS situation appears to be enough to thwart the achievement of sustained reactor lifetime, both from an excess-reactivity and temperature-coefficient standpoint.

The effect of Gd^{155} on temperature coefficient has been found to be of such significance that the use of Gd^{155} as a burnable poison is severely limited. Each specific reactor must be analyzed to determine the applicability of the concept and its effect on transient reactor performance.

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