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ANALYSIS OF SNAPTRAN-1 AND -2
REACTOR KINETICS EXPERIMENTS

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ABSTRACT

Extensive analyses of the SNAP Transient (SNAPTRAN)-1 non-destructive test series were performed to extract basic kinetics data for a wide range of transient conditions. Detailed analyses of the SNAPTRAN-2 test were also performed, to assess the behavior of the SNAP 10A/2-type reactor during a destructive transient. Assessment of the kinetics data was also of principal interest as was the mode of reactor shutdown. A detailed correlation between experiments and analyses was made. These studies were undertaken as a major effort in the Aerospace Nuclear Safety Program.

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SUMMARY

The short-period kinetics of the SNAPTRAN-1 and -2 reactors were investigated. Kinetic behavior in the absence of fuel temperature reactivity feedback effects was found to be adequately described by a standard 6-group model up to about \$3 reactivity insertion, with somewhat decreasing accuracy at larger reactivities. Drum calibrations performed by the standard period technique were used to determine the reactivity insertions, and were found to be adequate.

The ratio of the prompt neutron generation time to the total delayed neutron yield ($\Lambda/\beta_{\text{eff}}$) was measured at 1210 sec^{-1} . Based on a calculated β_{eff} of 0.00794, this gives a Λ of $6.6 \mu\text{sec}$. The estimated confidence interval for Λ is $\pm 0.02 \mu\text{sec}$.

The prompt temperature coefficient was measured in a large number of transients under different conditions. The results showed a trend toward larger (more negative) coefficients for transients in which higher temperatures or shorter periods were encountered. The results were satisfactorily correlated by the following equation:

$$\alpha = (A + C\bar{T})(1 + \tau\omega_0), \quad \dots(1)$$

where

α = observed temperature coefficient ($¢/°\text{F}$)

A = -0.121 ¢/°F

C = $-6.23 \times 10^{-5} \text{ ¢/°F}^2$

τ = $1.17 \times 10^{-4} \text{ sec}$

ω_0 = inverse period at beginning of temperature compensation

$\bar{T}$ = "mean temperature" = $1/2(T_i + T_f)$

T_i = volume-average temperature before transient

T_f = volume-average temperature after transient.

The standard deviation of the fit was $S = 0.015 \text{ ¢/°F}$.

The initial asymptotic period of the SNAPTRAN-2 destructive test was $206 \mu\text{sec}$. The inserted reactivity was \$5.05, as determined from previous drum calibration measurements. Based on a standard 6-group model and the lifetime measured in the SNAPTRAN-1 tests, the measured period gives a calculated reactivity

of \$5.01. These two reactivities agree to within less than 1%, less than the uncertainty in the drum calibration measurements.

A temperature coefficient of $-0.208 \text{ } \ell / ^\circ\text{F}$ was calculated, based on the data from the 10% hydrided energy probes EP-154 and EP-155. This result was in reasonable agreement with the correlation of the temperature coefficient measurements quoted above.

Shutdown was achieved by mechanical disassembly which began when the center of the core reached approximately 1930°F . The shape of the power burst, the energy release, and the disassembly rate were in satisfactory agreement with the results from an analytical model (ENRECO) based on fuel disintegration caused by the buildup of internal hydrogen pressure which, in turn, causes core disassembly.

The peak power and energy release were 73,100 Mw and 47.2 Mw-sec, respectively. Both of these results are on the basis of total released energy.

The results of the destructive test were in general agreement with prediction, and no unexpected phenomena were observed.

The period-dependence of the temperature coefficient arises from spatial neutron kinetics effects. An analysis of such effects was performed, using a time-, temperature-, and space-dependent neutron transport theory code (TSN). The calculation predicts a similar variation and magnitude of the temperature coefficient to that determined by correlating the experimental data. A comparison between the experimental and theoretical constants for Equation 1 is given below.

	<u>A</u>	<u>τ</u>	<u>C</u>
SNAPTRAN Experiment	$-0.121 \text{ } \ell / ^\circ\text{F}$	$117 \text{ } \mu\text{sec}$	$-6.23 \times 10^{-5} \text{ } \ell / ^\circ\text{F}^2$
TSN Calculation	$-0.134 \text{ } \ell / ^\circ\text{F}$	$132 \text{ } \mu\text{sec}$	$-8.75 \times 10^{-5} \text{ } \ell / ^\circ\text{F}^2$

The agreement is regarded as satisfactory.

The results of the comparison between theory and experiment indicate that TSN and ENRECO can be used with confidence to describe the consequences of high-input-reactivity accidents with SNAP reactors.

PART 1

SNAPTRAN-1 EXPERIMENTS

NAA-SR-11850

1-1

I. INTRODUCTION

A. PURPOSE OF SNAPTRAN-1 EXPERIMENTS

The SNAPTRAN-1 test series included a number of nondestructive power transients, the purpose of which was to enable a detailed study of the prompt neutron kinetics of the SNAP core with a beryllium reflector. Extensive analyses of these tests were performed to extract basic kinetics data over a wide range of variables. This report will describe the methods used and results obtained in the analyses. The report will also discuss briefly, but is not intended to be a comprehensive reporting of, the test series itself, the apparatus and machines, and the detailed data from each of the transients.

B. DESCRIPTION OF TEST FACILITY

1. Test Site

The test program was conducted by Phillips Petroleum Company (PPCo) at the Initial Engineering Test (IET) Facility of the National Reactor Testing Station (NRTS). A detailed description of this facility is given in Reference 1.

The reactor assembly was mounted on a four-track test dolly that was constructed from two railroad flatcars. The tests were performed in a corrugated metal building which was also mounted on tracks. This allowed removal of the structure during the SNAPTRAN-2 destructive test described in Part 2 of this report. A photograph of these facilities is shown in Figure 1-1.

2. Reactor Assembly

The SNAPTRAN-1 reactor consisted of a SNAP 10A/2-type core and reflector, with special control drum assemblies designed to provide rapid reactivity insertions and withdrawals.^{(1,2,3)*} The upper and lower vessel heads were also modified to allow for special instrumentation. A diagram of the machine is shown in Figure 1-2.

The thermal insulation box surrounding the core in Figure 1-2 was used only for a special series of transients which were initiated at elevated temperatures.

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* Superscripts refer to the References listed in Appendix A.

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Figure 1-1. SNAPTRAN Experimental Site at NRTS

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Figure 1-2. Diagram of SNAPTRAN-1 Reactor

Figure 1-3. SNAPTRAN-1 Reactor With Thermal Box Installed

Under normal operation, the box was removed. Photographs of the assembled machine with and without the thermal box are shown in Figures 1-3 and 1-4, respectively.

A cross-sectional diagram of the core-reflector assembly is shown in Figure 1-5. A description of the core loading can be found in Reference 3, while a summary of the core properties is given in Table 1-1.

Control Drums A and D are referred to as impulse drums, since their drive mechanisms were designed to insert and remove large amounts of reactivity with a single continuous drum rotation, giving "impulse" reactivity insertions (see Figure 1-5). Similarly, Drums B and C are referred to as stepper drums because of their use in step reactivity insertions, rapid start-stop movements of a drum to insert small amounts of reactivity rapidly. Two different shim-ming configurations were used during the test series, as will be explained in the section on control drum calibrations. A detailed description of the reactor assembly is given in Reference 2.

3. Instrumentation

The analyses described in this report are based on power, temperature, and drum position data. A summary of the instrumentation used to obtain these data is presented in Tables 1-2 through 1-4. A brief discussion of the instrumentation follows. A more complete description can be found in Reference 4.

a. Power

The power level of the reactor was determined by means of neutron and gamma detectors. The neutron detectors consisted of small in-core and out-of-core ionization chambers and neutron scintillation detectors. The various detector amplifier ranges were set to provide data from a reactor power of 1 watt to 1.5×10^{11} watts. The sensitivity ranges of the various detectors are shown in Figure 1-6, which is taken from Reference 1. The detectors for both the SNAPTRAN-1 and SNAPTRAN-2 tests are shown.

The sensitivities of the detectors in units of detector current per watt were originally determined at PPCo's Materials Testing Reactor Calibration Facility.⁽¹⁾ Final normalization of the detector sensitivities was made when the detectors were in position for the test series. The reactor was then operated

Figure 1-4. SNAPTRAN-1 Reactor (Inverted Position,
With Thermal Box Installed)

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1-8

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Figure 1-5. Cross Section of Core-Reflector Assembly and Drum Travel Diagrams

TABLE 1-1
PROPERTIES FOR SNAPTRAN-1 REACTOR*

<u>FUEL</u>		<u>GRID PLATES</u>	
Number of fuel rods	37	Material	Hastelloy-C
Lattice spacing (in.)	1.26	Thickness:	
Fuel rod OD (in.)	1.212	Upper (in.)	0.125
Fuel rod length (in.)	12.25	Lower (in.)	0.50
Weight of U ²³⁵ (gm)	4752.5	<u>REFLECTORS</u>	
Total weight of fuel (gm)	50937.809	Material	Be
Average fuel density (gm/cm ³)	6.061	Cladding	None
Average fuel composition: †		Number of internal reflector pieces	6
Fuel alloy (wt % U in Zr)	10	Fixed reflector thickness (in.)	1.875
U ²³⁵ enrichment (%)	93	<u>CORE VESSEL</u>	
Atom densities (10 ²⁴ atoms/cm ³ of fuel):		Material	316-SS
U ²³⁵	0.001450	Inside diameter (in.)	8.875
U ²³⁸	0.0000747	Thickness (in.)	0.032
Zr	0.03537	<u>REFLECTOR CONTROL DRUMS</u>	
H	0.0650	Number	4
H/Zr atom ratio	1.841	Material	Be
<u>CLADDING</u>		Nominal thickness:	
Material	Hastelloy-N	Maximum shim (in.)	3.50
Outside diameter (in.)	1.25	Minimum shim (in.)	1.875
Thickness (in.)	0.015		

* These properties are based on data from Ref. 5, 6, and 7.

† Certain other materials present only in very small amounts have not been included here. They are given in Ref. 5.

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TABLE 1-2
SNAPTRAN-1 NUCLEAR INSTRUMENTATION

Detector	Type*	Distance From Reactor (ft)	Mounting Location	Maximum Power Range (watts)
N-1	FNIC	10	Flatcar	1.1×10^9
N-2	FNIC	7	Reactor support structure	3.0×10^8
N-3	FNIC	10	Flatcar	3.5×10^9
N-4	FNIC	10	Flatcar	1.6×10^{10}
N-5	FNIC	14	Test cell floor	3.5×10^9
N-6	FNIC	7	Reactor support structure	7.0×10^9
N-7	FNIC	10	Flatcar	9.0×10^8
N-8	FNIC	10	Flatcar	4.5×10^{10}
N-9	γ IC	10	Flatcar	1.5×10^{11}
N-10	γ IC	7	Flatcar	4.0×10^{11}
N-11	γ IC	10	Flatcar	1.0×10^{10}
N-12	TNIC	7	In-core mini chamber	1.5×10^{11}
N-13	TNIC	-	In-core mini chamber	1.1×10^{11}
N-14	TNIC	-	In-core mini chamber	1.2×10^{11}
N-15	FNSCINT	-	Under flatcar	2.2×10^{11}
N-16	FNSCINT	14	Under flatcar	2.2×10^{11}
N-17	TNIC	14	Under flatcar	1.1×10^{11}
N-18	TNIC	2.5	Reactor support structure	1.5×10^{11}
N-19	TNIC	-	In-core mini chamber	1.2×10^{11}
N-20	TNIC	-	In-core mini chamber	1.5×10^{11}

* FNIC - Fast neutron ionization chamber

TNIC - Thermal neutron ionization chamber

γ IC - Gamma Ionization chamber

FNSCINT - Fast neutron scintillation detector (NaI)

TABLE 1-3
SNAPTRAN-1 ENERGY PROBES
(Sheet 1 of 3)

Probe No.	Type*	AI Serial No.	Interstice †	Elevation (in.)
EP-1	10H	78-6.5-10H	1-3-4	Centerplane
-3	10H	104-6.5-10H	1-2-7	Centerplane
-4	5H	40-6.5-5H	4-11-12	Centerplane
-5	5H	41-6.5-5H	3-9-10	Centerplane
-6	10H	105-6.5-10H	2-3-9	Centerplane
-7	10H	106-6.5-10H	2-8-9	Centerplane
-8A	5N	M-2-5N	5-13-14	+ 6
-8B	5N	M-2-5N	5-13-14	+ 4
-8C	5N	M-2-5N	5-13-14	+ 2
-8D	5N	M-2-5N	5-13-14	Centerplane
-9A	5N	M-1-5N	4-12-13	+ 6
-9B	5N	M-1-5N	4-12-13	+ 4
-9C	5N	M-1-5N	4-12-13	+ 2
-9D	5N	M-1-5N	4-12-13	Centerplane
-10	5H	43-6.5-5H	6-7-17	+ 1.5
-12	0H	71-6.5-0H	5-14-15	Centerplane
-13	10H	107-6.5-10H	12-25-26	Centerplane
-14	10H	1-6.5-10H	10-22-23	Centerplane
-16	5H	47-6.5-5H	9-10-22	Centerplane
-17	10H	108-6.5-10H	9-21-22	Centerplane
-18	5N	121-5N	8-19-37	Centerplane
-19	5N	190-5N	13-27-28	Centerplane
-20A	5N	M-3-5N	18-34-35	+ 6
-20B	5N	M-3-5N	18-34-35	+ 4
-20C	5N	M-3-5N	18-34-35	+ 2
-20D	5N	M-3-5N	18-34-35	Centerplane
-21	5H	84-6.5-5H	17-33-34	Centerplane
-22A	5N	M-5-5N	5-14-15	+ 6

TABLE 1-3
SNAPTRAN-1 ENERGY PROBES
(Sheet 2 of 3)

Probe No.	Type*	AI Serial No.	Interstice [†]	Elevation (in.)
EP-22B	5H	M-5-5H	5-14-15	+ 4
-22C	5N	M-5-5N	5-14-15	+ 2
-22D	10H	M-5-10H	5-14-15	Centerplane
-23	10H	109-6.5-10H	12-13-27	Centerplane
-24	5H	85-6.5-5H	25-26-R	Centerplane
-25	5H	110-6.5-5H	21-22-R	Centerplane
-26A	5N	M-4-5N	27-28-R	+ 6
-26B	5N	M-4-2.5N	27-28-R	+ 4
-26C	5N	M-4-5N	27-28-R	+ 2
-26D	10N	M-4-10N	27-28-R	Centerplane
-27	10N	123-6.5-10N	34-35-R	Centerplane
-28	5H	111-6.5-5H	33-34-R	Centerplane
-29	5H	112-6.5-5H	32-33-R	Centerplane
-30	5H	113-6.5-5H	31-32-R	Centerplane
-31	10H	2-6.5-10H	29-30-R	Centerplane
-32	10H	5-6.5-10H	26-27-R	Centerplane
-34	5H	45-6.5-5H	0.5 in. into Be Reflector [§]	Centerplane
-35	5H	46-6.5-5H	1.0 in. into Be Reflector	Centerplane
-36	10H	10-6.5-10H	1-2-3	Centerplane
-37	5H	37-6.5-5H	1-2-3	+ 2.4
-38	0H	11-6.5-0H	1-5-6	Centerplane
-39				Centerplane
-40				+ 2.4
-41A	5H	M-13-5H	4-5-13	+ 6
-41B	5H	M-13-5H	4-5-13	+ 4
-41C	5H	M-13-5H	4-5-13	+ 2
-41D	5H	M-13-5H	4-5-13	Centerplane
-42	10H	143-65-10H	2-8-19	Centerplane

TABLE 1-3
SNAPTRAN-1 ENERGY PROBES
(Sheet 3 of 3)

Probe No.	Type*	AI Serial No.	Interstice †	Elevation (in.)
EP-43	1OH	145-65-1OH	23-24-R	Centerplane
-44	1OH	133-65-1OH	1-3-4	Centerplane
-45	1OH	134-65-1OH	1-3-4	+ 2
-46	1OH	148-65-1OH	5-14-15	Centerplane
-47	1ON	151-0-1ON	5-6-15	Centerplane
-48	1OH	152-65-1OH	3-10-11	Centerplane
-49	1OH	146-65-1OH	18-19-36	Centerplane

* 1OH - 10 wt % Uranium - Hydrided (normal fuel)

5H - 5 wt % Uranium - Hydrided

1ON - 10 wt % Uranium - Not hydrided

5N - 5 wt % Uranium - Not hydrided

OH - No Uranium - Hydrided

† The three numbers refer to fuel rods surrounding the interstice. An R refers to a reflector piece and is used to denote an interstice at the outer edge of the core.

§ This is the radial distance from the vessel outside wall to the probe location in the fixed reflector block.

TABLE 1-4
SNAPTRAN-1 THERMOCOUPLES
(Sheet 1 of 2)

No.	Type	Position	Interstice	Elevation (in.)
TC-1	C-A 10-mil spot-welded	Top grid	11-12-25,12-25-26	
TC-2	C-A 10-mil spot-welded	Top grid	4-12-13,4-11-12	
TC-3	C-A 10-mil spot-welded	Top grid	1-4-5,1-3-4	
TC-4	C-A 10-mil spot-welded	Top grid	7-17-18,7-18-19	
TC-5	C-A AI diaphragm	Lower grid	25-26-R,12-25-26	Upper part of grid
TC-6	C-A 10-mil spot-welded	Lower grid	25-26-R	Lower part of grid
TC-7	C-A AI diaphragm	Lower grid	4-11-12	Upper part of grid
TC-8	C-A AI diaphragm	Lower grid	4-11-12	Lower part of grid
TC-9	C-A 10-mil spot-welded	Lower grid	3-4-11	Upper part of grid
TC-10	C-A 10-mil spot-welded	Lower grid	3-4-11	Lower part of grid
TC-11	C-A 10-mil spot-welded	Lower grid	1-3-4,1-2-3	Upper part of grid
TC-12	C-A 10-mil spot-welded	Lower grid	1-3-4,1-2-3	Lower part of grid
TC-13	C-A 10-mil spot-welded	Lower grid	25-26-R	Upper part of grid
TC-14	C-A 10-mil spot-welded	Vessel wall		Centerplane
TC-14	C-A 10-mil spot-welded	Vessel wall		Centerplane
TC-22	C-A 10-mil spot-welded	11	10-22-23	+ 4
TC-25	C-A 10-mil spot-welded	15	15-16-31	+ 4
TC-26	C-A 10-mil spot-welded	11	11-24-25	Centerplane
TC-30	C-A AI diaphragm	1	(see note 6)	+ 5
TC-31	C-A AI diaphragm	1	(see note 6)	+ 3
TC-32	C-A AI diaphragm	1	(see note 6)	Centerplane
TC-33	C-A AI diaphragm	3	(see note 6)	+ 5

TABLE 1-4
SNAPTRAN-1 THERMOCOUPLES
(Sheet 2 of 2)

No.	Type	Position	Interstice	Elevation (in.)
TC-34	C-A AI diaphragm	3	(see note 6)	+ 3
TC-35	C-A AI diaphragm	3	(see note 6)	Centerplane
TC-36	C-A AI diaphragm	16	(see note 6)	+ 5
TC-37	C-A AI diaphragm	16	(see note 6)	+ 3
TC-38	C-A AI diaphragm	16	(see note 6)	Centerplane
TC-39	C-A AI diaphragm	37	(see note 6)	+ 5
TC-40	C-A AI diaphragm	37	(see note 6)	+ 3
TC-51	C-A 10-mil spot-welded	36	35-36-R	Centerplane
TC-52	C-A 10-mil spot-welded	36	19-36-37	Centerplane
TC-53	C-A 10-mil spot-welded	36	36-37-R	Centerplane
TC-54	C-A 10-mil spot-welded		1-4-5	Centerplane
TC-59	C-A 2-mil spot-welded	8	8-9-21	Centerplane
TC-1A	C-A 10-mil spot-welded	On coupon	10-11-24	+ 1.5
TC-2A	C-A 10-mil spot-welded	On coupon	3-4-11	Centerplane

Notes:

1. C-A type indicates chromel-vs-alumel thermocouples.
2. TC-1A and TC-2A are installed on dummy coupons for radiation test.
3. Numbers in position column indicate fuel rods.
4. Elevations are referenced to core centerplane.
5. For interpretation of interstice designation, see footnote in Table 1-3.
6. These thermocouples are located within the fuel rods.

NAA-SR-11850
1-17

5-19-66

7611-2504

Figure 1-6. Nuclear Detector Sensitivity Ranges

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7611-5533A

Figure 1-7. Energy Probe (Disassembled)

5-25 65

7611 0843

Figure 1-8. Multiple Energy Probe Assembly

at an estimated power level of 1000 watts, and the various detector calibration factors were established. These factors were used on Tests 110 through 116. A detailed analysis of foils irradiated during this calibration run revealed that the power level had actually been 910 watts, and the calibration factors were then adjusted accordingly. ⁽⁸⁾

Except for the adjustment just described, detectors N-1, N-2, N-3, N-4, N-5, N-7, and N-8 were calibrated properly at the start of the test series. The sensitivities for detectors N-12 and N-13 were determined on Test 112, using a correlation between those detectors and detectors N-1 through N-8. Similarly, detector N-14 was calibrated on Transient 115, and detectors N-15 and N-16 were calibrated on Test 116. After Test 121, it was felt that the time durations of the tests were sufficiently short so that delayed gammas would not significantly affect the responses of the gamma detectors during the usable part of the power burst. Gamma detectors N-9, N-10, and N-11 were subsequently normalized to the average readings of the other detectors. ⁽⁸⁾

b. Temperature

The temperature information used in the analysis was based on data obtained from small interstitial temperature-sensing devices called "energy probes". ^(9,10) Basically, an energy probe consists of a small fuel sample with half-mil chromel and alumel thermocouple wires spot-welded to it to form an intrinsic junction. The fuel sample is typically 0.06 x 0.06 x 0.2 in. and weighs about 0.07 gm. The fuel is supported by a small stainless-steel trough and canned in a stainless-steel sheath. The sheath is packed with MgO powder before being sealed off. A typical energy probe is shown in Figure 1-7. The sheath and MgO powder have been omitted in the photograph to reveal the construction details.

Several probes were constructed with three fuel samples at different axial positions within the same sheath. These were called "multi-probes". With these probes, temperature measurements at three different axial positions within the same interstice could be made. The construction details of such a probe are shown in Figure 1-8.

Most of the energy probes used fuel samples of the same composition as the fuel in the core, but probes with other fuel compositions were used as well. A summary of the compositions is given in Table 1-3.

NOTE: ONLY 2 OF 6
THERMOCOUPLES PER
ELEMENT SHOWN IN
SIDE VIEW.

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Figure 1-9. Diagrams of Thermocouple Instrumentation

NAA-SR-11850

1-20

Four fuel elements in the core were equipped with internally placed, diaphragm-ended thermocouples. These fuel elements occupied Positions 1, 3, 16, and 37 (see Figure 1-5). Each fuel element had three thermocouples placed in holes drilled axially into the fuel and terminating at different axial positions. These fuel elements were spring-loaded so as to maintain a pressure contact between the thermocouple diaphragms and the fuel. A diagram of this arrangement is shown in Figure 1-9.

Twelve 10-mil thermocouples were also spot-welded directly to the surface of the two fuel rods in Locations 2 and 36 (see Figure 1-5). This was accomplished by removing sections of the cladding at three interstice positions at both the axial centerplane and 2 in. from the tops of the elements. The upper grid plate was modified to facilitate easy replacement of the fuel rods in these two locations. A diagram of the spot-welded thermocouple arrangement is also shown in Figure 1-9.

c. Drum Position

The drum position data used in the analysis were obtained from two sources. The time-dependent drum positions during a transient were obtained from potentiometers mounted directly on the drum shafts. Static drum positions, such as the criticality positions, were obtained from a more accurate servosynchro digital indication system. The readout from this system was located on the control console.

d. Other Measurements

Six fuel elements were modified to allow the installation of pressure transducers to monitor hydrogen release from the fuel material. These transducers had sensitivity ranges from 50 to 3000 psi. Dummy transducers were also used to determine the amount of radiation-induced signal.

Axial and circumferential strain gages were used to monitor the cladding strain of the fuel elements in Locations 7 and 18. Again, dummy strain gages were used to evaluate radiation sensitivity. Circumferential strain gages were mounted on the reactor vessel at the axial centerplane to measure vessel expansion.

C. TYPES OF EXPERIMENTS PERFORMED

The experiments fell into two major categories characterized by the method of reactivity insertion. In the first series, a reactivity insertion approximating a "step" was obtained. Two drums, B and C, the "stepper drums", were withdrawn to a position 45 degrees from full in. The remaining two drums, A and D, the "impulse drums", were positioned as required to give the desired reactivity insertion. Then the stepper drums were inserted to their full-in position, driven by a pneumatic cylinder acting through a rack-and-pinion linkage. At the full-in position, the motion was stopped by a pneumatic damping cylinder.

This device gave a rapid insertion of up to \$2.00 reactivity, and also gave nearly constant reactivity through the subsequent nuclear transient. (In practice, the drums were found to bounce slightly at the ends of their travel.) The near-constant reactivity input simplifies analysis of the transient; consequently, step tests were performed in preference to the impulse tests, described below, whenever possible.

The step mechanism also contained provision for an automatically sequenced scram to follow the step insertion after a predetermined delay. This allowed transients to be performed safely at reactivities as high as \$1.50, even though they would not have been self-terminating without damage to the reactor fuel.

In the "impulse" tests, the impulse drums were withdrawn to a position 135 degrees from the full-in position. The stepper drums were positioned such as to give the desired peak reactivity. Then the impulse drums were fired through 450 degrees of travel by a pneumatic cylinder, being stopped at the end of this rotation by a pneumatic snubber. The drums could be given quite high angular velocity during their pass through the reactor. At the end of the pass, the reactor was once again subcritical. The stepper drums were scrambled after the impulse, further reducing the reactivity and diminishing the tail of the burst.

The impulse tests allowed reactivities as high as \$4.12 to be used, since the insertion and removal of reactivity were much faster than that of the stop-and-start action of the stepper tests. Impulse tests were performed with the impulse drums moving, coupled by a chain, and also with a single moving drum. (When the two drums were coupled, it was found that they did not move quite

simultaneously, giving some uncertainty as to what peak reactivity would be reached in each test.)

In principle, the impulse tests gave a very large reactivity input, but one which was always varying. Due to the very low differential drum worth near the full-in position, however, the reactivity remained near its maximum value for 1 to 2 msec. This was long enough to establish an asymptotic period; therefore the impulses were adequate to extend the inhour curve to high reactivities. Temperature compensation, if any, occurred while the drums were moving out again. Thus, the temperature coefficient information from these tests was very sensitive to uncertainties in the drum position data and in the drum calibration. As will be seen later, the temperature coefficient analyses of these transients were somewhat successful in spite of this limitation.

Another series of tests was performed in which several steps or impulses were performed in succession, or "cascaded". The objective was to obtain a high, uniform fuel temperature, as high as 900°F, before starting a transient. In this way, higher core-average temperatures could be obtained without fuel damage in the hot spot than could be obtained if the core were initially at the ambient temperature 60 to 90°F. This was of interest, since there was reason to believe that the temperature coefficient might be dependent on temperature. The steps or impulses were separated in time sufficiently to have little effect on each other, except for the residual heat left in the fuel. Thus, each of these tests may be considered as having been identical to the ordinary steps and impulses except for the elevated initial temperature.

D. TYPES OF RESULTS OBTAINED

In these experiments, it was possible to measure the prompt temperature coefficient at periods sufficiently short to eliminate heat transfer as a factor in the experiment. This improves the accuracy of the resulting data, since its analysis involves no heat-transfer calculation with its associated uncertainty. It was also possible to carry the measurements to temperatures near the threshold of fuel damage in order to observe the variation of coefficient with temperature, if any. In addition, a wide range of peak powers and minimum periods were covered from which a variation of coefficient with these parameters, if any, could be proved or disproved. While measurements of temperature coefficient

have been made on other SNAP assemblies, this series covered much wider ranges of the variables mentioned and, thus, contributed to the overall confidence in these data.

The effective prompt neutron generation time was measured by direct determination of the inhour curve above prompt critical. Both steps and impulses were used over different reactivity ranges. The result was a determination of generation time appropriate to drum positions and reactivities close to those of the later destructive experiment (SNAPTRAN-2). This result was more applicable to the analysis of the destructive experiment than data obtained by less direct methods in which control drums remain near their critical positions.

Another result was the detailed checkout of the capability to measure power correctly and to calculate reactivity from it. This was accomplished in the impulse experiments by comparing reactivity calculated from the power data with the reactivity determined from the control-drum position and fuel temperature. After some work in obtaining correct parameters and in removing difficulties in the data acquisition, this comparison was quite successful. By examining transients at very high reactivities, the need to include spatial neutron kinetics effects was evaluated.

Each type of result will be discussed in more detail in the following sections along with the pertinent data and conclusions.

II. METHODS OF ANALYSIS

A. EXPERIMENTAL DATA ANALYSIS

1. Data Description

The analysis described in this report is based on power, temperature, and drum position data. The instrumentation used to obtain these data was described earlier. The data were originally recorded on analog magnetic tape and later converted to digitized data channels. Each digitized channel consisted of 2000 points, equally spaced in time. Each data point consisted of three decimal digits, accompanied by a sign. The data channels were placed on magnetic tape, along with pertinent control and calibration information, and transmitted to AI for analysis. Upon receipt, each data channel was scaled using the calibration factors provided on the tape. Channel identification was made using a deck of punched channel identification cards made up for each transient. Labeled plots of each scaled data channel were made using graphical computer output.

2. Numerical Smoothing

It was found to be advantageous in several instances during the analysis to smooth certain data channels numerically. Considerable attention was given to developing a suitable smoothing technique. The following method appeared to be the most satisfactory.

A quadratic equation was fit to a present number of points, N , (usually 50) at the right-hand end of the curve using the method of least-squares. The first point was replaced with the point calculated from the equation. Then the N points to the left of the first point were fit with a quadratic which was constrained to have the same magnitude and slope at the abscissa of the first point as was found from the previous fit. The point to the left of the first point was then replaced with the magnitude calculated from the new fit. The process was repeated based on the newly calculated point each time, until the whole curve had been replaced with a new curve continuous in magnitude and first derivative. The derivatives of the individual fits were retained and used in other calculations. In the case of reactor power, the smoothing was performed on the natural log of power.

The degree of smoothing obtained is a function of the smoothing interval. The 50-point interval was found to provide adequate smoothing of the power and temperature data from the transients above prompt critical, while not distorting the shape near peak power. (Some distortion was observed in the power data of the slower transients which were terminated by a scram, but this was of no importance.)

3. Power and Reciprocal Period

The final power trace for each transient was found by splicing together up to seven individual power channels covering overlapping ranges. The splicing was done by matching the areas of adjacent power channels in a narrow region of overlap. This splicing is only done up to peak power. (Attempts to show power shape details, following peak power, failed below 50% of peak power due to the gamma sensitivity of the neutron detectors.) The reconstructed power trace was then renormalized to conform to the original calibration of the peak power channel.

The reconstructed power trace was smoothed using the technique described in the previous section. The inverse period as a function of time was then determined by taking the slope of the power trace in natural log space. This was obtained as a byproduct of the smoothing process.

4. Reactivity Calculated from Power

Reactivity as a function of time was calculated from the power trace during the standard space-independent kinetics equations, with six delayed neutron groups, solved in the following integral form:

$$\rho_6(t) = \frac{\Lambda}{\beta} \omega(t) + 1 - \frac{1}{N(t)} \sum_{i=1}^6 \frac{\beta_i}{\beta} W_i(t_0) e^{-\lambda_i(t-t_0)} + \int_{t_0}^t \lambda_i N(t') e^{-\lambda_i(t-t')} dt', \quad \dots (1-1)$$

where:

$\rho_6(t)$ = reactivity in six-group dollars,

Λ = prompt neutron generation time,

$N(t')$ = neutron density,

ω = reciprocal period,

β_i = total delayed neutron yield,

$\beta = \sum_{i=1}^6 \beta_i$ = total delayed neutron yield,

$W_i(t_0)$ = normalized precursor concentration of the i^{th} group at t_0 , and
 t = time.

The above equation was solved numerically using $N(t)$ and $\omega(t)$ from the reconstructed power trace being analyzed. A Λ of 6.6 μsec , β of 0.0080, and delayed neutron parameters from ANL 5800⁽¹¹⁾ were used. The numerical methods employed to solve the precursor concentration integral were taken from the AIRCOMP II code.^(12, 13) In this method, $N(t')$ as defined in Equation 1-1, is approximated by a three-point Lagrangian fit, and the integral is carried out over successive three-point ranges. No source term has been included in Equation 1-1 because the power data being considered were always high enough to make the source insignificant.

5. Heat Capacity Model

The following Einstein-oscillator heat capacity model was used for this analysis:

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

where:

T = temperature ($^{\circ}\text{K}$),

$C_D = C_E = 5.96 \text{ cal/mole Zr } ^{\circ}\text{K}$,

X = H/Zr ratio,

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Figure 1-10. Heat Capacity of $\text{ZrH}_{1.84}$
(Data Adjusted to an H/Zr ratio of 1.84)

$$\begin{aligned}\theta_E &= 1924 \text{ }^\circ\text{K, and} \\ C_L &= 3.92 \times 10^{-3} \text{ cal/mole Zr.}\end{aligned}$$

The development of this model is described in detail in Reference 14. The parameters were determined to provide a best fit of the equation to several sets of experimental data. The experimental data and subsequent fit are shown in Figure 1-10. All the data have been adjusted to an H/Zr ratio of 1.84.

6. Average Temperature

The volume-weighted average temperature was determined using energy probe data taken at several spatial positions in the core. The data from each temperature channel being used was first converted to energy, using the integral of the heat capacity as follows:

$$E_i^K = \int_{\tau_0}^{\tau_i^K} C_p(T) dT, \quad \dots (1-3)$$

where:

$$\begin{aligned}E_i^K &= \text{energy release at the } K^{\text{th}} \text{ spatial position and } i^{\text{th}} \text{ time point,} \\ &\quad \text{scaled to the volume of the whole core,} \\ C_p(T) &= \text{heat capacity described in the previous section,} \\ \tau_i^K &= \text{measured temperature at the } K^{\text{th}} \text{ spatial position and } i^{\text{th}} \\ &\quad \text{time point, and} \\ \tau_0 &= \text{initial temperature of fuel.}\end{aligned}$$

The resulting energy traces were then averaged by scaling a predetermined energy deposition spatial profile (taken to be the same as the power profile) to the data, using the following least-squares equation:

$$\bar{E}_i = \frac{\sum_K E_i^K \phi^K}{\sum_K \phi^K}, \quad \dots (1-4)$$

where:

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

$$\phi^K = \frac{\psi_r(r^K) \psi_z(z^K)}{\psi_\eta} = \text{ratio of energy deposition at the } K^{\text{th}} \text{ spatial position to the average core energy,} \quad \dots(1-5)$$

$\psi_r(r^K)$ = radial power profile at the K^{th} radial position,

$\psi_z(z^K)$ = axial power profile at the K^{th} axial position,

$$\psi_\eta = \frac{1}{\pi R^2 H} \int_0^R 2\pi r \psi_r(r) \int_0^H \psi_z(z) dz dr = \text{profile normalization factor,} \quad \dots(1-6)$$

R = core radius, and

H = core height.

The conversion of average core energy to average temperature was accomplished as follows: first the temperature change associated with the spatially uniform deposition of an amount of energy (E) was tabulated, using the following equation:

$$E = \int_0^T C_P(T') dT' \equiv f(T) \quad \dots(1-7)$$

The temperature change, $T(E)$, associated with a given uniform energy deposition is then given formally by:

$$T(E) = f^{-1}(E) \quad \dots(1-8)$$

and was found by interpolation in the T -vs- E table. The desired relationship between average temperature and average energy deposited spatially (according to the power profile) is then given by

$$\bar{T}(E) = \frac{1}{\pi R^2 H} \int_0^R 2\pi r \int_0^H T\left(\frac{E \psi_r(r) \psi_z(z)}{\psi_\eta}\right) dz dr. \quad \dots (1-9)$$

Since the integral is tedious to recalculate each time it is required in practical computation, the average temperature was actually obtained from the expression,

$$\bar{T}(E_1) = \{1 - \delta[T(E_1)]\} T(E) \quad , \quad \dots (1-10)$$

where the $T(E)$ is obtained as indicated by Equation 1-8 and

$$\delta[T(E)] = \frac{T(E) - \bar{T}(E)}{T(E)} \quad \dots (1-11)$$

is a tabulated correction factor to account for the spatial energy distribution when making the conversion from energy to temperature. This allows the integral to be calculated only once by an auxiliary code.

A polynomial representation of $\bar{T}(E)$ was originally used for the energy-temperature conversion, but was found to be less accurate than the correction factor applied to $T(E)$, as described. Although small, the correction factor, $\delta[T(E)]$, did eliminate a significant source of error.

The use of a different approach was necessary for those transients with periods longer than about 10 msec. This was because energy probe heat losses became significant as will be shown in the next section. Two methods of circumventing the heat-loss problem were employed.

In the first approach, "Plan A", a simple three-node heat-loss model, was postulated to describe the thermal behavior of the probes. A diagram of this model is shown in Figure 1-11. This model yields the following equation for the trough temperature:

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Figure 1-11. Energy Probe Heat-Loss Model

$$T_2(t) - T_0 = \frac{U_1 A_1}{M_2 C_2} e^{-\frac{U_1 A_1 (1-X)}{M_2 C_2} t} \int_0^t \left[T_1(\tau) - T_0 \right] e^{\frac{U_1 A_1 (1+X)\tau}{M_2 C_2}} d\tau, \quad \dots (1-12)$$

where:

$$X = U_2 A_2 / U_1 A_1.$$

The symbols used in this equation are defined in Figure 1-11. After choosing values of $U_1 A_1$ and X and using the measured values of T_1 , the above equation can be solved numerically for the trough temperature (T_2). The true energy release of the probe can then be calculated from the following equation:

Energy Release = Energy retained in the fuel + Correction for heat loss,
or

$$E(t) = \int_{T_0}^{T_1(t)} M_1 C_1(T_1) dT_1 + U_1 A_1 \int_0^t (T_1 - T_2) dt \quad \dots (1-13)$$

The shape of the resulting energy trace, $E(t)$, can then be compared to the shape of the energy trace obtained by integrating the nuclear detector data. Parameters $U_1 A_1$ and X are determined which give the best agreement between the two energy shapes. From the computed energy, $E(t)$, a corrected probe temperature is calculated, using the known mass and heat capacity.

A second approach, "Plan B", was used to determine calibration factors for various nuclear detectors which would bring the energy determined from the particular detectors into agreement with that determined from the energy probe data. These factors were determined from the short-period step tests, which had valid energy probe data, and were subsequently applied to the long-period transients for which suitable energy probe data were not available.

7. Drum-Inserted Reactivity

The drum positions as a function of time were determined by applying calibration factors to the drum position potentiometer voltage data. The reactivity associated with each drum position was determined by interpolation in an appropriate drum calibration table. The reactivity inserted by the movement of each drum was then determined by subtracting the reactivity associated with the measured "critical" position for each particular drum from the reactivity corresponding to each instantaneous position. The total inserted reactivity at each instance was then obtained by summing over the reactivity inserted by the individual drums.

The drum calibrations were performed using the well-known "period" or "rod-bump" technique. In this technique, the differential reactivity worths of a series of small incremental drum insertions are accumulated to give the total integral worth. The details of these calibrations are given in References 1 and 15.

A detailed error analysis was performed for both a single-drum and coupled-drum calibration result. The differential drum worth data were first broken into 10-degree intervals. The uncertainty in the magnitude of a typical differential data point in each interval was then calculated. This was done by estimating the uncertainty in measuring a typical position change as ± 0.05 degree and the uncertainty in determining the inserted reactivity from the measured stable periods at $\pm 0.2\%$.

The uncertainty in the differential worth curve was then estimated for each 10-degree interval. This was done by considering the uncertainty of the data points in a particular interval, as calculated above. The uncertainty of the fitted curve was reduced below the uncertainty of the individual data points in regions where there were sufficient points to allow for an "averaging" effect. In regions where data points were sparse or nonexistent, the uncertainty of the fitted curve was estimated as the spread in magnitude of all fits that would appear reasonable for the available data.

The differential worth curve was thus represented by a histogram with contiguous elements of 10-degree widths and heights corresponding to the average differential worth with the associated uncertainty as determined above. The

integration of the differential curve was then determined by summing the elements of the histogram and accumulating the uncertainties of the various elements as though they were independent. The accumulated uncertainty as a function of angle then gives an estimate of the error in the integral worth curve.

8. Temperature Coefficient

The temperature coefficient was determined as the slope of a plot of temperature-compensated reactivity vs average core temperature change. The volume-weighted average temperature determined as described earlier was used in this calculation. The slope calculation was usually started after the temperature had increased by 50°F. This insured enough reactivity compensation to have reasonable accuracy. It was continued until the power level had decreased to 40% of its peak. This latter criterion was chosen to terminate the calculation before gamma sensitivity of the neutron detectors could affect the calculated reactivity.

Two methods were employed to determine the compensated reactivity. In the first approach, it was interpreted as the difference between the drum-inserted reactivity and the reactivity calculated from power. This method was generally applicable to all of the transients. In the second approach, the compensated reactivity was interpreted as the change in the power-calculated reactivity as the temperature increases. This method was only meaningful for constant reactivity insertions and could thus be applied only to step transients. It had the advantage of being usable when no suitable drum data were available, and it was used in the analysis of most of the step tests for this reason.

Three methods of interpreting the slope of the compensated-reactivity vs temperature-change curve were employed. In this first approach, the slope was found by fitting a straight line over an eleven-point interval centered about the temperature point being considered. This was done at each temperature point in the range of interest. The straight lines were fit using the method of least squares.

This method gave a fairly microscopic interpretation of the temperature coefficient as a function of temperature, provided the data were sufficiently smooth.

In the second approach, the coefficient at a particular temperature was taken as the slope of a straight line drawn from the first point on the compensated-reactivity vs temperature-change graph to the temperature point being considered. This method was less microscopic than the previous one, but it still indicated the temperature dependence of the coefficient, if any. It was less sensitive to noise in the data, especially at larger temperature changes.

In the third approach used, a straight line was fit over the entire range of interest using the method of least squares. The resulting slope thus represented an average temperature coefficient for that particular temperature range. Although this approach was the least sensitive to the quality of the data, it did not retain any temperature dependence. When the results calculated in this manner are presented, both the original curve and the straight-line fit are plotted. The linearity of the original data can then be evaluated. The tabulated coefficients presented in the data summary tables were also calculated using this method.

9. Prompt Neutron Generation Time

Two approaches were employed to determine the prompt neutron generation time. In the most direct approach, use was made of the inhour data from the step tests above prompt critical. The drum-inserted reactivities were plotted against their corresponding inverse periods, and a straight line was fit to the resulting data. The slope of such a line is just the ratio of the prompt neutron generation time to the total delayed neutron yield (Λ/β).

In the second approach, a value for Λ/β was determined for each of the impulse tests analyzed. Use was made of the variable reactivity insertion characteristic of the impulse tests. A plot of reactivity above prompt critical vs inverse period was made for each test and a straight line was fit to the result. It was necessary to first remove the temperature compensation from the drum-inserted reactivity, using a predetermined estimate of the temperature coefficient. The resulting Λ/β is thus dependent on the temperature coefficient estimate. It is also sensitive to the quality of the inverse period (and, therefore, the power) data. This approach was intended to serve as a check on the results of the first method, but is not considered to be as accurate.

10. Power Profile

To evaluate the validity of the power profiles being used, the following procedure was employed. The energy probe data from the various core positions were first converted to energy using the heat capacity as described in the thermal analysis section. The energy deposition at each location was then averaged over a preset time interval. These averaged energy depositions were then displayed as a function of position, along with the precalculated profile, for comparison.

11. Power Calibration

A plot of the energy from integrated power vs energy determined from the energy probe data was made for each transient. A straight line was then fit to the resulting data. The slope of this line is a factor (defined as PCAL) which must be divided into the power data in order to bring the energy from power into agreement with the energy from the probes.

The degree of linearity of the data also serves as a check on the instrument performance. For example, a simple heat transfer time lag in the energy probe response would appear as a nonlinearity.

B. THEORETICAL DETERMINATION OF KINETICS PARAMETERS

It was necessary to calculate values of the prompt neutron generation time, Λ , and the effective delayed neutron fraction, β_{eff} , in order to calculate reactivity from the power transient data. From the kinetics equations derivation by Henry,⁽¹⁶⁾ it can be seen that the value of β_{eff} may be expressed as

$$\beta_{\text{eff}} = \frac{\nu_0 - \nu'}{\nu_0} \quad \dots (1-14)$$

where ν' is the fictitious value of ν for the fuel material required to make the reactor critical, and ν_0 is the value which would be required if the delayed neutrons were removed. This was evaluated with two neutron transport calculations, the first resulting in a k_{eff} of k' with all the neutron spectrum used, and the second with the delayed neutrons removed resulting in a k_{eff} of k_0 .

From the definitions of ν and k , it follows that $\nu' = \nu/k'$, etc. Then Equation 1-14 is equivalent to

$$\beta_{\text{eff}} = \frac{k' - k_0}{k'} \quad \dots (1-15)$$

The delayed neutron energy spectrum from Keepin⁽¹⁷⁾ was used with the DTF-II computer code⁽¹⁸⁾ and the AI 23-group thermalization cross-section library.⁽¹⁹⁾ The core model used is summarized in Tables 1-5 and 1-6. The result was

$$\beta_{\text{eff}} = 0.00794 \quad .$$

From Henry's derivation, it is seen that if the flux is changing proportionally to $e^{\omega t}$, and if $\omega \gg \lambda_1$, then

$$\rho = \Lambda \omega + \beta \quad \dots (1-16)$$

Thus, Λ is the slope of the curve, ρ vs ω , for suitably small values of ω . (For large ω , flux perturbation is important, and Equation 1-16 does not apply with constant Λ .) This was easy to apply to the experimental data, once values of the reactivity and corresponding asymptotic period became available. It can be shown, from the Boltzmann transport equation, that the inverse period, ω , corresponding to a given reactor configuration, can be determined by solving the steady-state transport equation with a fictitious absorption cross section of ω/v inserted into every velocity group and every region of the reactor. (The v is the neutron velocity for a given velocity group.) A two-dimensional transport solution is required to perform this calculation properly. In the actual reactor configuration, small ω 's can be achieved only by movement of the control drums to create a local streaming path for fast neutrons escaping through the reflector. Moreover, a range of ω 's can be achieved only by moving the control drums to different positions. There is no accurate way to simulate this effect in a one-dimensional calculation. A change in reflector thickness is not quite correct, since the rotation of a drum leaves a small section of the core nearly bare, while the rest retains its nominal reflector thickness. On the other hand,

TABLE 1-5
RADIAL MODEL OF SNAPTRAN-1 REFERENCE REACTOR

Element	Number Density (atoms/cm ³ x 10 ⁻²⁴)		
	Core Region	Interface	Reflector
U ²³⁵	0.001166	0	0
U ²³⁸	0.0000846	0	0
H in ZrH	0.05454	0	0
Zr	0.02917	0	0
Fe	0.000282	0.003768	0
Cr	0.000379	0.001701	0
Ni	0.003339	0.000599	0
Mo	0.000708	0.000054	0
Na	0.000528	0.000279	0
K	0.001142	0.000604	0
Be	0	0.09557	0.1230
Radial thickness (cm)	10.221	1.291	5.398
Number of intervals	20	4	14
Mesh spacing (cm)	0.51105	0.32275	0.38557

two-dimensional transport codes are too expensive to use for ω calculations of this type if enough mesh points are used to describe the control-drum geometry correctly.

A one-dimensional calculation can be used with the help of some assumptions. If k_{eff} is represented in terms of volume- and velocity-averaged cross sections and a nonleakage factor, L , then

$$k_{\text{eff}} = \delta \frac{\nu \Sigma_f}{\Sigma_a} L \quad \dots (1-17)$$

The factor, δ , corresponds to whatever (small) error results from inaccuracies in the cross sections. A value of leakage, L , corresponding to the critical drum position will give

TABLE 1-6
AXIAL MODEL OF SNAPTRAN-1 REFERENCE REACTOR

Element	Number Density (atoms/cm ³ x 10 ⁻²⁴)		
	Upper Grid Plate	Core Region	Lower Grid Plate
U ²³⁵	0	0.001166	0
U ²³⁸	0	0.0000846	0
H in ZrH	0	0.05454	0
Zr	0	0.02917	0
Fe	0.000711	0.000282	0.000796
Cr	0.005618	0.000379	0.006290
Ni	0.019349	0.003339	0.021664
Mo	0.005513	0.000708	0.006173
Na	0.002879	0.000528	0.002631
K	0.006001	0.001142	0.005485
Region thickness (cm)	1.806	31.115	1.562
Number of intervals	6	50	4
Mesh spacing (cm)	0.3010	0.6223	0.3905

$$k_c = \delta \frac{\nu \Sigma_f}{\Sigma_a} L_c = \delta \quad \dots (1-18)$$

Suppose the leakage is changed, by rotation of the control drums, to a value L_s . Then

$$k_s = \delta \frac{\nu \Sigma_f}{\Sigma_a} L_s \quad \dots (1-19)$$

According to Henry,⁽¹⁶⁾ the reactivity resulting from a physical perturbation in the system is

$$\rho = \frac{\nu_0 - \nu'}{\nu_0} \quad \dots (1-20)$$

where ν' is the value of ν required to make the reactor critical in the new configuration, and ν_0 is the corresponding value in the original (critical) configuration. (Obviously, ν_0 is the real ν for the physical material.) This gives the familiar formula,

$$\rho = \frac{k - 1}{k} \quad \dots (1-21)$$

From Equation 1-19, then:

$$\rho = \frac{L_s - L_c}{L_s} = \frac{k_s - k_c}{k_s} \quad \dots (1-22)$$

If the perturbed system is then poisoned with an ω/ν poison to a k_{eff} of 1:

$$k_{sp} = \frac{\delta \nu \Sigma_f}{\Sigma_a + \omega/\nu} L_s = \delta \quad ; \quad \dots (1-23)$$

and this is the correct value of ω corresponding to ρ . But supposing the unperturbed system were poisoned by this same amount.

$$k_p = \frac{\delta \nu \Sigma_f}{\Sigma_a + \omega/\nu} L_c \quad \dots (1-24)$$

It follows from Equations 1-18, 1-22, 1-23, and 1-24 that

$$\rho = \frac{k_c - k_p}{k_c} \quad \dots (1-25)$$

Assuming that the static calculations have been performed with the prompt neutron spectrum only, i. e., with $\beta = 0$, then

$$\Lambda = \frac{k_c - k_p}{k_c \omega} \quad (\omega \text{ small}) \quad \dots (1-26)$$

Figure 1-12. Theoretical Inhour Curve

Figure 1-13. Theoretical Determination of Generation Time

The point of this exercise is that if flux perturbations caused by changes in L and ω are not so large as to change $\nu \Sigma_f$ and Σ_a significantly, then Λ can be calculated from the same one-dimensional calculation which determines ω .

As a practical matter, it is necessary to choose ω to give values of $k_c - k_p$ large enough to be calculated accurately. The error in the convergence of a transport code such as DTF-II is typically 1/2 to 2 times the convergence criterion, ϵ . Since $\epsilon = 1.0 \times 10^{-4}$ was used in these calculations, this means that $k_c - k_p$ on the order of 0.05 can be calculated with about $\pm 0.4\%$ accuracy, but $k_c - k_p$ of 0.001 is only accurate to $\pm 20\%$. Calculations using DTF-II and the 23-group cross sections were performed for $0 \leq \omega \leq 5000 \text{ sec}^{-1}$. A theoretical "inhour curve" was then produced by artificially adding β to the ρ calculated with $\beta = 0$ and converting to $\$$ units. This result, shown in Figure 1-12, shows good linearity between 0 and $\$2.00$, with some departure from linear at large ρ . It would be possible to use only data for small ρ , but these are the least accurate data. Instead, recourse was made to the model described by Gamble⁽²⁰⁾ to describe a similar nonlinearity in the KEWB inhour curve. In the AI notation, and for $\omega \gg \lambda_i$

$$\rho = \left[\Lambda = \frac{C_7}{\lambda_7} \frac{\omega}{(\lambda_7 + \omega)} \right] \omega + 1 \quad \dots (1-27)$$

In this formulation, the reflector-moderated neutrons are treated as an additional group of delayed neutrons, described by the parameters λ_7 and C_7 . If the linear portion of the inhour curve is not exceeded by far, it is reasonable to assume that $\lambda_7 \gg \omega \gg \lambda_i, i=1,2,\dots,6$. Then

$$\frac{\rho - 1}{\omega} = \Lambda = \frac{C_7}{\lambda_7} \omega \quad \dots (1-28)$$

This result is shown in Figure 1-13, as well as a least-squares fit weighted to account for the larger uncertainty at smaller ω values. This shows that

$$\Lambda = 6.47 \text{ } \mu\text{sec}$$

$$\frac{C_7}{\lambda_7} = 8.49 \times 10^{-5} \text{ } \mu\text{sec}^2 \quad .$$

It is also of interest to consider the value of C_7 and λ_7 , but one of these parameters must be calculated separately. The method used here was to repeat the above calculation with the poison in the core alone. This resulted in a value of Λ which accounts for time spent by neutrons in the core alone. This value, Λ_c , is $\Lambda_c = 5.06 \text{ } \mu\text{sec}$. If this is identified with the "mean lifetime of neutrons in the core" in Gamble's model,

$$\Lambda = \Lambda_c + C_7/\lambda_7 \quad , \quad \dots (1-29)$$

then $C_7/\lambda_7 = 1.41 \text{ } \mu\text{sec}$. And from the results of Equation 1-28,

$$l_7 = 1/\lambda_7 = 60 \text{ } \mu\text{sec}$$

$$\alpha_7 = C_7/\beta = \$2.94.$$

A somewhat different result is obtained if reflector motion is assumed to be equivalent to a change in reflector thickness. This is discussed in more detail later. Attempts were also made to analyze the SNAPTRAN results in terms of Gamble's "seventh-group" model. Although it correctly represented the inhour curve over the range of the experiments, it was inadequate to describe the kinetic behavior during the rapid temperature rises, and a detailed spatially-dependent-kinetics treatment was required.

C. POWER DISTRIBUTION PROFILES

The calculation of the axial power profile, $\psi_z(z)$, was performed with a DTF-II calculation using the 23-group library and the core model given in Table 1-6. Figure 1-14 shows the resulting curve. The radial profile was originally calculated using the same library and the radial model given in Table 1-5. It was later decided that the beryllium cross sections in the 23-group library

were not adequate, and a newer 26-group library was used. This result is shown in the first graph in Figure 1-15.

In addition, a correction was made to account for the fact that the power distribution is not quite independent of azimuthal angle. This is because the core is really hexagonal, not circular. Power traverses made on the SCA-4A critical assembly were used to make this correction. (The SCA-4A had a SNAP 10A/2-type core, similar to SNAPTRAN-1.) The profiles used in making the correction are shown in Figure 1-15b. The lower of the two curves was taken along a radius toward a corner of the hexagon, while the higher was taken toward the midpoint of a side of the hexagon.

The ϕ^{K_i} s determined by Equation 1-5 were adjusted slightly to account for their azimuthal position, by considering the azimuthal dependence of power to vary linearly between the limiting curves shown in the figure. This correction was as large as 10% for probes at the outer edge of the core, but was negligible away from the core periphery.

Figure 1-14. Calculated Axial Power Profile

Figure 1-15. Calculated and Measured Radial Power Profiles

III. RESULTS

A. DRUM CALIBRATIONS

The pertinent drum calibrations are summarized in Figure 1-16. These curves give the integral worths of the drums for each of the three types of transients performed. The details concerning the drum calibrations can be found in Reference 15.

An error analysis was performed for both the single-drum and the coupled-drum calibrations. The data from the calibration of Drum A were used for the single-drum case, while the data from the calibration of Drums B and C were used for the coupled-drum case. The uncertainties in the corresponding integral reactivity worth curves as a function of drum position are shown in Figure 1-17. These uncertainties were determined using the technique described in Sect. II. A. 7. As can be seen from the figure, the uncertainty in the coupled-drum calibration per drum is more than twice as large as that for the single-drum calibration. The coupled-drum worth results are less accurate because much smaller incremental angles must be employed in the calibration process.

The possibility of control drum interaction must also be considered when using the "rod bump" technique for calibrating. This arises from the fact that as the drums being calibrated are inserted, the other drums are being withdrawn to keep the reactor near critical. The resulting spatial change in flux and, therefore, neutron importance can cause the worth of the drums being inserted to be overestimated.

A series of experiments to help evaluate this effect was performed on the SNAPTRAN-1 assembly. The results of those experiments, along with a discussion of the interaction effect, are presented in Reference 21. It is concluded in that study that the interaction effect is significantly less than the calibration uncertainty due to inherent experimental error.

B. STEP TESTS SIGNIFICANTLY ABOVE PROMPT CRITICAL ($\rho > \$1.10$)

A total of 33 step transients were run with reactivity insertions larger than \$1.10. These included 10 transients initiated from elevated temperatures. A summary of the analysis of these tests is presented in Table 1-7. A description of the format used in the summary table is given in Table 1-8.

Figure 1-16. Integral Drum Worth Curves for SNAPTRAN-1

Figure 1-17. Uncertainty Estimate for Integrated Worth Curves

TABLE 1-7
RESULTS FROM STEP TESTS ABOVE PROMPT CRITICAL

Trans- ient No.	Date of Test	Peak Power (Mw)	Energy Release to Peak Power (Mw-sec)	Drum- Inserted Reactiv- ity (\$)	Period (msec)	Initial Temp. (°F)	Avg. Temp. Rise to Peak Power (°F)	Normalized Peak Power (Mw)	Prompt Temp. Coef. ($^{\circ}\text{F}$)	Range of Temp. Coefficient Calculation (°F)	PCAL	Inverse Period (msec ⁻¹)	Mean Temp. $\bar{T}$ (°F)
128	11-12-64	64.0	1.16	1.10	8.3	77.	80	42.0	-0.121	130 - 219	1.524	0.12048	174.5
131	11-18-64	151.3	1.57	1.18	5.15	72.	121	110.5	-0.125	124 - 286	1.369	0.19417	205.
133	11-20-64	375.8	2.55	1.26	3.2	82.	184	304.4	-0.136	132 - 410	1.235	0.31250	271.
136	12-11-64	656.0	2.58	1.32	2.4	73.	247	544.4	-0.151	123 - 500	1.205	0.41666	311.5
138	12-14-64	787.0	3.48	1.35	2.2	66.	266	617.7	-0.136	118 - 523	1.274	0.45454	320.5
140	12-16-64	982.0	3.96	1.39	2.0	70.	274	755.6	-0.149	125 - 395	1.300	0.50	260.
142	12-18-64	1280.0	4.65	1.50	1.7	52.	315	953.6	-0.150	103 - 510	1.342	0.58823	306.5
144	12-22-64	1267.0	4.51	1.48	1.8	62.	289	990.1	-0.144	115 - 604	1.280	0.55555	360.
159	01-14-65	663.0	3.30	1.31	2.55	82.	236	498.8	-0.148	136 - 412	1.329	0.39215	274.
200	04-22-65	650.0	3.61	1.38	2.2	66.	203	421.0	-0.172	118 - 406	1.544	0.45454	262.
201	04-23-65	1385.0	6.27	1.48	1.77	68.	296	888.9	-0.152	120 - 451	1.558	0.56497	285.5
203	04-28-65	1487.0	6.37	1.52	1.675	70.	317	981.4	-0.146	122 - 616	1.515	0.59701	369.
204	04-29-65	1480.0	5.52	1.52	1.65	75.	311	997.1	-0.145	126 - 667	1.484	0.60606	396.5
205	04-30-65	1727.0	6.42	1.53	1.57	75.	330	1140.1	-0.150	126 - 652	1.515	0.63694	389.0
206	05-03-65	2267.0	7.43	1.57	1.42	50.	366	1373.9	-0.155	102 - 702	1.650	0.70422	402.
207	05-04-65	2120.0	6.53	1.58	1.45	55.	351	1374.6	-0.155	107 - 701	1.542	0.68965	404.
209	05-05-65	1225.0	4.87	1.47	1.77	63.	292	838.9	-0.146	114 - 565	1.460	0.56497	339.5
210	05-05-65	63.0	1.1	1.10	9.17	432.1	122	-	-	486 - 549		0.10905	517.5
211	05-06-65	1587.0	11.02	1.50	1.59	82.5	311	1069.2	-0.159	135 - 623	1.484	0.62893	379.0
212	05-06-65	33.9	0.50	1.1	8.1	696.5	121	-	-	812 - 834		0.12345	823.
213	05-06-65	740.0	3.8	1.35	2.3	630.2	181	584.2	-0.164	708 - 969	1.267	0.43478	838.5
246	08-02-65	1550.0	5.27	1.52	1.69	75.	292	909.7	-0.173	150 - 563	1.704	0.59171	356.5
247	08-02-65	1258.0	4.99	1.42	1.94	598.8	202	797.0	-0.181	650 - 965	1.578	0.51546	807.5
248	08-02-65	330.0	2.77	1.20	4.18	900.	101	221.5	-0.193	956 - 1001	1.490	0.23923	978.5
249	08-03-65	1671.0	5.63	1.52	1.61	231.	276	1041.0	-0.182	288 - 660	1.605	0.62111	474.
250	08-03-65	958.0	4.87	1.36	2.25	693.	169	593.1	-0.181	745 - 1000	1.615	0.44444	872.5
251	08-03-65	376.0	3.03	1.20	4.0	911.	109	249.1	-0.189	963 - 1001	1.509	0.25	982.
252	08-03-65	253.0	2.62	1.15	4.9	1072.	84	164.1	-0.191	1123 - 1234	1.542	0.20408	1178.5
253	08-05-65	3200.0	8.1	1.68	1.21	95.	405	1995.5	-0.182	145 - 762	1.604	0.82644	453.5
254	08-05-65	855.0	3.24	1.36	2.27	642.	175	579.1	-0.175	694 - 956	1.476	0.44052	825.
255	08-05-65	315.0	2.50	1.19	4.18	900.	100	228.9	-0.196	955 - 1000	1.376	0.23923	977.5
256	08-05-65	241.0	2.46	1.14	5.29	1061.	79	153.8	-0.188	1114 - 1208	1.567	0.18903	1161.0

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TABLE 1-8
DESCRIPTION OF DATA SUMMARY TABLES
(See Tables 1-7, 1-10, and 1-11)

Column Designation	Description
Transient Number	Identification number assigned to each test
Date of Test	Date the test was performed
Peak Power (Mw)	Peak power in Megawatts, based on PPCo nuclear detector calibrations
Energy Release to Peak Power (Mw-sec)	Integral of the reconstructed power trace up to the time of peak power in units of megawatt-seconds
Drum-Inserted Reactivity (\$)	Reactivity insertion in units of dollars, based on drum position data and drum calibration tables. On step tests, it is the constant inserted reactivity, while on impulse tests it is the peak of the insertion.
Period (msec)	On step tests, this gives the stable period in milliseconds. On impulse tests, the minimum period is given.
Initial Temperature (°F)	Temperature of core when test was initiated
Average Temperature Rise to Peak Power (°F)	Rise in volume-weighted average temperature up to the time peak power occurred
Normalized Peak Power (Mw)	This column gives peak power normalized to the energy probe data.
Prompt Temperature Coefficient (ϕ /°F)	Prompt temperature coefficient based on volume-weighted average temperature from energy probe data
Range of Temperature Coefficient Calculation (°F)	This gives the range over which both the temperature coefficient and power normalization calculations were performed.
PCAL	A factor which must be divided into the power data to bring the energy from power into agreement with the energy from the probes
Calculated Lifetime (μ sec)	Prompt neutron lifetime calculated from individual impulse transients
Mean Temperature $\bar{T}$ (°F)	$1/2 (T_i + T_f)$ where T_i = volume-average temperature before transient and T_f = volume-average temperature after transient

An inhour plot based on a set of results from the step tests above prompt critical is shown in Figure 1-18. A straight line has been fit to the data using the method of least squares. Since all these transients were well above prompt critical, the slope of this line is just the ratio of the prompt neutron lifetime to the effective delayed neutron fraction (Λ/β). The resulting slope of the line was found to be $(8.3 \pm 0.1) \times 10^{-4}$ sec, which gives a prompt neutron lifetime of 6.6 ± 0.1 μ sec, based on a β_{eff} of 0.008. Both of these results are for 95% confidence intervals.⁽²²⁾

The data from the 10% hydrided energy probes were used directly in the thermal analysis of these transients. An examination of the plot of energy from power vs energy from energy probe data lent confidence that the energy probe time response was adequate and heat losses were negligible. The relationship was linear throughout the usable data range.

Energy profile plots for Transients 136 and 203 are shown in Figure 1-19. Both axial and radial profiles have been included. The axial profile has been compared with data from the unhydrided multi-probes with 5 wt % uranium

Figure 1-18. Inhour Data From Step Tests Above Prompt Critical

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Figure 1-19. Comparison of Calculated Power Profiles with Energy Probe Data From Tests 136 and 203

concentration. These probes were the only ones located at a sufficient number of axial positions to allow a meaningful comparison with the calculated profile.

As can be seen from the figure, the axial profile compares well with the data from Transient 136. This was found to be typical for the transients in the initial step series. A much larger amount of scatter is observed in the data from Transient 203, however, and the comparison is less conclusive.

The energy probe results were also compared with those obtained from the thermocouples which had been spot-welded directly to the fuel. A comparison between the average core temperature rise determined from TC-61 and TC-65 and that calculated from the energy probe data is given in Table 1-9. Although the spot-welded thermocouple data were available in only a limited number of tests, examination of the table shows that they are in excellent agreement with the energy probe results.

The cascaded step tests provided temperature coefficient data over a broad range of temperature. In Figure 1-20, plots of compensated reactivity vs average core temperature change are shown for cascaded step series Tests 246, 247, and 248. As can be seen from Table 1-7, these three tests cover a range of average core temperature from 75 to 1100°F. In each of the transients, the

TABLE 1-9
COMPARISON BETWEEN ENERGY PROBE AND
SPOT-WELDED THERMOCOUPLE DATA

Test	$\Delta \bar{T}$ Using 10% EP's (°F)	$\Delta \bar{T}$ Using TC-61 and TC-65 (°F)
200	284	280
201	420	422
203	493	496

Note: These $\Delta \bar{T}$'s give the temperature rise to some arbitrarily chosen time, and do not represent the total temperature rise for these tests.

3 31 67 UNC

7700 25109

Figure 1-20. Temperature Coefficient Results From Cascaded Step Test Series Composed of Tests 246, 247, and 248

relationship between compensated reactivity and temperature change is seen to be fairly linear. However, the slopes (i. e., constant temperature coefficients) are seen to increase slightly in successive tests, as T_i increases. The linearity observed in the compensated reactivity vs temperature change curves was typical for all the tests in this series. The seeming paradox posed by coefficients which do not vary with temperature change but do vary with initial temperature was deduced to be one of the results of the strong spatially-dependent-kinetics found in these experiments. This is discussed in Part 3 of this report.

C. STEP TESTS NEAR OR BELOW PROMPT CRITICAL ($\rho \leq \$1.06$)

The results obtained from transients initiated by step reactivity insertions of $\$1.06^*$ or less are presented in this section. Thirteen transients were analyzed, with reactivities ranging upward from $\$0.29$. The pertinent results from this analysis are summarized in Table 1-10. A description of the format used in this summary table is presented in Table 1-8.

A plot of drum-inserted reactivity vs reciprocal period for all the step transients is shown in Figure 1-21. This figure also includes the data from the super prompt critical step tests which were discussed in the previous section. A plot of the standard inhour equation with $\beta_{\text{eff}} = 0.008$ and $\Lambda = 6.6 \times 10^{-6}$ μsec is also shown in the figure. It is seen to be in excellent agreement with the experimental data throughout the entire range.

The main reason for separating the results presented in this section from those obtained from the shorter-period step tests was that an entirely different analytical technique was required to determine the average core temperature. The reason for this can be seen by examining Figure 1-22. The data presented are for Test 116, and are typical for the transients in this series.

The figure shows average core temperature as determined from the three potential sources of thermal information. Trace Number 1 was calculated from the power, using the Einstein oscillator heat capacity fit described earlier.

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*Reactivities are excess reactivity above delayed critical.

TABLE 1-10
RESULTS FROM STEP TESTS NEAR OR BELOW PROMPT CRITICAL

Transient No.	Date of Test	Peak Power (Mw)	Energy Release to Peak Power (Mw-sec)	Inserted Reactivity (\$)	Period (msec)	Initial Temp. (°F)	Avg. Temp. Rise to Peak Power (°F)	Normalized Peak Power (Mw)	Prompt Temp. Coef. (β /°F)	Range of Temp. Coefficient Calculation (°F)	PCAL	Mean Temp. $\bar{T}$ (°F)
111	09-28-64	0.124	2.24	0.485	6670.	72	186	0.0907	-0.13	111 - 420	1.367	265.5
112	09-30-64	0.333	2.44	0.66	2410.	69	210	0.243	-0.14	109 - 525	1.370	317.0
113	10-02-64	0.713	2.24	0.80	1020.	67	244	0.521	-0.12	113 - 328	1.369	220.5
115	10-09-64	1.31	1.87	0.91	340.	64	178	0.976	-0.14	111 - 431	1.342	271.
116	10-15-64	2.04	1.58	0.96	152.	65	130	1.59	-0.132	119 - 300	1.283	209.5
119	10-27-64	3.12	0.62	0.99	50.6	83	53	2.35	-0.141	120 - 334	1.328	227.
121	10-30-64	10.3	0.48	1.03	22.0	78	30	8.01	-0.117	112 - 174	1.286	143.
124	11-06-64	9.8	0.46	1.02	22.0	61	40	7.64	-0.112	99 - 199	1.283	149.
126	11-10-64	30.9	0.72	1.06	12.0	79	60	24.0	-0.122	114 - 192	1.288	153.
147	01-05-65	7.57	0.42	1.02	26.3	71	36	5.83	-0.120	107 - 186	1.298	146.5
149	01-06-65	6.15	4.41	1.02	31.4	84	35	4.51	-0.119	110 - 153	1.364	131.5
155	01-12-65	0.038	1.40	0.29	21,750.	75	163	0.028	-0.110	94 - 244	1.357	169
157	01-13-65	0.650	3.30	0.72	1490.	79	264	0.48	-0.125	109 - 371	1.354	24.0

Note: The average temperatures used to obtain the results given for Transients 111 through 115 were obtained using "Plan A" described in Section II. The remainder of the results were based on "Plan B".

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Although the magnitude of this trace is somewhat uncertain (since it depends on both the power calibration and heat capacity), its time profile should be quite accurate during the time of rapid temperature rise. This latter restriction is mentioned because of the possibility of a delayed gamma contribution to the nuclear detector signal on the back side of the power burst.

Examination of Trace Number 2 in the figure shows the difficulty encountered with the internal diaphragm-ended fuel thermocouples. As can be seen, these thermocouples showed large time lags, even on relatively long-period transients, such as the one being considered. These thermocouples had time constants on the order of 1 sec, and thus could not be used to obtain prompt temperature data of the type required in making a reactivity feedback analysis.

The third trace shows the average core temperature as determined from the 10% hydrided energy probe data. As can be seen, this temperature trace has a completely different shape than the other two. This is due to effect of heat flowing from the small fuel sample in the energy probe to the stainless-steel trough and sheath which surround it. Eventually, the sheath, trough, and fuel sample will be heated by the surrounding fuel elements and reach thermal equilibrium with them, but this has not yet occurred in Figure 1-22.

During the time of the power burst, the junction temperature in the energy probe was somewhat lower than that of the core fuel, due to the heat loss from the fuel sample. On all transients except those described in this section, the periods were sufficiently short to make the heat loss negligible over the time interval of interest. On the transients being considered here, however, there were no directly measured temperature data that could be applied in making a temperature coefficient analysis.

Two approaches were taken to circumvent this problem. The original results⁽²³⁾ were determined by correcting the energy probe data for heat losses. This was done using the approach described as "Plan A" in Part I, Section II. A. 6 of this report. This approach was used to obtain the average temperature rises, normalized powers, and temperature coefficients for Transients 111 through 115 in Table 1-10.

A sample calculation using this model is shown in Figure 1-23. The original and corrected data from EP-13 are compared to the normalized temperature calculated from power on Transient 113. The agreement is seen to be quite good up to the time the reactor was scrammed. Although this model appeared to give reasonable results, it was probably somewhat oversimplified in that the sheath temperature was held constant and the only heat loss path from the fuel sample was through the supporting trough.

The second approach made use of nuclear detector calibrations determined from the short-period step transients discussed in the previous section. This approach was described as "Plan B" in Part 1, Section II. A. 6. It was felt that this approach was more reliable than the heat loss correction on those transients which had several power channels that could be normalized and their results averaged. Transients 116 through 157 fall into this category; and the average temperature, normalized power, and temperature coefficient results were determined in this manner.

D. IMPULSE TESTS

The results obtained from the single-drum and coupled-drum impulse tests are summarized in Table 1-11. A description of the format used in this summary table is presented in Table 1-8. A detailed analysis was performed only for those transients which could provide useful lifetime and/or reactivity feedback information. A complete tabulation of all impulse transients performed, including those of low energy yield, is given in Reference 1.

The first series of single-drum impulse transients consisted of Tests 163 to 178. In the first 8 of these transients, the reactivity insertions were increased in successive transients to the maximum single-drum worth of \$2.37. The starting powers for these transients were sufficiently low (between 0.001 and 0.006 watts on the latter 6 tests) to keep the total energy release below 2 kw-sec.

The remaining 7 tests in this series were all initiated by the maximum reactivity insertion of \$2.37. Starting powers were successively increased from 0.01 to 41 watts. The last few tests experienced sufficient rise in average core temperature to allow the possibility of temperature reactivity feedback calculations.

Figure 1-23. Energy Probe Heat Loss Correction

TABLE 1-11
RESULTS FROM IMPULSE TESTS

Transient No.	Date of Test	Peak Power (Mw)	Energy Release to Peak Power (Mw-sec)	Drum-Inserted Reactivity (\$)	Period (msec)	Initial Temp. (°F)	Avg. Temp. Rise to Peak Power (°F)	Normalized Peak Power (Mw)	Prompt Temp. Coef. ($\beta/^\circ\text{F}$)	Range of Temp. Coefficient Calculation (°F)	Generation Time (μsec)	PCAL	Inverse Period (msec ⁻¹)	Mean Temp. $\bar{T}$ (°F)
First Series of Single-Drum Impulse Tests:														
177	02-12-65	1623	2.78	2.37	0.59	95.8	245.	-	-	-	-			
178	02-15-65	1975	3.42	2.37	0.59	68.3	224.	-	-	-	6.6×10^{-6}			
First Series of Coupled-Drum Impulse Tests:														
197	04-19-65	13	0.02	3.85	0.27	61.5	0	-	-	-	6.7×10^{-6}			
198	04-19-65	500	0.83	3.99	0.27	61.8	23.3	-	-	-	6.5×10^{-6}			
Second Series of Single-Drum Impulse Tests:														
215	06-11-65	750	1.42	2.37	0.61	70.0	92.5	-	-	-				
217	06-16-65	1048	1.89	2.37	0.60	60.0	114.4	-	-	-	6.3×10^{-6}			
220	06-18-65	1873	3.16	2.37	0.61	61.5	149.5	-	-	-	6.25×10^{-6}			
221	06-18-65	1850	3.18	2.37	0.61	82.0	190.	1163.	-0.187	134 - 415	6.42×10^{-6}	1.591	1.64	274.5
222	06-21-65	2410	3.80	2.37	0.60	80.	224.	1547.	-0.196	135 - 479	6.3×10^{-6}	1.558	1.67	307.
223	06-22-65	2543	4.05	2.37	0.60	78.	260.	1784.	-0.177	131 - 521	6.4×10^{-6}	1.425	1.67	326.
224	06-22-65	2910	4.49	2.37	0.60	120.	259.	1917	-0.194	170 - 557	6.4×10^{-6}	1.518	1.67	363.5
225	06-23-65	3307	5.10	2.37	0.63	85.	406.	4418	-0.176	145 - 639	6.4×10^{-6}	0.7485	1.59	392.0
226	06-23-65	3783	5.65	2.37	0.62	120.	328.	2569	-0.172	192 - 685	6.6×10^{-6}	1.473	1.61	438.5
227	06-24-65	4543	6.65	2.37	0.60	90.0	310.	1656	-0.172	139 - 630	6.3×10^{-6}	2.743	1.67	384.5
Second Series of Coupled-Drum Impulse Tests:														
231	07-06-65	67	0.88	3.86	0.28	73.4	6.2	-	-	-	6.4×10^{-6}			
232	07-07-65	141	1.90	3.97	0.27	70.8	11.2	-	-	-	6.4×10^{-6}			
233	07-08-65	445	0.55	3.75	0.28	78.	24.2	-	-	-	-			
234	07-09-65	826	1.03	3.75	0.29	72.7	45.3	-	-	-	6.5×10^{-6}			
235	07-12-65	1267	1.11	3.75	0.29	30.	100.0	-	-	-	6.4×10^{-6}			
236	07-12-65	1463	1.83	3.75	0.29	100.	120.5	-	-	-	6.6×10^{-6}			
237	07-13-65	2190	2.58	3.75	0.29	80.	153.	-	-	-	6.6×10^{-6}			
238	07-13-65	2157	2.30	3.75	0.29	103.	280.	-	-	-	6.5×10^{-6}			
240	07-15-65	3083	3.23	3.71	0.28	85.	203.	-	-	-	6.2×10^{-6}			
241	07-16-65	3733	4.01	3.75	0.29	86.	250.	2500	-0.173	138 - 442	6.4×10^{-6}	1.493	3.45	290.
242	07-16-65	3700	4.16	3.75	0.29	103.	193.	2302	-0.175	156 - 498	6.7×10^{-6}	1.607	3.45	327.0
243	07-19-65	4267	4.74	3.75	0.29	78.	269	2787	-0.186	131 - 539	6.3×10^{-6}	1.531	3.45	335.
244	07-20-65	5720	6.56	3.75	0.29	76.	376.	4133	-0.158	130 - 697	6.4×10^{-6}	1.384	3.45	413.5

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Figure 1-24. Drum Data Interpretation for Tests 177 and 178

Figure 1-25. Reactivity Comparison for Test 178

Difficulty in analyzing these transients was encountered because of an apparent time lag in the digitized drum position data. The reason for this lag was traced to the use of an electronic noise filter in the analog-to-digital data conversion process. The original analog data had a large noise content that appeared to be strongly biased in the direction of lower magnitude. These downward noise spikes were possibly produced by the wire wipers in the drum position potentiometers momentarily breaking contact with the resistance coil wires as they rapidly passed over them.

The drum data were originally filtered with a cutoff frequency of 0.078 kc. Although this filter arrangement removed a large portion of the noise, the filtered data trace was inevitably biased in the same direction as the noise spikes. A diagram illustrating this situation is shown in Figure 1-24. As can be seen from the figure, a shift in the magnitude of the data cannot readily be distinguished from a displacement in time. When the drum data were converted to reactivity and compared to the reactivity calculated from power in the region of no temperature feedback, the result shown in Figure 1-25 was obtained for Test 178. The apparent time shift between the two reactivity curves led to the conclusion that the data filtering was leading to erroneous results.

The data were subsequently redigitized with the filter cutoff frequency increased to 1.25 kc. This left much of the noise in the digitized data, but largely eliminated the shift in magnitude described above. A comparison between data filtered at 0.078 kc and 1.25 kc is shown in Figure 1-26. The data are for Drum A on Test 178.

The redigitized drum data, of the type shown in Figure 1-26b, had a noise content too large for direct use in the analysis. Application of the usual numerical smoothing techniques would have introduced errors similar to those of the low-frequency filter. A method of smoothing the data that appeared satisfactory is indicated in Figure 1-24. In this approach, the downward bias of the noise was considered, and a smooth curve was drawn through the uppermost portions of the data. The resulting reactivity trace as determined for Transient 178 is shown in Figure 1-25.

a. DATA FILTERED AT 0.078 kc

3-31-67 UNC

b. DATA FILTERED AT 1.25 kc

7700-25115

Figure 1-26. Effect of Electronic Noise Filtering on the Drum Data From Test 178

A detailed analysis was conducted for Transients 177 and 178, using the method described above to interpret the drum data. The results of this analysis have been included in Table 1-11. These tests were the last two in the series; they had the largest energy releases.

The first series of coupled-drum impulse transients consisted of Tests 180 through 198. The first 12 tests were initiated by reactivity insertions ranging from \$2.00 to \$3.50. These transients were all initiated at low power levels, and the resulting energy releases were all less than 1 kw-sec. Thus, there was no significant temperature rise on any of these transients.

The final seven tests were initiated by peak reactivity insertions ranging up to \$4.00. Although significant temperature increases were obtained from these tests, none were sufficiently large to allow a meaningful temperature coefficient analysis.

Useful inhour data were obtained from those transients with sufficient power rise to render an accurate determination of reactor period. A summary of the analysis of the final two tests in this series has also been included in Table 1-11.

Twelve more single-drum impulse tests were run during the extended test series. A detailed analysis of ten of these was performed; the results are summarized in Table 1-11. All of these transients were initiated by the maximum single-drum reactivity insertion of \$2.37. The initial powers ranged up to a maximum of 260 watts in the final test (227), producing average core temperatures up to 750°F.

The final series of impulse transients consisted of Tests 229 through 244. These were coupled-drum transients with the final eleven being initiated by reactivity insertions of \$3.75. Initial powers ranging from 1 mw up to 1.7 watts were used to produce maximum average core temperatures of up to 720°F. This corresponds to a temperature in the center of the core of about 1300°F. A summary of the analysis of the final 13 of these tests is also included in Table 1-11.

The calculated generation time given in the table were found by plotting the drum-inserted reactivity vs the inverse period for individual tests, as was described earlier. A 95% confidence interval for the means of these data is $6.44 \pm 0.06 \mu\text{sec}$.

IV. EVALUATION OF EXPERIMENTAL DATA AND INSTRUMENTATION

A. GENERATION TIME (Λ)

Two generation time results were presented in the previous section. The first of these was obtained by fitting a straight line to the inhour data from the step tests above prompt critical. The second result was obtained from the impulse test data, by averaging the slopes (Λ/β) of the super prompt critical reactivity vs reciprocal period data from the individual tests.

This latter approach was found to be somewhat inaccurate for two reasons. First, the period as a function of time was often quite inaccurate at the lower power levels on the front side of the power burst. Secondly, the result is sensitive to the temperature coefficient being used to correct the inserted reactivity for temperature feedback during the latter portions of the transient.

The first approach described is not sensitive to these inaccuracies. The asymptotic period resulting from a step reactivity insertion can be determined more accurately than the time history of the period resulting from an impulse insertion. Also, the asymptotic period can be determined before temperature feedback occurs. The high degree of linearity in the inhour data can be noted in Figure 1-18. Thus, the generation time of $6.6 \pm 0.1 \mu\text{sec}$ obtained from this approach is considered to be the most accurate result.

The accuracy of this result is limited, of course, by the accuracy of the drum calibrations, drum position data, and the ability to determine the asymptotic period. The magnitude of the random errors involved, such as those encountered in determining the drum positions and asymptotic periods, are reflected by the width of the confidence interval quoted above. Systematic errors, such as could be introduced by errors in drum calibrations, are not included in the confidence interval calculation, however. It is thus required to estimate the possible error in the drum calibrations used to obtain the inhour data in Figure 1-18.

The calibration curve being used was that for "B or C" in Figure 1-16 in the range from 30 to 40 degrees. According to the error analysis results presented in Figure 1-17, the uncertainty in the calibration curve in this range is about 3.7%. This gives an uncertainty in the integral worth curve of about $\pm 2.8\%$ in this range.

Figure 1-27. Evaluation of Inhour Data From Entire SNAPTRAN Test Series

Considering a possible systematic error of this magnitude, it would appear that a reasonable uncertainty band on the lifetime result would be $6.6 \pm 0.2 \mu\text{sec}$. Due to the intuitive nature of the error analysis, it is difficult to assign a precise confidence level to this interval. The level would appear to be quite high (greater than 90%), since the measurement uncertainties used as a basis for the error analysis were "outer limit", or high confidence, estimates.

B. APPLICABILITY OF SPATIALLY-INDEPENDENT (POINT) KINETICS MODEL

An inhour plot based on the data from all the tests above prompt critical is shown in Figure 1-27. The straight-line fit shown in Figure 1-18 has been included on this plot. The inhour curve calculated with the ANISN S_N transport theory code, ⁽²⁴⁾ using the AI 23-group thermalization library, ⁽¹⁹⁾ is shown as a dashed line in the figure. This curve was calculated using the $1/v$ poison technique discussed earlier.

The calculated curve shows somewhat more curvature than the data, although they both show the same qualitative deviation from the straight-line fit. The slope of the calculated curve below $\beta 1.50$ corresponds to a Λ of $6.5 \mu\text{sec}$, in good agreement with the data.

The downward drift of the data from the straight-line fit at large reactivities suggests the possibility of a reflector thermalization effect, similar to that observed in KEWB. ⁽²⁰⁾ It should be noted, however, that the magnitude of this drift is about the same as the uncertainty in the drum calibrations, thus making quantitative evaluation from these data difficult.

Thus, it would appear that the standard point kinetics model with $\Lambda = 6.5 \mu\text{sec}$ would be entirely adequate for calculating reactivity from asymptotic periods below 4000 sec^{-1} , with error due to reflector thermalization delay of less than 3%. It is shown in Part 3 of this report, however, that this is valid only during the time when reactivity is changing due to control drum movement alone. A more detailed treatment is required when temperature changes become important.

A comparison between the drum-inserted reactivity and the reactivity calculated from power for the coupled-drum impulse Test 232 and the single-drum

Figure 1-28. Comparison Between Calculated and Drum-Inserted Reactivities for Impulse Transients 224 and 232

Figure 1-29. Comparison of a SNAPKIN VI Forward Kinetics Calculation With the Experimental Results From Test 232

impulse Test 224 is shown in Figure 1-28. The reactivity calculated from power used the standard point kinetics model with a Λ of 6.6 μ sec. The calculated results cover the range of usable power up to the inception of significant temperature feedback. This range is quite large for Transient 232, as temperature feedback was negligible during most of the transient. The reactivity comparisons shown were typical for the respective types of tests.

For the single-drum transient, the calculated reactivity is about 4¢ larger than the drum reactivity at peak insertion. As can be seen from Figure 1-17, the maximum uncertainty in the single-drum calibration was estimated at 3¢. Thus, the disagreement between the two reactivity traces is of about the same magnitude as the uncertainty in the drum calibration.

The calculated reactivity for the coupled-drum transient is seen to be about 10¢ larger than the drum-inserted reactivity. Again referring to Figure 1-17, the uncertainty in the coupled-drum calibration was estimated at about 7¢. Thus, the disagreement at peak reactivity is again only slightly larger than the estimated uncertainty in the drum calibration. It should also be noted that the agreement is considerably better at lower reactivities.

The tendency for the calculated reactivity to be slightly higher may be an indication of a spatial kinetics effect. Although it is difficult to make a quantitative evaluation due to the uncertainties in the drum-inserted reactivity, it would appear that the 3% error limit for a \$4 reactivity insertion is borne out by these results.

A forward calculation of the power burst resulting from the drum-inserted reactivity for Transient 232 was made using the SNAPKIN VI⁽²⁵⁾ computer code. A generation time of 6.6 μ sec was used in this calculation. The resulting power is shown in Figure 1-29, along with the measured power burst. The curves have been normalized at peak power. The agreement is considered satisfactory, considering the estimated uncertainty in the drum-inserted reactivity used to initiate the forward calculation. The shape of the calculated power burst is seen to be in reasonable agreement with the experimental result, except that the measured power levels off at a considerably higher value. This is no doubt due to the gamma sensitivity of the nuclear detectors, as discussed earlier. In conclusion, it would appear that the point kinetics calculation is adequate for a

transient of this type, to within the uncertainty of this comparison, in the absence of substantial heating of the fuel.

C. TEMPERATURE COEFFICIENT

In order to evaluate the temperature coefficient data, it would appear appropriate to first consider the relative accuracy of the results obtained from the various types of tests.

There were three general categories of tests that provided temperature coefficient information. There were: (1) step tests below $\beta_{1.10}$, (2) step tests above $\beta_{1.10}$, and (3) impulse tests.

The step tests were broken into two classes because of the energy probe heat losses which occurred during the slower transients, as was described earlier. Because of the corrections required in interpreting the temperature data from the tests below $\beta_{1.10}$, the resulting temperature coefficients are considerably less accurate than those obtained from the step tests above $\beta_{1.10}$.

The possibility of axial heat conduction in the fuel must also be considered in the analysis of the long-period transients. The change in the axial peak-to-average temperature due to this effect was calculated with the CREKIN⁽²⁶⁾ computer program. The change in the axial peak-to-average temperature ratio at the time of peak power was found to be less than 2% for a 29¢ step reactivity insertion. Since this was the smallest reactivity insertion considered in this analysis, no adjustment in the axial profile was required to account for axial conduction. It is likely that heat loss to the fuel cladding caused errors of the order of 10% in the long-period results (80¢ and below), but no attempt was made to remove this effect analytically.

The temperature coefficient results obtained from the impulse transients are considerably less accurate than those obtained from either class of step tests. This is due to the larger uncertainty in determining the compensated reactivity for an impulse transient. Although it is difficult to determine quantitatively the uncertainty for the various types of tests, the above qualitative conclusions are borne out by the amount of scatter in the results.

In summary, the most accurate temperature coefficient results were obtained from the step tests above $\beta_{1.10}$. In the graphical presentation of the temperature

coefficient results, all the data from these transients have been plotted with solid plotting symbols. The results from the less accurate impulse and low-reactivity step tests have been plotted with open symbols.

The temperature coefficient was observed to generally become more negative as the test series progressed. This is shown in Figure 1-30, in which the temperature coefficient results are displayed as a function of transient number. It should be noted that only those transients which produced usable temperature coefficient results have been included and that, therefore, the abscissa does not have a numerically continuous scale. The increased scatter in the results from the impulse tests can also be seen in the figure.

Several causes for the changes observed in the temperature coefficient were considered. These included possible dependences on temperature, power, spatially dependent kinetics effects, or gradual changes in energy probe response or fuel properties. These considerations will be discussed in later sections.

Figure 1-30. Temperature Coefficient Results From SNAPTRAN Test Series

D. INSTRUMENT PERFORMANCE

To aid in the understanding of the results of this analysis, an evaluation of the performance of the instrumentation is presented in the following sections. Instrumentation for the three basic types of data (i. e. , nuclear, temperature, and drum position) will be considered.

1. Nuclear Detectors

In general, the nuclear instruments were found to perform adequately. No appreciable time delays were observed during these transients due to detector response or thermalization delays in the detector enclosures. Also, the detector range staggering appeared adequate to allow a good reconstruction of the power burst (in most cases).

The greatest area of difficulty with respect to the nuclear instruments was in their calibration. As was discussed earlier, there was a considerable variation in the magnitude of peak power as indicated by the various detectors during the early stages of the test series. This was due to inconsistent calibrations. The variation was reduced as the test series progressed by adjusting the calibration factors of the detectors to bring them into agreement with those detectors considered most reliable.

Because of difficulties encountered in obtaining the basic data used in performing the original calibrations, a check of these calibration factors was not made as a part of this analysis. It is thus difficult to assess their accuracy quantitatively, although the inconsistency between peak power values would indicate the possibility of a fairly large uncertainty.

The power normalization was checked against the energy probe normalization on each transient using the approach described earlier. The normalization factors were applied to the power data and the results included in the summary tables (see Tables 1-7, 1-10, and 1-11) as "Normalized Peak Power." On the average, the normalized peak powers are about 30% lower than the unnormalized peak powers.

One reason for this difference is that the nuclear detector calibrations were based on total "released" energy, while the energy probe normalizations were made on the basis of "absorbed" or sensible energy. To evaluate this effect, a

detailed study of the energy deposition was made. A summary of this study is presented in Table 1-12. Beta particles and gammas from fission product decay were assumed to deposit no significant energy in the core, due to the short duration of the transients being considered in this analysis.

The DTF-II S_N transport code⁽¹⁸⁾ was used to perform the gamma-ray flux calculations, utilizing the GAMLEG computer program⁽²⁹⁾ to generate the required cross-section data, and a buckling iteration was employed to yield an accurate two-dimensional gamma-ray flux profile.

The neutron energy deposition was computed using a modified version of the FORM computer program.⁽³⁰⁾ A consistent β_1 slowing-down calculation was performed to obtain the fast spectrum.

The results of this study indicated that 88.4% of the released energy would be absorbed in the core during a typical SNAPTRAN transient. Thus, this effect

TABLE 1-12
SNAPTRAN CORE ENERGY DEPOSITION (mev/fission)

<u>Available Energy:</u>	
Fission fragment K. E.	168.0
Prompt gamma ray sources	10.0
Neutron sources	4.9
Fission product decay gammas	7.0
Beta energy	6.0
Neutrino energy	<u>8.0</u>
Total	203.9
<u>Energy Absorbed:</u>	
Fission fragment K.E.	168.0
Gamma ray energy	8.5
Neutron energy	<u>3.9</u>
Total	180.4

TABLE 1-13
ENERGY PROBE INSTALLATION AND FAILURE HISTORY

Energy Probe Number EP-	Installation Date	Date of Failure	Transient Number Failure Observed	Energy Probe Number EP-	Installation Date	Date of Failure	Transient Number Failure Observed
1	09-64	12-16-64	138	23	09-64	None	
3	09-64	None		24	09-64	06-24-65	227
4	09-64	None		25	09-64	None	
5	09-64	04-30-65	205	26A	11-03-64	12-02-64	133
6	09-64	12-21-64	142	26B	11-03-64	None	
7	09-64	04-30-65	205	26C	11-03-64	None	
8A	11-03-64	05-05-65	209	26D	11-03-64	None	
8B	11-03-64	None		28	09-64	11-12-64	128
8C	11-03-64	None		29	09-64	10-27-64	119
8D	11-03-64	05-05-65	209	30	09-64	08-03-65	252
9A	11-03-64	12-14-64	136	31	09-64	12-10-64	133
9B	11-03-64	05-03-65	206	32	09-64	04-30-65	205
9C	11-03-64	08-03-65	252	34	09-64	05-06-65	213
10	09-64	05-06-65	213	35	09-64	None	
11	09-64	None		36	11-03-64	04-30-65	205
12	09-64	12-21-64	142	37	11-03-64	05-06-65	213
13	09-64	08-03-65	252	38	11-03-64	07-13-65	238
14	09-64	06-24-65	227	40	06-04-65	06-16-65	218
15	09-64	None		41A	06-04-65	None	
16	09-64	05-03-65	205	41B	06-04-65	07-08-65	233
17	09-64	07-07-65	232	41C	06-04-65	None	
20A	11-03-64	None		41D	06-04-65	None	
20B	11-03-64	05-06-65	213	42	06-04-65	06-23-65	225
20C	11-03-64	None		43	06-04-65	None	
20D	11-03-64	None		44	06-04-65	06-23-65	225
21	09-64	05-10-65	213	45	06-04-65	06-24-65	227
22A	11-03-64	None		46	06-04-65	06-24-65	227
22B	11-03-64	None		47	06-04-65	None	
22C	11-03-64	None		48	06-04-65	None	
22D	11-03-64	08-03-65	252	49	06-04-65	08-03-65	249

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accounts for less than half the difference noted between the nuclear detector calibrations and the energy probe normalizations. Another 3 to 4% of this difference may be accounted for by flux depression in the energy probes; this is the magnitude of the core-to-probe power disadvantage factor calculated for a cell centered around an interstice with an energy probe, using DTF S_4 transport theory.

In summary, the energy releases determined from the energy probe data and heat capacity model described earlier were found to run about 10 to 15% lower than the energies from integrated power using nuclear detector calibrations. This could be due to error in either the heat capacity, the power calibrations, or both. The heat capacity used has been shown to agree with published experimental data. Documentation to support the power calibrations is not available at the time of this report; consequently, their validity is not really known.

2. Energy Probes

The time response of the energy probes appeared to be adequately fast for the transients in this series. This was demonstrated by plots of energy in terms of energy probe vs integrated power. No nonlinearities in plots of this type were observed, even on the shortest-period transients.

Significant heat losses from the energy probe fuel samples were observed for those transients initiated by reactivity insertions below about \$1.10. This corresponds to periods longer than 10 msec. Thus, the energy probes were found to be unsatisfactory for transients in that range.

Resistance checks were made on all of the energy probes during the instrument calibration tests. When the resistance through the junction was observed to change significantly, the energy probe was removed from service. A summary of all the energy probes, giving the installation dates and failure dates, is given in Table 1-13. As can be seen, some of the energy probes survived the entire test series.

As was previously mentioned, the possibility of a change in the emf response of the energy probes as the test series progressed was suspected. To investigate this possibility, a post-test examination of the energy probes was conducted. In general, the probes were found to be in good mechanical condition. Figure 1-31 shows such a probe, EP-25, with the sheath and MgO powder removed.

7611-51103

Figure 1-31. EP-25 Disassembled After Use in SNAPTRAN-1 Test Series

The condition of this probe was typical of all those disassembled, except for those using epoxy for bonding the fuel to the trough instead of spot-welding. Only the first six probes built used this design. It was noted that in many cases the 0.5-mil thermocouple wires had fused into the surface of the fuel samples.

Several fuel samples from the used energy probes were mounted in epoxy and highly polished, and photomicrographs were taken. The resulting photomicrographs of the polished and etched cross sections of the fuel from EP-42 and EP-43 are shown in Figures 1-32 and 1-33, respectively. The material around the outside of the square fuel sample is the mounting epoxy used to hold the fuel for polishing.

The banded structure of the etched fuel indicates the presence of epsilon-phase hydride. It thus appears that these fuel samples had not experienced any serious hydrogen depletion. A certain amount of surface oxidation was evident, however, especially on the samples from EP-42.

It should be noted that both of these probes were installed late in the transient series, and their condition may not be typical for all the probes used.

A detailed photomicrograph of the surface oxidation on EP-42 is shown in Figure 1-34. The light gray area is the oxide material. The small white patches are caused by difficulties in the etching process. The depth of the oxidation varies up to about 5 mils.

3-31-67 UNC

7700-25120

Figure 1-32. Fuel Sample From EP-42 Used in SNAPTRAN-1
Test Series (100X)

3-31-67 UNC

7700-25121

Figure 1-33. Fuel Sample From EP-43 Used in SNAPTRAN-1
Test Series (100X)

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UNETCHED

ETCHED

3-31-67 UNC

7700-25122

Figure 1-34. Enlargement of Surface Oxidation
on EP-42 (500X)

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To investigate the condition of the fuel in the area of the thermocouple spot-welds, the photomicrograph shown in Figure 1-35 was taken. This shows a cross section taken through one of the spot-welds on EP-25. As can be seen, there appears to be increased oxidation in the area of the weld, as well as some evidence of fuel cracking. The depth of the oxidation is about 5 mils.

The possibility of a change in the emf response of chromel-vs-alumel thermocouples with high-temperature use has been pointed out in the literature.⁽³¹⁾ The emf responses of two energy probes which were not disassembled for examination were checked. One of the probes checked was EP-44, while the other probe could not be identified. The millivolt responses at the freezing point of tin (449.38°F) were found to be 9.450 and 9.410 mv, respectively. A similar check had been made on all the probes at the time of construction. Originally, EP-44 had given a response of 9.423 mv at the same temperature. All of the probes originally fell in the 9.4 ± 0.1 range. It was thus apparent that these two probes had not experienced any appreciable change in emf

3-31-67 UNC

7700-25123

Figure 1-35. Surface Conditions of EP-25 at
Position of a Spot-Weld (100X)

response at this temperature. It should also be noted that the loop resistance of EP-44 had been observed to change, and it had been considered unreliable after Test 225 (see Table 1-13). Further, this probe was installed in the core only before the extended test series, and thus was not exposed to the entire test series.

The possibility of probe degradation due to high-temperature exposure was also investigated by keeping several new probes at elevated temperatures in a furnace and monitoring the emf response as a function of time. It was found that the calibration of the probes did not change significantly during a time at high temperature of the same magnitude as experienced by the SNAPTRAN energy probes. When the calibration did change appreciably, there was also a marked change in loop resistance. When similar changes in resistance were observed in the SNAPTRAN probes, use of the affected probes was discontinued. Thus, probe degradation is not believed to have been a significant factor.

3. Drum Position Potentiometers

The quality of the drum-position data from the potentiometers was found to be less than satisfactory. As was discussed earlier (Section III.D), much of the data from the first few impulse tests had a large noise content in the direction of lower magnitude. This may have been due to the wiper in the potentiometer momentarily breaking contact with the resistance coil under rapid motion. The quality of the signal was improved on the latter transients by employing a different grounding arrangement.

Since the impulse drum travel was more than 360 degrees on the impulse tests, a multi-turn potentiometer would have been better than the single-turn type that was employed. This would have eliminated the voltage flip in the data with rotations of more than 360 degrees.

PART 2

SNAPTRAN-2 EXPERIMENT

NAA-SR-11850

2-1

2

3

4

5

I. INTRODUCTION

The purpose of the SNAPTRAN-2 test was to provide information on the behavior of a SNAP 10A/2-type reactor during a destructive transient. Assessment of the total energy release, mode of shutdown, and the degree of fission product release were of interest. The test was also to serve as an extension of the SNAPTRAN-1 transient series, described in Part 1 of this report, by providing kinetics and reactivity feedback information at a higher power and temperature level and on a shorter period than could be achieved with a nondestructive transient.

Part 2 of this report will deal primarily with the kinetics analysis of the data and mode of reactor shutdown. Unless otherwise noted, the analytical methods used are the same as those employed in the analysis of the SNAPTRAN-1 test series and are described in Part 1 of this report. The radiological aspects of the test are presented in Reference 32.

II. TEST FACILITY

A. TEST SITE

The destruct test was performed at the same site as the SNAPTRAN-1 transient series. The movable IET building was rolled back to expose the reactor during this test, however, as shown in Figure 1-1. A complete radiation monitoring grid was set up to measure and track the fission product gases.

B. CORE-REFLECTOR ASSEMBLY

The SNAPTRAN-2 destruct machine was similar in design to the SNAPTRAN-1 reactor described in Part 1. The impulse drum drive mechanisms were modified to allow a rapid (approximately 12-msec) step reactivity insertion. To accomplish this, the impulse drums were stopped at their "full in" position using pneumatic shock absorbers.

Although a new core was used for this test, its properties were the same as the SNAPTRAN-1 core, except for minor changes in the fuel composition. A summary of these changes is given in Table 2-1. A different control drum shimming configuration was also used.

C. INSTRUMENTATION

1. Power

A summary of the power instrumentation is given in Tables 2-2 and 2-3. The ranges covered by the various detectors were shown in Figure 1-6. A brief description of the nuclear detector calibration follows. More detailed discussions can be found in References 3 and 4.

Several nuclear detectors that had been calibrated during the SNAPTRAN-1 test series were used for the destruct test. Certain of these retained their same enclosures and geometrical positions, and were used as a basis for the SNAPTRAN-2 power calibration. The calibration of these base instruments was checked by operating the SNAPTRAN-2 reactor at a power level of about 700 watts, as determined from foil and wire activation. This check was applicable to only a few detectors, however, since it produced a significant deflection on only a few of the more sensitive power channels.

TABLE 2-1
 SNAPTRAN-2 CORE PROPERTIES* (Supplement to Table 1-1)

Number of fuel rods	37
Lattice spacing (in.)	1.26
Fuel rod OD (in.)	1.212
Fuel rod length (in.)	12.25
Weight of U ²³⁵ (gm)	4729.4
Total weight of fuel (gm)	51152.997
Average fuel density (gm/cm ³)	6.067
Average fuel composition: [†]	
Fuel alloy, wt % U in Zr	10
U ²³⁵ enrichment (%)	93
Atom densities (10 ²⁴ atoms/cm ³ of fuel):	
U ²³⁵	0.001438
U ²³⁸	0.0000886
Zr	0.03540
H	0.0653
H/Zr atom ratio	1.848

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* These properties are based on data from References 5, 6, and 7.

[†] Certain other materials present only in very small amounts have not been included here. They are given in Reference 5.

TABLE 2-2
SNAPTRAN-2 NUCLEAR INSTRUMENTATION

Detector	Type	Location*
N-1	WL-8105 (FN)	Enclosure 2
N-2	RSN-76A (FN)	Z = +5 in., R = 1 ft
N-3	RSN-76A (FN)	Enclosure 3
N-6	20th Cent. (FN)	Enclosure 3
N-8	20th Cent. (FN)	Enclosure 2
N-9	GE - gamma	Z = -3.5 ft, R = 7 ft
N-10	GE - gamma	Enclosure 1
N-11	GE - gamma	Z = +5 in., R = 1 ft
N-12	GE - (FN)	Enclosure 6
N-16	Scint. (FN)	Enclosure 5
N-18	Scint. (FN)	Enclosure 7
N-20	Scint. (FN)	Enclosure 2
N-22	Scint. (FN)	Enclosure 7
N-23	Scint. (FN)	Enclosure 1
N-25	Scint. (FN)	Enclosure 4
N-26	GE - gamma	Z = 40 ft between 3 and 4
N-27	GE - gamma	Enclosure 4
N-30	Scint. (FN)	Enclosure 7
N-32	GE - mini (TN)	In-core 6-7, Z = 0
N-33	GE - mini (TN)	In-core 6-5, Z = +3.1 in.
N-34	GE - mini (TN)	In-core 6-6, Z = 0
N-35	GE - mini (TN)	In-core 9-5, Z = 0
N-43	Scint. (FN)	Enclosure 5

* See Table 2-3 for a description of the enclosures.

TABLE 2-3
SNAPTRAN-2 NUCLEAR DETECTOR ENCLOSURE DESCRIPTIONS

Enclosure Number	Location	Distance From Center of Core (ft)		Enclosure Construction [†]
		R	Z*	
1	Flat car	9	- 8	2 in. Pb, covered with 0.02 in. Cd, placed on 0.02 in. Cd
2	North end of flat car	11	- 8	2 in. Pb, 0.02 in. Cd, 4 in. wax placed on 0.02 in. Cd
3	Ground next to test cell	40	-14	4 in. Pb, 0.02 in. Cd, 8 in. wax placed on 0.02 in. Cd and 2 in. Pb
4	Next to No. 3	40	-14	8 in. Pb, placed on 0.02 in. Cd and 2 in. Pb
5	Test cell floor	-14	-14	2 in. Pb
6	Coupling station			Polyethylene-borated-lead brick covered with 0.02 in. Cd placed on 0.02 in. Cd
7	Test cell floor	20	-14	2 in. Pb covered with 0.02 in. Cd placed on 0.02 in. Cd

* +Z is in the upward direction.

[†] The materials in multi-layer enclosures are given in order from inside out.

Several preliminary transients initiated by reactivity insertions in the \$1.02 to \$1.05 range were also performed. These resulted in peak power levels of up to about 25 Mw. The calibrations of the base instruments were then checked again by comparing the results from these transients with those obtained from SNAPTRAN-1 tests with comparable reactivity insertions. The comparison was found to be satisfactory, and no adjustments were made to the calibration factors as a result of these checks.

The final calibration of those detectors less sensitive than the detectors used as a calibration basis was not determined until the destruct transient was performed. The calibration factors of these less-sensitive detectors were then adjusted to bring them into agreement with the base detectors. This was done by adjusting successively less-sensitive channels to bring their power levels into agreement in the regions of overlap. This was similar to the power trace reconstruction process described in Part 1.

Detectors N-23 and N-43 were used as the calibration basis for the fast neutron scintillation detectors. These two detectors gave data which agreed within a few percent. The calibrations of N-16 and N-30 were then adjusted to give agreement with N-43. Both N-16 and N-30 gave peak power results on the destruct test.

N-9 was used as a basis for the gamma detector calibrations and N-10, N-26, and N-27 were adjusted to it. Similarly, N-2 was the calibration basis for the ion chambers, with N-1, N-8, N-6, N-12, and N-3 being adjusted as described above.

Before the final calibration adjustments were made for those gamma detector and ion chamber channels which showed peak power, the following time adjustment was made. The reconstructed power traces from the ion chambers and gamma detectors were compared with the reconstructed trace from the scintillation detectors, and the time shift (Δt) between these traces was determined. Before adjusting the calibration factors of the peak power channels, they were shifted in time so as to make the time differences between the peak powers be the same as the Δt 's observed at lower power levels.

Since it is impossible to distinguish between a time shift and a calibration error (except near peak power), the Δt 's determined at the lower power levels

TABLE 2-4
SNAPTRAN-2 ENERGY PROBE INSTRUMENTATION

Detector No.	Type*	Location	Detector No.	Type*	Location
EP-147	ZrH	10-4, Z=0	M10-2	ZrU _{2.5}	14-3, Z=4
EP-149	ZrH	8-4, Z=0	M10-3	ZrU _{2.5}	14-3, Z=2
EP-150	ZrH	12-4, Z=0	M10-4	ZrU _{2.5}	14-3, Z=0
EP-153	ZrHU ₁₀	8-4, Z=0	M11-1	ZrU _{2.5}	13-3, Z=6
EP-154	ZrHU ₁₀	10-4, Z=0	M11-2	ZrU _{2.5}	13-3, Z=4
EP-155	ZrHU ₁₀	12-4, Z=0	M11-3	ZrU _{2.5}	13-3, Z=2
M6-1	ZrU _{2.5}	9-4, Z=+6	M11-4	ZrU _{2.5}	13-3, Z=6
M6-2	ZrU _{2.5}	9-4, Z=4	M14-1	ZrU ₅	$\theta = 60, Z=6$
M6-3	ZrU _{2.5}	9-4, Z=2	M14-2	ZrU _{2.5}	$\theta = 60, Z=4$
M6-4	ZrU _{2.5}	9-4, Z=0	M14-3	ZrU _{2.5}	$\theta = 60, Z=2$
M8-1	ZrU _{2.5}	11-4, Z=6	M14-4	ZrU _{2.5}	$\theta = 60, Z=0$
M8-2	ZrU _{2.5}	11-4, Z=4	M15-1	ZrU ₅	$\theta = 60, Z=6$
M8-3	ZrU _{2.5}	11-4, Z=2	M15-2	ZrU ₅	$\theta = 60, Z=4$
M8-4	ZrU _{2.5}	11-4, Z=0	M15-3	ZrU _{2.5}	$\theta = 60, Z=2$
M9-1	ZrU _{2.5}	13-4, Z=6	M15-4	ZrU _{2.5}	$\theta = 60, Z=0$
M9-2	ZrU _{2.5}	13-4, Z=4	M16-1	ZrU ₅	$\theta = 60, Z=6$
M9-3	ZrU _{2.5}	13-4, Z=2	M16-2	ZrU ₅	$\theta = 60, Z=4$
M9-4	ZrU _{2.5}	13-4, Z=0	M16-3	ZrU ₅	$\theta = 60, Z=2$
M10-1	ZrU _{2.5}	14-3, Z=6	M16-4	ZrU ₅	$\theta = 60, Z=0$

* Subscripts refer to wt % of uranium in U-Zr alloy.

could be all or partially due to calibration discrepancies. Thus, all the calibration factors are ultimately based on the N-43 and N-23 scintillation detectors.

2. Temperature

All the temperature data used in this analysis were obtained from energy probes. A summary of the energy probes used is given in Table 2-4. As can be seen, only three of the probes had fuel of the same composition as the core. Most of the in-core probes had only 2.5% uranium content and were not hydrided. Three of the in-core probes were hydrided, but had no uranium content. Several unhydrided probes in the beryllium reflector used a 5% fuel concentration.

By using fuel of 2.5% uranium content in the probes, an energy deposition of roughly one-fourth that of the adjacent reactor fuel was reached during the prompt burst. Further, since these probes were unhydrided, they did not undergo a buildup of internal hydrogen pressure as the temperature increased. Thus, it was thought possible for such probes to produce usable data for a short time after fuel disintegration had ensued in adjacent fuel rods.

3. Drum Position

The drum position instrumentation was the same as that previously described in Part 1 (Section I.B.3) for the SNAPTRAN-1 test series.

4. Mechanical Measurements

To document the initial expansion of the core during the destruct transient, strain gages, radial and axial motion detectors, and acoustic detectors were installed on or in the area of the reactor. A summary of this instrumentation is given in Table 2-5.

Extensive photographic coverage was also made during the test. This included high-speed coverage of a cross-sectional view of the core (similar to Figure 1-5). The upper grid plate was removed and a strong light source was placed below the core to make the fuel rod interstices clearly visible. A summary of the photographic coverage is given in Table 2-6, while a complete description is given in Reference 33.

TABLE 2-5
 SNAPTRAN-2 MECHANICAL MEASUREMENTS INSTRUMENTATION
 (Sheet 1 of 2)

Detector No.	Type	Location
<u>Motion detectors:</u>		
DT-1	Linear motion pot.	Top of element, Position 5
-2		Top of element, Position 14
-3		Top of element, Position 29
-4		Top grid plate, 7-4
-5		Top grid plate, 7-5
-6		Top grid plate, 7-6
-7		Be reflector, $Z=0$, $\theta=0^\circ$
-8		Be reflector, $Z=0$, $\theta=90^\circ$
-9		Be reflector, $Z=0$, $\theta=180^\circ$
-10		Be reflector, $Z=0$, $\theta=270^\circ$
-11		Under dolly
-12		Top of element, Position 7
-13		Be reflector, $Z=0$, $\theta=0^\circ$
-14	Linear motion pot.	Be reflector, $Z=0$, $\theta=270^\circ$
<u>Acoustic detectors:</u>		
P-1	Kaman ac bridge	R=7.5 ft, side-on
-2		R=7.5 ft, face-on
-3		R=22.5 ft, side-on
-4		R=22.5 ft, face-on
-5		Z=30-ft radiological tower, side-on
-6	Kaman ac bridge	Z=30-ft radiological tower, face-on
-7	B & K microphone	R=100 m
-8	B & K microphone	R=300 m
<u>Strain gages:</u>		
SG-1	Axial A-D	7-4, $Z=-2$
-2	Axial A-D	7-5, $Z=-2$

TABLE 2-5
SNAPTRAN-2 MECHANICAL MEASUREMENTS INSTRUMENTATION
(Sheet 2 of 2)

Detector No.	Type	Location
SG-4	Axial A-D	8-5, Z= -2
-7	Axial A-D	13-3, Z= -2
-8	Axial A-D	14-2, Z= -2
-11	Circ. A-D	7-6, Z= -2
-13	Circ. A-D	13-3, Z= -2
-14	Circ. A-D	Vessel, Z=0, $\theta=0^\circ$
-15	Circ. A-D	Vessel, Z=0, $\theta=180^\circ$
-16	Hoop 180°	Vessel, Z= -0.5, $\theta=0$ to 180°
-17	Hoop 180°	Vessel, Z= -0.5, $\theta=180$ to 360°
-18	Dummy	Next to reactor
ICD-1S	Compression	7-4, Z=0
-2S		6-5, Z=1
-3S		7-5, Z=0
-4S		7-6, Z=0
-5S		5-5, Z=0
-7S		9-3, Z=0
-8S		10-1, Z=0
-9S		7-1, Z=0
-10S	Compression	Vessel, Z=0
AC-1	4-arm strain	Element, Position 1
-2	4-arm strain	Element, Position 20
-3	4-arm strain	Be reflector, $\theta = 270^\circ$
-4	4-arm strain	Be reflector, $\theta = 270^\circ$
ICD-1M	Magneto-strictive	7-4, Z=0
-3M		7-5, Z=0
-4M		7-6, Z=0
-9M		7-1, Z=0
-10M		Vessel, Z=0
MA-1		Vessel, Z=0, $\theta=0^\circ$
-2		Vessel, Z=0, $\theta=90^\circ$
-3		Vessel, Z=0, $\theta=180^\circ$
-4	Magneto-strictive	Vessel, Z=0, $\theta=270^\circ$

TABLE 2-6
SNAPTRAN-2 DOCUMENTARY PHOTOGRAPHY COVERAGE

Camera Type	Camera Station	Distance From Reactor (ft)	Frame Speed (ft/sec)
Bell and Howell	E	500	64
Bell and Howell	E	500	48
Bell and Howell	C	110	64
Arriflex	-	1 mi	48
Infra Red	C	110	1000
Photo Sonic	E	500	1000
↑	A	130	1000
↑	C	110	1000
↑	E	500	1000
↑	E	500	1000
↓	D	125	1000
Photo Sonic	D	125	1000
Waddell	A	130	2000
Waddell	D	125	2000
Waddell	C	110	2000
Hulcher	D	125	20
Hulcher	E	500	20

III. RESULTS

A. DRUM CALIBRATION

The drum calibration curves determined for the destruct test shimming configuration are shown in Figure 2-1. Although a detailed error analysis was not performed for this particular configuration, it is felt that the fractional uncertainty is no larger than that determined for the coupled drum calibration in Part 1 of this report.

B. PRELIMINARY TRANSIENTS

Several low-yield calibration transients were run on the SNAPTRAN-2 reactor prior to the destruct test. A detailed analysis of two of the larger-yield transients, 332 and 340, was performed. Certain results from this analysis were useful in the interpretation of the destruct test and will be presented here.

Transient 332 was initiated by a \$1.03 step reactivity insertion. Although the resulting average core temperature rise, as determined from the three 10% hydrided energy probes, was only 78°F, a reasonably accurate power normalization was determined. The normalized peak power and energy release were 7.6 Mw and 0.83 Mw-sec, respectively. A power calibration factor of 1.21 was determined for nuclear detector N-30. This is the factor by which the data from N-30 must be divided to bring the resulting energy release into agreement with that obtained from the energy probe data. The heat capacity model described in Part 1 was used in this calculation.

A prompt temperature coefficient of $-0.13 \text{ } \$/^{\circ}\text{F}$ was determined. This is in substantial agreement with similar tests in the SNAPTRAN-1 transient series. A comparison of the calculated radial energy deposition (or power) profile with the 10% hydrided energy probe results is shown in Figure 2-2.

Transient 340 was initiated by a \$1.05 step reactivity insertion. An average core temperature rise to a peak of 187°F was determined from the 10% hydrided probes. The temperature coefficient was $-0.13 \text{ } \$/^{\circ}\text{F}$. The normalized peak power and energy release to peak power were found to be 20.8 Mw and 0.54 Mw-sec, respectively. A comparison of the energy probe results to the calculated radial profile is shown in Figure 2-3.

Figure 2-1. Drum Calibration Curves for SNAPTRAN-2 Reactor

Figure 2-2. Relative Energy Deposition in 10% Probes on Test 332

Figure 2-3. Relative Energy Deposition in 10% Probes on Test 340

C. DESTRUCT TEST

1. Drum-Inserted Reactivity

The destruct test was initiated by the full insertion of the four reflector control drums. Examination of the drum position data showed that all drums were fully inserted before the time interval considered in these analyses. The fact that full insertion was achieved was verified by examination of slow-motion moving pictures taken of the test. The data showed that the stepper drums were completely stationary, while the impulse drums were oscillating slightly. The magnitude of the oscillation was only 3.5 degrees, however, which corresponds to less than 1¢ in reactivity worth. The total drum-inserted reactivity was determined to be \$5.05.

2. Power Data

A considerable variation in the time of peak power registered by various nuclear detectors was observed. This appeared to be due to time delays experienced by certain detectors that were sensitive to thermal neutrons and placed in moderating enclosures. Due to the extremely short period of the destruct test, thermalization time in the enclosure walls could cause a

significant time delay. The time of flight of thermal neutrons between the reactor and those detectors located at large distances could also be significant.

After careful consideration, it was decided that N-30 provided the most reliable peak power data and was, therefore, used in this analysis. As can be seen from Table 2-2, N-30 was a fast-neutron scintillation detector enclosed in 2 in. of lead. It would thus appear not to suffer from the time delay effects discussed above. Also, the time of peak power was observed to be in substantial agreement with that obtained from N-16 and N-26. Since N-26 was a gamma detector, this lends credence to the conclusion that N-30 had not experienced any time delays. N-30 was used instead of the other two detectors mentioned, because its data appeared to have less noise content.

The power trace shown in Figure 2-4 was reconstructed from the seven nuclear detector data channels given in scaled plots. The data are considered to be valid from 0.002 to 0.006 sec (on an arbitrary zero scale), giving nearly eight decades of usable power data. This reconstructed power trace was used in all the analyses to be discussed.

3. Period

The log plot of the reconstructed power trace in Figure 2-4 is seen to be essentially linear between 0.0020 and 0.0045 sec. The slope of this portion of the curve gives a period of 206 μ sec.

4. Average Temperature Data

Since most of the energy probes used in the destruct test had fuel compositions different from that of the actual core fuel, it was necessary to determine correction factors that would relate the temperature rise observed from these probes to the temperature rise that could be experienced by fuel of the regular composition at the same spatial positions. To accomplish this, the ratio of the energy deposition in the probes under consideration to that of probes having regular fuel composition was calculated. The fact that energy deposition is due to gamma and neutron interaction, as well as fission fragment slowing-down, was considered. The neutron and gamma flux spatial profiles were calculated as described in Part 1 of this report.

The following heat capacity was used for the 2.5% unhydrided probes:

$$C_{p-2.5\%} = (6.0 + 2.0 \times 10^{-3} T) \text{ cal/mole-}^{\circ}\text{K.}$$

The heat capacity used for the 5% unhydrided probes was:

$$C_{p-5\%} = (6.05 + 2.10 \times 10^{-3} t) \text{ cal/mole-}^{\circ}\text{K.}$$

The development of these two heat capacities is given in Reference 14. The fuel heat capacity described in Part 1 of this report was used for probes having the same composition as the core.

The relative energy deposition determined from all of the probes that gave usable data on the destruct test is shown in Figure 2-5. The energy-probe results have been compared with a calculated profile. This profile was calculated using the ANISN S_N transport code.⁽²⁴⁾ A 200- μ sec period was simulated in this calculation by the addition of a suitable amount of $1/v$ poison to the system.

This comparison shows data varying over a range as large as $\pm 25\%$. It was based on energy deposition near the end of the useful energy probe data. Similar comparisons made earlier in the transient showed much larger variations, especially in the 2.5% probe data. Clearly, one must exercise some discretion in deciding which data to use in the determination of average temperature:

1) As shown in Figure 2-5, the energy deposition indicated by most of the 2.5% probes lies below that indicated by the 10% probes. In addition, much of the 2.5% data showed high noise content or other erratic behavior which made it obviously unsatisfactory. This was partly due to unsatisfactory scaling of the data acquisition equipment, but this is not an adequate explanation for all of the eccentricity of the data.

2) Most of the 5% probes showed energy deposition somewhat higher than that predicted by the 10% probes. The 5% probes were all located in the reflector, however, and thus experienced flux of a different magnitude and of a different energy spectrum, as well as a different gamma and neutron background, than that experienced by the fuel. The calculation of the "power

Figure 2-4. Reconstructed Power Data for SNAPTRAN-2 Destruct Test

Figure 2-5. Relative Energy Deposition Determined for All Probes

profile in the reflector" (somewhat an artificial concept in itself) is subject to uncertainty. Thus, the temperature of the fuel can be deduced only quite indirectly by the data from these probes.

3) A rather complicated calculation was required to relate energy absorbed in the 2.5% and 5% probes to that of fuel material in the same location, as explained earlier. The accuracy of this calculation is not precisely known.

4) By contrast, each of the three 10% probes gave at least marginally acceptable data. These probes have the same composition as the fuel material, and their data can be converted to average fuel temperature by a relatively direct and accurate process.

5) All of the previous SNAPTRAN-1 and SNAPTRAN-2 preliminary transients were analyzed on the basis of the 10% probes. Little use was made of the 2.5% probes in SNAPTRAN-1 as only one probe of this type was used. Examination of the 2.5% probe data from Transients 332 and 340 might have removed some of the uncertainty in using these probes. Those data were made unusable by the amplifier scaling chosen, however.

From these observations, it is felt that the following conclusions are justified:

1) While the 5% probes may be useful as an indicator of flux in the reflector, their use to determine average core temperature is so indirect as to be undesirable.

2) The 2.5% probes are more directly applicable to the determination of average temperature, but they were troubled with erratic performance and lack of agreement with each other and with the 10% probes. The reasons are largely unknown. Little previous experience in using these probes is available to assist in interpreting these problems.

3) The data from the 10% probes seem to be most acceptable for the average temperature calculation.

Thus, most of the following discussion will concern the 10% probes. The data from all probes were analyzed, however, and these results will be presented for sake of comparison.

Figure 2-6. Energy From EP-153

Figure 2-7. Energy From EP-154

Figure 2-8. Energy From EP-155

Expanded-scale plots of the 10% probes are shown in Figures 2-6, 2-7, and 2-8. The data from EP-155 are seen to be quite smooth in the region above 2 Mw-sec. Some erratic behavior is observed in the data from both EP-153 and EP-154 at the upper ends of their traces, however.

Examination of Figures 2-2, 2-3, and 2-5 shows that the relative energy deposition of EP-153 was not consistent between the preliminary transients and the destruct test. The relationship between EP-154 and EP-155, on the other hand, is qualitatively similar in these three tests. The data from EP-153 are seen to fall well below both the calculated spatial profile and the spatial profile shown in preliminary Transients 332 and 340. This may indicate a possible time lag in the response of EP-153 during the destruct test. Comparison with Transients 332 and 340 was made because the possibility of a time lag can be excluded on these relatively long-period transients.

The relationship between energy from integrated power and energy from the energy probe data was also examined. It was this relationship that was considerably more linear for EP-154 or EP-155 than it was for EP-153.

These observations indicate that EP-153 should possibly be excluded from the analysis. Analyses based on all three probes, as well as with EP-153 excluded, have been performed for comparison.

In summary, three combinations of the available energy probe data are considered in this analysis. These combinations are: (1) all probes that gave usable data, (2) all the 10% hydrided probes, and (3) the two 10% hydrided probes, EP-154 and EP-155 only. Of these combinations, the third is considered the most reliable. The results of all three combinations were considered in determining the power normalization and temperature coefficient results to be discussed later.

5. Power and Energy Release Normalization

Power normalizations based on each of the combinations of energy probe results discussed in the previous section were determined. These have been summarized in Table 2-7. The same method described in Part 1 was used to determine these normalization factors. Results have been given in terms of both "released" and "absorbed" energy as was discussed in Part 1.

Figure 2-9. Temperature Coefficient
Results from SNAPTRAN-2 Based
On EP-154 and EP-155

TABLE 2-7
POWER, ENERGY RELEASE, AND TEMPERATURE COEFFICIENT
RESULTS FROM SNAPTRAN-2

	N-30	All Probes	All 10% Probes	EP-154 and EP-155
PCAL	1.13	1.376	1.218	1.120
$P_{\max}$ (Mw):				
Absorbed	63,980	52,590	59,420	64,620
Released	72,370	59,490	67,220	73,100
E_{∞} (Mw-sec):				
Absorbed	41.35	33.99	38.41	41.77
Released	46.78	38.45	43.45	47.25
α ($\phi/^{\circ}\text{F}$)	-0.207	-0.232	-0.223	-0.208

The total energy releases obtained by integrating the normalized power traces are also given in Table 2-7. The power data recorded after the power had decreased below one-half of its peak value were considered unreliable (see Part 1, Section II-A-3) and were replaced with data corresponding to the power shape predicted by the ENRECO reactor disassembly code (described in Part 3 of this report).

6. Temperature Coefficient

The temperature coefficients based on the various temperature traces are summarized in Table 2-7. The compensation curve based on EP-154 and EP-155 is shown in Figure 2-9. As can be seen from the figure, the coefficient is quite constant over the entire range of the calculation. This coefficient was found to be consistent with the results of the SNAPTRAN-1 test series when the appropriate temperature and period adjustments are made. This is discussed in Part 3 of this report.

IV. EVALUATION OF INSTRUMENTATION

A. NUCLEAR

Of the nuclear detector enclosure combinations that gave peak power, only three, N-16, N-26, and N-30, had adequate time response to accurately follow the destruct transient. The main reason for the time delays appeared to be the use of nuclear detectors that were sensitive to thermal neutrons being placed in moderating enclosures. The thermalization and diffusion times in the moderating enclosure walls introduced significant time lags into the data.

Another area that needs consideration in evaluating the nuclear detector performance is calibration. A brief discussion of the calibration procedure was given earlier. Because of the adjustment process used, the peak powers as indicated by the various instruments are made to agree with each other. Ultimately, however, the basis for the calibration was the calibration of scintillation detectors N-23 and N-43. The calibration of these detectors was subject to the same uncertainties as were discussed in connection with the SNAPTRAN-1 instrumentation.

The peak power and energy release based on the calibration of N-30 has been included in Table 2-7. As can be seen, the results are in good agreement with those obtained by normalizing the data to the 10% probes. Thus, the agreement between the energy probe and nuclear detector results was much better than was found during the SNAPTRAN-1 test series.

B. ENERGY PROBES

By using unhydrided energy probes with a uranium concentration lower than the core fuel, it was hoped that a temperature history slightly beyond the inception of core fuel disintegration could be obtained as was mentioned earlier. It was found, however, that the 2.5% probes did not give consistent results. Moreover, the disturbance caused when the core fuel started to disintegrate was sufficient to disrupt all the energy probe instrumentation immediately.

The performance of the 10% probes was generally satisfactory, as discussed earlier.

PART 3

CORRELATION BETWEEN EXPERIMENT AND ANALYSIS

NAA-SR-11850

3-1

I. ANALYSIS OF TEMPERATURE COEFFICIENT RESULTS

A. EXPERIMENTAL RESULTS

As discussed earlier and shown in Figure 1-30, the magnitude of the temperature coefficient observed in the SNAPTRAN-1 and -2 experiments showed a tendency to increase (i. e. , the algebraic value value becomes more negative) as the test series progressed. Figure 3-1 shows the coefficient plotted vs the accumulation of energy released before peak power by transients up to and including the transient plotted. The type of behavior seen in the figure could conceivably result from gradual instrument degradation or changes in fuel properties.

Another possible explanation is shown by the plots of coefficient vs inverse period and temperature (see Figures 3-2 and 3-3). These figures show a tendency for the magnitude of the coefficient to increase with temperature, and for the impulse test results (obtained at high inverse period) to be higher than the other results at comparable temperature. Also, an increase with inverse period is apparent, with the elevated-temperature step results (obtained at high temperature) higher than other results at comparable inverse period.

This suggests a correlation with both temperature and inverse period. Such a correlation to the data was done by a least-squares fitting technique. The data were correlated by

$$\alpha = (A + C \bar{T}) (1 + \tau \omega_0) \quad , \quad \dots (3-1)$$

where

α = observed temperature coefficient

$A = - 0.121 \text{ } \mu/\text{ }^\circ\text{F}$

$C = - 6.23 \times 10^{-5} \text{ } \mu/\text{ }^\circ\text{F}^2$

$\tau = 1.17 \times 10^{-4} \text{ sec}$

ω_0 = inverse period at beginning of temperature compensation

$\bar{T}$ = "mean temperature" = $1/2 (T_i + T_f)$

T_i = volume-average temperature before transient

T_f = volume-average temperature after transient.

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Figure 3-1. Variation of Temperature Coefficient with Accumulated Energy

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Figure 3-2. Variation of Temperature Coefficient With Inverse Period

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Figure 3-3. Variation of Temperature Coefficient With Temperature

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3-6

The standard deviation of the fit was

$$S = 0.015 \text{ } \epsilon / ^\circ\text{F}.$$

Figures 3-4 and 3-5 show the data of Figures 3-2 and 3-3, with the dependence on first $\bar{T}$ and then ω_0 removed according to the correlation. Shown also in Figures 3-4 and 3-5 are the results calculated by TSN; these are discussed and compared in a later section. Figure 3-6 shows the data of Figure 3-1, with both dependences removed. From this, it can be seen that the $\bar{T}$ and ω_0 correlations are sufficient to remove the apparent increase during the test series. A correlation with $\bar{T}$ and peak power was also investigated, but it was less successful than the $\bar{T}$ -vs- ω_0 correlation.

B. CALCULATIONAL METHOD

It was desired to compare these results with the results of the best current calculational techniques and data. The following will describe the calculational method used. Briefly, a multigroup transport code was used to obtain spectrum-weighted cross sections for the reactor. These data were then used in a 2-group, time-dependent, temperature-dependent transport-theory code to simulate the experimental test conditions.

Temperature changes (below 1900°F) on a SNAP core influence the reactivity through three classes of effects. First, the fuel rods increase in length and decrease in density due to axial fuel expansion. They also expand radially, but the fuel rods do not fill the cladding until about 925°F, and the claddings of adjacent rods do not touch until about 1260°F, assuming that cladding-fuel contact warms or stretches the cladding enough to just maintain intimate contact. Second, the thermal spectrum shifts toward higher neutron velocities, causing a change in the effective absorption, fission, and transport cross sections. Finally, the rates of resonance capture in zirconium and U^{238} are altered by Doppler broadening.

The fuel expansion is calculated as a function of temperature using a theoretical relationship fitted to experimental data. The thermal spectrum for each of several temperatures is calculated using the ANISN⁽²⁴⁾ neutron transport code with AI's 26-group thermalization library. Both fast and thermal neutron

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Figure 3-4. Variation of Corrected Temperature Coefficient With Inverse Period (Experiment vs Calculation)

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Figure 3-5. Variation of Corrected Temperature Coefficient With Mean Temperature

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Figure 3-6. Variation of Corrected Temperature Coefficient With Accumulated Energy

cross sections are then spectrum-weighted by ANISN, producing 2-group cross sections at each temperature.

The effects of axial temperature distribution are added by an importance-weighting method. A computer code is used to calculate the axial temperature distribution in a fuel rod for a given total energy deposition, to find (by interpolation) the cross sections appropriate to the temperature in each axial interval, to apply the effects of density changes, and then to importance-weight the cross section variation resulting from the spectrum and density changes. The result is a cross section, appropriate to a given temperature in a radial calculation, which includes the effects of axial temperature distribution and importance variation. (The Doppler changes can also be included in this calculation by appropriate modification of the absorption cross sections, although an indirect method was used in these calculations.)

These cross sections as a function of axially averaged temperature are then fit with polynomials and used as input to the TSN time-dependent transport theory (or spatially dependent kinetics) code.⁽³⁴⁾ TSN calculates the neutron flux as a function of time, position, velocity-group, and direction. The energy deposition and local temperature are determined from this and used to calculate modified cross sections and fuel lengths from the polynomial fits. The code then uses these modified data to continue the calculation. The code also provides volume-averaged temperature and "apparent reactivity," the reactivity which would be associated with the instantaneous reactor period according to space-independent kinetics, as was done in the analysis of the raw experimental data.

Comparing temperature change with apparent reactivity change gives a calculation of apparent temperature coefficient as it is observed in a SNAPTRAN transient. In the case of impulse tests, where the reflector thickness varied with time, the reactivity change due to this effect was subtracted from the gross change before it was compared with temperature. This is analogous to the method used in the analysis of the experimental data.

The desired initial reactivity was obtained either by changes in ν ("v-shimming"), reflector thickness ("R-shimming"), or by changing the transverse buckling to simulate unreflected fast leakage through a bare core surface

Figure 3-7. SNAPTRAN 10A/2 Temperature Coefficient - Static Transport Calculation (ANISN)

Figure 3-8. Comparison of T_f and $\bar{T}$ Plots - Static Transport Calculation (ANISN)

("B²-shimming"). Considering the actual geometry of the control drums, the experiment is probably most like a combination of B²- and R-shimming, perhaps more nearly like the R-shimming case. In any case, the three methods gave results agreeing within a 10% spread. R-shimming was used to calculate the data reported herein.

The polynomial coefficients used as input to TSN were also used to produce spatially dependent effective cross sections for use in the ANISN code. In this way, a static temperature coefficient which included both axial and radial weighting effects was calculated.

C. CALCULATED STATIC COEFFICIENTS

The static coefficient components due to fuel expansion, spectral changes, and Doppler broadening of resonances were evaluated separately using experimental data for the fuel expansion,⁽³⁵⁾ the spectrum-weighted cross sections as discussed, and calculated values of the resonance changes with temperature in the ANISN code. The results are shown in Figure 3-7.

The effect of spatial distribution of temperature changes on the density and spectral components were evaluated directly by using the axial importance-weighting method discussed, and then using these cross sections and densities in a radial ANISN calculation including the appropriate radial temperature distribution. The results of this are also shown in Figure 3-7. The Doppler coefficient was assumed to be increased by the same ratio as was the spectral component. The total weighted coefficient is also shown in Figure 3-7.

It should be noted that the graphs are of mean temperature coefficient, $\Delta\rho/\Delta T$, over the interval T_i to T_f , and not the differential temperature coefficient, $d\rho/dt$. The mean coefficient can be obtained more accurately from experimental data than can the differential; therefore, it is more suitable for comparison. Figure 3-8 shows the mean coefficient, $\bar{\alpha}$, plotted versus T_f as well as $\bar{T}$, the variable used in the correlation of experimental data. (The use of $\bar{T}$ allows transients which start at different initial temperatures to be compared on a suitable basis.)

Figure 3-9. Apparent Temperature Coefficient vs Inverse Periods – TSN Calculation (Doppler Effect Excluded)

Figure 3-10. Long-Period Temperature Coefficient – ANISN vs TSN Results

D. CALCULATED DYNAMIC COEFFICIENT

The axially weighted data were used with the TSN code to simulate reactor step transients over a range of inverse period from 400 to 5000 sec^{-1} . The mean coefficient to each of several temperatures was evaluated, giving a set of coefficients over a range of temperatures and inverse periods. The results are shown in Figure 3-9. As shown, the coefficient to a given temperature depends linearly on inverse period. The slope of the line increases with the magnitude of the dynamic coefficient (as it increases with temperature). The data are well fit by the formula,

$$\bar{\alpha}(\bar{T}, \omega_0) = \bar{\alpha}(\bar{T}, 0) (1 + \tau \omega_0) \quad ,$$

where τ = fitting constant = 1.32×10^{-4} sec.

The temperature dependence of the coefficient, $\alpha(\bar{T}, 0)$, is compared with the static result in Figure 3-10. Curves for three values of initial temperature, T_i , are given. The coefficient for transients with $T_i = 100$ shows a slightly increasing coefficient up to about $\bar{T} = 400^\circ\text{F}$, and then a gradual decrease. The curve for $T_i = 450^\circ\text{F}$ shows a similar phenomenon, but the start of this curve does not lie close to the result shown for the same $\bar{T}$ with $T_i = 100^\circ\text{F}$. Apparently, the coefficient applicable to a large ΔT centering around 600°F is not the same as that applicable to a small ΔT centered there.

A curve was passed through the low- ΔT results obtained for each of the three T_i 's. This is shown in the figure. The low- ΔT curve agrees very well with the ANISN static calculation, as it should.

The reason for some of the discrepancy may be due to the basic nonlinearity of the $\bar{\alpha}$ -vs- T_f curve (see Figure 3-7), but this, alone, is not the complete story. The curve in Figure 3-10 for $T_i = 100^\circ\text{F}$ shows a smaller coefficient at $\bar{T} = 600^\circ\text{F}$ than at 200°F . This result could not come from the type of nonlinearity shown in Figure 3-7.

Another possibility is that this effect comes from the variation in temperature distribution and flux shape. The ANISN calculation assumes that the energy deposition has the same spatial distribution as the static power distribution.

This is not quite correct, however. The temperature change affects the power shape, which, in turn, perturbs the temperature shape. This could have some effect at large ΔT . Examination of the data shows the variation in energy deposition profile to be negligible, however.

Probably the most important effect can be seen by remembering that the effective temperature coefficient increases with inverse period. Thus, correcting a transient for its "period effect" based on a correlation in $\bar{T}$ and ω_0 insures that the first portion of the compensation will be treated correctly. For a transient covering a large ΔT , however, a wide range of ω 's are encountered. Toward the end of the transient, the inverse period may be zero or negative. Thus, the magnitude of the observed should decline with increasing ΔT as observed. Using the calculated value of τ , we estimate a variation of 12% in $\bar{\alpha}$ due to this effect for an ω_0 of 5000 sec^{-1} . This is just about the magnitude of the observed variation.

Correcting for $\bar{\omega}$ would only make the TSN curve match the ANISN curve at its midpoint, not much better. Actually, the problem can only be solved by either confining the comparison to the low- ΔT data, or by correcting the dynamic calculation for the instantaneous ω . The latter is beyond the scope of this study.

It is of some interest to fit a curve to the low- ΔT data for $\bar{T} \leq 400^\circ\text{F}$, the range which included most of the transients:

$$\bar{\alpha}(\bar{T}, 0) = \bar{\alpha}(0, 0) + C \bar{T} \quad ,$$

where

$$\begin{aligned} \bar{\alpha}(0, 0) &= -0.134 \text{ } \epsilon / ^\circ\text{F} \\ C &= -8.75 \times 10^{-5} \text{ } \epsilon / ^\circ\text{F}^2. \end{aligned}$$

All the TSN calculations have been adjusted slightly to place them on a λ_r basis, as will be discussed below.

E. IMPULSE TEST STUDIES AND THE GENERATION TIME PROBLEM

Calculations representing both single-drum and double-drum impulses were also made with TSN. The objective was to establish whether coefficients evaluated from the impulse tests should be included in the same correlation with step results, or whether they were inherently different from step test data.

It should be noted that the experiments and calculations involving impulse-type drum rotation are more accurately characterized by the inverse period at the beginning of the temperature rise, as was done in the present correlation, than by the highest inverse period reached.

The first conclusion from the study was the observation that the impulse tests are more sensitive to the generation time used in their analysis than a comparable step test. This can be seen from the fact that the total reactivity change during the temperature rise in an impulse test may be 2 to 3 times as large as that in a step test of comparable ΔT . This is due to the fact that the drum movement is withdrawing reactivity at the same time that the temperature rise is compensating it. Thus, an error in generation time (for example, a +3% error) would make a +3% error in the coefficient determined in a step test, but would make a +6 to +9% error in a comparable impulse measurement. Thus, the impulse coefficient would appear to disagree with the step result by 3 to 6% in this case.

Further, it was found that the generation time calculated in the usual way, by balancing changes in ν with a $1/\nu$ poison representing an inverse period ω , did not accurately represent changes caused by reflector drum movement. This is illustrated in Figure 3-11, an inhour curve calculated by plotting reactivity changes brought about by changes in ν vs the "inverse-period poison" required to counteract the changes. Curves for two reflector thicknesses are shown; their asymptotic slopes at small ω are different by 11%. Clearly, the slope, and thus the generation time, Λ , depends on reflector thickness. Thicker reflectors give larger Λ 's, as would be expected.

To go a step further, it can be seen that for a single fixed ν , there is one point on each curve which correlates the reactivity change caused by a change in reflector thickness (which is used to represent reflector control drum movements) with the inverse period required to counteract it. These points were found for a series of reflector thicknesses, resulting in Figure 3-12. This is a

Figure 3-11. Inhour Curves for Two Reflector Thicknesses (R)

Figure 3-12. Inhour Curve for Variable Reflector Thickness

new "inhour curve" appropriate to changes in a reactor caused by reflector thickness changes. It is quite linear, and has a slope 7% less than that of the full-reflector curve (2.125 in.) of Figure 3-11. A new generation-time-like constant, " Λ_r ," can be obtained from the slope of this curve, just as Λ is obtained from Figure 3-11.

Which is the real Λ ? The value of Λ corresponding to a given reflector thickness should be used to calculate the reactivity changes which correspond to period changes brought about by temperature changes at that (constant) reflector thickness. The value of Λ_r should be used to calculate the reactivity changes corresponding to period changes caused by control drum motion.

The experiments were analyzed using a value of Λ obtained from a drum-movement curve such as Figure 3-12, and were therefore analyzed using data corresponding to Λ_r . To obtain results comparable with static calculations, however, one must use Λ_c , corresponding to the reflector thickness required to make the unperturbed reactor critical, in all calculations. Then the effect of the changing Λ will be to cause an apparent variation in coefficient with inverse period, aside from any variation already present. Assuming this to be the "correct" method, the calculations indicate that the experimental results could be put on this basis by reducing all the coefficient measurements by 2.6%. It can also be concluded that the effect of the variation in lifetime would be to cause the coefficient at high periods to be underestimated by the use of the single Λ , since a larger Λ would be appropriate to the analysis. This would give a period effect in itself. In the absence of other effects,

$$\alpha(\bar{T}, \omega_0) = \alpha(\bar{T}, 0) (1 + \tau_{\Lambda} \omega_0) ,$$

where $\tau_{\Lambda} = -21 \mu\text{sec.}$

This effect is in the direction opposite to the ω_0 dependence actually observed, and is apparently more than counteracted by other effects.

It should not be concluded, however, that it was a mistake to use Λ_r in the SNAPTRAN analysis. This is the only basis on which impulse and step tests give comparable results. In addition, Λ_c would have been difficult to obtain, whereas Λ_r was obtained quite directly. It is probably most practical, then,

to proceed as was done, keeping in mind that a small overestimate of temperature coefficient is being made, in addition to whatever other errors are involved in this determination. Impulse experiments should be expected to give coefficients as much as 5% higher than comparable step experiments because of this effect, aside from disagreement due to drum calibration and Λ determination errors.

The " Λ dilemma" situation is related to the conceptual arguments of Wasserman,⁽³⁶⁾ who pointed out that the reflector thermalization delays (seen as curvature in Figure 3-11) can affect the results of changes in the core, but that experiments in which the reactivity is adjusted by moving the reflector drums can give no information as to this effect (represented by lack of curvature in Figure 3-12). It may be noted, however, that the 2-node model used by Wasserman failed to predict the large magnitude of the period dependence observed and calculated here.

Finally, the TSN results were adjusted to put them on the same basis as the experiments. With this done, the data indicated that impulse calculations and step calculations give the same coefficient results if analyzed on the basis of Λ_r . The experimental impulses would then presumably give results comparable to the step results, in the absence of drum-calibration errors and given a perfectly accurate determination of Λ_r .

F. COMPARISON OF CALCULATION AND EXPERIMENT

Figures 3-4 and 3-5 show the experimental correlation data compared with the results calculated by TSN. TSN appears to calculate the period dependence quite well. It tends to overestimate the coefficient below 450°F and to underestimate the elevated-temperature step results. The calculated results fall within 16% of the experimental correlation over the entire temperature range. The dependence of the coefficient on temperature, however, as indicated by the correlation, is found in the data only below $\bar{T} = 400^\circ\text{F}$.

Comparing the parameters of Equation 3-1, as determined from calculation and experiment,

$$\alpha(\bar{T}, \omega_0) = (A + C \bar{T}) (1 + \tau \omega_0) \quad ,$$

where:

	<u>SNAPTRAN Experiment</u>	<u>TSN Calculation</u>	<u>Difference (%)</u>
A =	-0.121 ¢/°F	-0.134 ¢/°F	11
τ =	117 μsec	132 μsec	13
C =	$-6.23 \times 10^{-5} \text{ ¢/°F}^2$	$-8.75 \times 10^{-5} \text{ ¢/°F}^2$	40

Considering the uncertainty in both determinations, the values of A and τ probably do not differ significantly. The calculated value of C is too large, which is to be expected, since many of the experiments involved fairly large ΔT 's, e. g., $T_i \approx 70^\circ\text{F}$ and $T_f = 600^\circ\text{F}$. Thus, they are affected by the "high- ΔT " effect shown as the difference between the two TSN curves in Figure 3-5.

It is also of interest to see what the theoretical prediction would have given for the SNAPTRAN-2 experiment. It is shown in Figure 3-5 that the analysis of SNAPTRAN-2 data covered a range of temperatures centered about 520°F . The $T_i = 100^\circ\text{F}$ curve must be used, since this is about equal to the initial temperature of the transient. This gives -0.153 ¢/°F as the " $\omega = 0$ " coefficient, $\bar{\alpha}(\bar{T}, 0)$. Then,

$$\bar{\alpha}(\bar{T}, \omega_0) = \bar{\alpha}(\bar{T}, 0) (1 + \tau \omega_0) = -0.25 \text{ ¢/°F} \quad .$$

This is in reasonable agreement with the measured -0.21 ¢/°F .

This difference of less than 20% may be the result of nonequilibrium thermodynamic effects, inertial effects delaying axial fuel expansion somewhat, or of inherent error in the measurement and calculation.

G. SUMMARY

The TSN calculations predict the magnitude of the SNAPTRAN temperature coefficient correlation within 16% for the range of mean temperatures studied ($200^\circ\text{F} < \bar{T} < 1000^\circ\text{F}$). They predict the period dependence within 13% for the range of periods studied ($0 < \omega_0 < 5000 \text{ sec}^{-1}$). The overall calculations show less variation with temperature than the correlation, although they show somewhat more variation for $\bar{T} < 400^\circ\text{F}$.

It was also shown that the coefficient for transients involving large ΔT are not properly predicted by a single correlation in ω_0 and $\bar{T}$. The spatially dependent effect, in this case, acts to give a coefficient smaller than the coefficient indicated by a series of small ΔT experiments at different initial temperatures.

As a result of this phenomenon, the coefficient for transients starting at ambient temperature appear nearly independent of temperature up to $\bar{T} \approx 500^\circ\text{F}$ or $T_f \approx 900^\circ\text{F}$. This probably explains why the temperature variation in coefficient predicted by the correlation of all the experiments was absent in the individual experiment results.

The study also showed that the measured results are probably high by 2.6% because of the "generation-time discrepancy" which arises as a result of changing drum positions to adjust reactivity. The TSN results have been adjusted to place them on the same basis as the experimental data.

Impulse tests were found to differ from corresponding step tests by about 5% due to the 2.6% discrepancy. They were also found to be more sensitive to errors in Λ_r determination and to drum calibration errors than are corresponding step tests.

It may be interesting to note that there is nothing in the calculational method which would have been done differently had the calculation been performed without knowledge of the experiments. The chief experiment-calculation interaction was that the dilemma posed by the results shown in Figure 3-1 spurred interest and support for the comparatively difficult dynamic coefficient calculational effort in a way that no theoretical argument could match.

II. CORE DISASSEMBLY

A. THEORETICAL DESCRIPTION OF REACTOR BEHAVIOR DURING DISASSEMBLY

The ENRECO code (ENergy RElease COde) is designed to compute the energy released during the disintegration of a spherical reactor system during a transient initiated by a step insertion of reactivity. The theoretical model was developed to assist in predicting the energy release, power profile, temperature profile, and other measures of reactor behavior during SNAPTRAN destructive transients. The model has been described elsewhere,⁽³⁷⁾ and is reviewed in this section.

The reactor system is assumed to be a uniform homogeneous sphere at the start of the transient. A step insertion of reactivity initiates the transient, and the normal kinetics equations with fuel temperature feedback are assumed valid until the maximum fuel temperature reaches the fuel disintegration temperature. In SNAP reactors, which have approximately a cosine-shaped power distribution, the central fuel will reach this temperature first. The boundary between disintegrated fuel and the fuel not yet disintegrated defines a volume expanding away from the fuel center. All energy in excess of that required for heating to the disintegration temperature is assumed to be utilized in dissociating hydrogen from the fuel material. The large pressure in the inner volume causes an acceleration force and expands the surrounding core (see Figure 3-13). The core expansion results in a reactivity decrease, and the power transient terminates as the core is disassembled and dispersed. No hydrogen is allowed to escape the fuel region.

The code is written in two sections dealing with the reactor behavior before and after fuel disintegration occurs. These sections are described in the following paragraphs.

1. Before Fuel Disintegration

The ENRECO code solves the transient equations describing the reactor behavior before fuel disintegration begins by increasing the mean fuel temperature by fixed increments until the maximum fuel temperature exceeds the fuel disintegration temperature. For each temperature step, the following quantities are computed:

3-31-67 UNC

7700-25145

Figure 3-13. Core Expansion

1) Energy release (Mw-sec):

$$E_i = E_{i-1} + \int_{T_{i-1}}^{T_i} C_p(T) dT \quad , \quad \dots (3-1)$$

where T = temperature ($^{\circ}\text{F}$), $T_i = T_{i-1} + \Delta T$, and C_p = specific heat (Mw-sec/ $^{\circ}\text{F}$).

2) Reactivity ($\$$):

$$\rho_i = \rho_0 + \alpha_T (T_i - T_0) \quad , \quad \dots (3-2)$$

where ρ_0 = initial reactivity step ($\$$), α_T = temperature coefficient ($\$/^{\circ}\text{F}$), and T_0 = initial temperature ($^{\circ}\text{F}$).

3) Reciprocal period (sec^{-1}):

$$\omega_i = \frac{\beta}{\ell} \rho_i \quad \dots (3-3)$$

where β = effective delay neutron fraction and ℓ = prompt neutron lifetime (sec).

4) Time step (sec):

$$\Delta t_i = \frac{1}{\omega_i} \ln \left[1 + \frac{\omega_i (E_i - E_{i-1})}{P_{i-1}} \right] \quad \dots (3-4)$$

5) Power (Mw):

$$P_i = P_{i-1} e^{\omega_i \Delta t_i} \quad \dots (3-5)$$

6) Maximum fuel temperature ($^{\circ}\text{F}$):

a) With quadratic specific heat $\left[C_p = C_{p0} (1 + AT + BT^2) \right]$

$$E_{\max} = \frac{\hat{\Phi}}{\bar{\Phi}} E_i = C_{p0} \left(T_{\max} + \frac{AT_{\max}^2}{2} + \frac{BT_{\max}^3}{3} \right) \quad \dots (3-6)$$

is solved for $T_{\max}$, where

$$\frac{\hat{\Phi}}{\bar{\Phi}} = \text{ratio of peak-to-average fission density, or}$$

b) Using an Einstein oscillator specific heat

$$\left[C_p = A + BT + C e^{\theta_E/T} \left(\frac{\theta_E/T}{e^{\theta_E/T} - 1} \right)^2 \right],$$

$$E_{\max} = \frac{\hat{\Phi}}{\bar{\Phi}} E_i = AT_{\max} + \frac{BT_{\max}^2}{2} + \frac{C\theta_E}{e^{\theta_E/T_{\max}} - 1} \quad \dots (3-7)$$

is solved for $T_{\max}$, using the Newton-Raphson method. θ_E = the Einstein temperature.

2. After Start of Fuel Disintegration

Fuel disintegration is assumed to start at the center of the reactor core when $T_{\max}$, the maximum fuel temperature (the center temperature) computed from procedure (6) above, exceeds an input disintegration temperature ($T_{\exp}$). The portion of the code that performs the computations after the start of fuel disintegration proceeds, using time steps of an input magnitude. At each time point, the following calculations are performed:

- 1) Energy density at start of disintegration ($T_{\max} = T_{\exp}$):

$$E_{\text{dis}} = \int_{T_0}^{T_{\exp}} C_p(t) dT \quad \dots (3-8)$$

- 2) Total integrated core energy at start of disintegration:

$$E_T = E_{\text{dis}} / (\hat{\Phi} / \bar{\Phi}) \quad \dots (3-9)$$

- 3) Energy release (Mw-sec):

$$E_i = E_{i-1} + \int_0^{\Delta t} P_{i-1} e^{\omega_i t} dt = E_{i-1} + \frac{P_{i-1}}{\omega_i} \left(e^{\omega_i \Delta t} - 1 \right) \quad \dots (3-10)$$

or

$$E_i = E_{i-1} + \frac{P_i - P_{i-1}}{\omega_i} , \quad \dots (3-11)$$

where P = power (Mw) and ω_i = inverse period (sec^{-1}).

4) Radius of boundary between dissociated region and region where no dissociation has begun:

$$R_{B_i} = \frac{2 R_{\text{ext}}}{\pi} \cos^{-1} \left(\frac{E_T}{E_i} \right) , \quad \dots (3-12)$$

where R_{ext} = extrapolated radius of core (in.)

5) Energy used in hydrogen dissociation (Mw):

$$E_{H_i} = E_{RB} - \frac{4}{3} \pi R_{B_i}^3 \frac{D_f}{1+V_i} E_{\text{dis}} , \quad \dots (3-13)$$

where D_f = fuel density and V_i = core void fraction, and E_{RB} = total energy generated within radius R_{B_i}

$$= E_i \frac{\hat{\Phi}}{\Phi} P_i \int_0^{R_{B_i}} 4 \pi r^2 \cos \frac{2 \pi r}{2 R_{\text{ext}}} dr \quad \dots (3-14)$$

6) Fraction of hydrogen dissociated and moles of hydrogen produced:

$$H_f = E_{H_i} / E_{H_{\text{tot}}} \quad \dots (3-15)$$

$$N = H_f N_t , \quad \dots (3-16)$$

where $E_{H_{\text{tot}}}$ = energy required to dissociate all the hydrogen in the core
(a total of N_T moles)

7) Gas temperature (°F): The gas temperature is computed for each time step by averaging the temperature of the hydrogen gas released on all previous time steps, which is assumed to cool isentropically as it expands, and the temperature of the gas dissociated on the i^{th} time step (assumed to be at the dissociation temperature T_{exp}). Thus,

$$T_{\text{gas}_i} = \frac{N_m T_m + N_{\text{dis}} T_{\text{exp}}}{N_m + N_{\text{dis}}} , \quad \dots (3-17)$$

where:

N_m = number of moles of hydrogen released on all previous time steps

N_{dis} = number of moles of hydrogen dissociated on the i^{th} time step

T_m = temperature of hydrogen released previous to this time step

$$\left(T_{m_i} V_{H_i}^{\gamma-1} = T_{m_{i-1}} V_{H_{i-1}}^{\gamma-1} \right), \text{ where } V_H = \text{volume of gas,}$$

and γ = ratio of heat capacity at constant pressure to heat capacity at constant volume for H_2 gas ... (3-18)

8) Gas pressure (psi):

$$P_i = \frac{(N_m + N_{\text{dis}}) R T_{\text{gas}_i}}{V_H} , \quad \dots (3-19)$$

where V_H = total volume of H_2 gas and is given by

$$V_H = \frac{4}{3} \pi \left[R_{B_i}^3 (v_c + v_f) + (r_c + r_a)^3 - r_c^3 \right] , \quad \dots (3-20)$$

where:

v_c = core void fraction (macroscopic voids in core)

v_f = fuel void fraction (microscopic voids in fuel)

$$r_c = \left(\frac{3 \text{ vol}_c}{4 \pi} \right)^{1/3} \quad \dots (3-21)$$

$$\text{vol}_c = \frac{w_f}{d_f} (1 + v_c) \quad \dots (3-22)$$

w_f = fuel weight

d_f = fuel density

and where the equation of motion is given by the following sequence of calculations:

$$\begin{aligned} m_{2_i} &= \text{remaining (undisintegrated) acceleration mass} \\ &= m_a - \frac{4}{3} \pi \frac{d_f}{1+v_c} R_{B_i}^3, \end{aligned} \quad \dots (3-23)$$

where m_a = acceleration mass of entire reactor,

and $\dot{m}_1 = \frac{m_{2_i} - m_{2_{i-1}}}{\Delta t}$, with $m_{2_0} = m_a$, and, by setting the force arising from the pressure equal to the time rate of change of momentum, $\dots (3-24)$

$$\begin{aligned} \Delta \left(\Delta r_{a_i} \right) &= \left(4 \pi P_i R_{B_i}^2 - \Delta r_{a_{i-1}} \frac{\dot{m}_1}{g} \right) \frac{g}{m_{2_i}} \\ \Delta r_{a_i} &= \Delta \left(\Delta r_{a_i} \right) \Delta t + \Delta r_{a_{i-1}}, \text{ with } \Delta r_{a_0} = 0, \\ r_{a_i} &= \Delta r_{a_i} \Delta t + r_{a_{i-1}}, \text{ with } r_{a_0} = 0. \end{aligned} \quad \dots (3-25)$$

9) Reactivity (β):

$$\rho_i = \rho_0 + \alpha_T (T_i - T_0) + \alpha_R r_{a_i}, \quad \dots (3-26)$$

TABLE 3-1
REACTOR PARAMETERS FOR SNAPTRAN-2 and -3

Parameters		SNAPTRAN-2	SNAPTRAN-3
P_0	Initial power (Mw)	1×10^{-8}	1×10^{-8}
T_0	Initial temperature (°F)	70.	70.
ρ_0	Reactivity insertion (\$)	4.93	3.60
α_T	Temperature coefficient (\$/°F)	-2.5×10^{-3}	-2.18×10^{-3}
T_D	Fuel disintegration temperature (°F)	1932.	1930
β_{eff}	Effective delayed neutron fraction	8×10^{-3}	8×10^{-3}
l	Prompt neutron lifetime (sec)	6.5×10^{-6}	1.4×10^{-5}
$\hat{\Phi}/\bar{\Phi}$	Peak-to-average fission density	1.86	1.61
P_D	Initial disintegration pressure (psi)	1.5×10^4	1.5×10^4
W_f	Fuel weight (lb)	112.	112.
m_{acc}	Acceleration mass (lb)	140.	140.
d_f	Fuel density (lb/in. ³)	0.202	0.202
	gm-moles of H ₂ in Fuel	460.	460.
	H ₂ dissociation energy (Mw-sec required to dissociate 460 gm-moles from fuel)	73.5	73.5
α_r	Expansion coefficient (\$/in.)	-10.	-10.
v_c	Core void fraction	0.1	0
v_f	Fuel void fraction	0.01	0.01
Specific Heat: (for both reactors)			
$C_p(T) = 0.938 \times 10^{-4} (1 + 3.36 \times 10^{-4} T + 3.43 \times 10^{-7} T^2) \text{ Mw-sec/lb}^\circ\text{F}$			

NOTE: The quadratic heat capacity is somewhat different from the heat capacity used in the analysis of the SNAPTRAN-1 and -2 experiments. It is about 20% higher at 32°F, 15% lower at 752°F, and 10% higher at 1472°F. Its average value over the temperature range of the calculation is probably slightly lower than the one used in the analysis.

where α_R = expansion coefficient of reactivity (\$/in.). Equations 3-8 to 3-26 are solved simultaneously with Equations 3-3 to 3-5.

B. COMPARISON OF THEORY AND EXPERIMENT

The ENRECO code was employed to calculate the power, energy, fuel disintegration, hydrogen pressure, and core disassembly for the conditions of the SNAPTRAN-2 destructive test.

The input parameters for the calculation were as listed in the first column of Table 3-1. For comparison purposes, the values previously used for the SNAPTRAN-3 analysis^(10, 38) are listed in the second column of Table 3-1.

A comparison of the calculated power trace with the traces from three of the nuclear detectors (N-16, N-26, and N-30) is shown in Figure 3-14.

In the experiment, high-speed photographs were taken of lights shining through the core interstices. Both framing and streak cameras were used. The fields of view of the cameras were as shown in Figure 3-15.

Fuel disintegration began at the center of the reactor (in the highest power region) and proceeded outwards. Shortly after disintegration began, adjacent interstices were closed as the light path was interrupted. The progression of fuel disintegration was as shown in Figure 3-16, from Streak Camera No. 4. The film was traveling at a known velocity in a direction perpendicular to the viewing slit. The time of fuel disintegration at each interstice location was determined from the streak photographs. The data points are plotted in Figure 3-17, along with the theoretical curve (from the ENRECO calculation).

Displacement measurements (axial fuel rod and radial reflector displacement) were obtained using clusters of time-of-arrival contact pins. The data points are plotted in Figure 3-18, along with the theoretical curve (from the ENRECO calculation).

The theoretical displacement is calculated from mechanical expansion only; the difference between the axial expansion data points and the curve, for times earlier than 5.7 msec and displacements of less than 0.1 in., is due to axial thermal expansion prior to core disassembly. The amount of this thermal expansion is about as expected.

Figure 3-14. Theoretical and Experimental Power

Note that during core disassembly, the radial and axial displacements are approximately equal (compare the rod data points with reflector data points in Figure 3-18). This implies that there would be no significant advantage to be gained by using a two-dimensional geometry in place of the one-dimensional equivalent spherical geometry used in the ENRECO model.

Fuel disintegration began at about the time of peak power. The reactivity effect of disassembly became the dominant factor in such a way as to effectively determine the shape of the power curve and the energy release.

The comparison between theory and experiment is satisfactory in all respects.

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Figure 3-15. Camera Fields of View of SNAPTRAN-2 Core

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4S-2 7611-01194A

Figure 3-16. Progression of Fuel Disintegration Recorded by Streak Camera No. 4

7611-01205

Figure 3-17. Time of Fuel Disintegration Determined From Streak Photographs with Theoretical Curve From ENRECO Calculation

NAA-SR-11850

3-34

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7611-01208

Figure 3-18. Contact Pin Measurements of Fuel Rod and Beryllium Reflector Displacement

APPENDIX A

REFERENCES

2

2

APPENDIX A

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APPENDIX B

ENRECO CODE DESCRIPTION

APPENDIX B

ENRECO CODE DESCRIPTION

1. General Information

The ENRECO code is written in FORTRAN-II for the IBM 709-7090-7094 computer series. Two decks have been placed on file at Atomics International. Deck 9W-145 contains the SNAPTRAN-3 quadratic specific heat model, and Deck 9W-146 contains the Einstein oscillator model. The Einstein oscillator data currently on file is for $\text{ZrH}_{1.81}$ using data obtained by AI.

The input data consist of the following:

- 1) CRT graphical output titles
- 2) Einstein oscillator coefficients (Deck 9W-145 only)
- 3) Title card
- 4) Numerical input in FORED format.

The deck setup is slightly different for the two decks and is shown below:

	<u>Deck 9W-145</u> (Quadratic C_p)	<u>Deck 9W-146</u> (Einstein Oscillator C_p)
On file	1) Binary program deck	1) Binary program deck
On file	2) *Data card	2) *Data card
On file	3) CRT titles	3) CRT titles
On file		4) Einstein oscillator parameters (one card, 4F12.0 format, contains A, B, C, θ_e)
Sent in by user,	4) Title card (12A6)	5) Title card (12A6)
for each case	5) FORED data	6) FORED data

2. Data Input

All numerical data (except the Einstein oscillator parameters) are input in the FORED format. A modified version of FORED, which allows data addresses to be entered right — adjusted in columns 2-6 or 7-12, or both, is used. (When both fields are used, the address is assumed to be the sum of the addresses in the

two fields.) This version was written by W. A. Rhoades. The data input for each case is described below.

Card 1, Title card (12A6):

Any BCD information. Used as title for graphical and printed output.

Cards 2 to N, FORED input:

Location	FORTTRAN Symbol	Description	Units
1	PØ	Initial power	Mw
2	TØ	Initial temperature	°F
3	DØLØ	Initial reactivity insertion	\$
4	ALPHAT	Temperature coefficient (magnitude, - sign assumed)	\$/°F
5	ALPHAR	Expansion coefficient (magnitude, - sign assumed)	\$/in.
6	T _{EXP}	Fuel disintegration temperature	°F
7	Beta	Effective delay neutron fraction	--
8	BL	Prompt neutron lifetime	sec
9	PA	Peak to average fission density	--
10	PRESØ	Initial pressure in disintegrated fuel	psi
11	Weight	Fuel weight	lb
12	AMASS	Acceleration mass	lb
13	CPØ	$\left\{ \begin{array}{l} \text{Coefficients for } C_p \text{ fit} \\ C_p(T) = C_{po}(1 + AAxT + BBxT^2) \\ \text{Enter only for deck 9W-145} \end{array} \right\}$	Mw-sec/ lb°F
14	AA		(°F) ⁻¹
15	BB		(°F) ⁻²
16	SFREQ1	Every SFREQ1th point will be edited before fuel disintegration occurs	--
17	SFREQ2	Every SFREQ2th time point will be edited after fuel disintegration begins	--
18	DTEMP	Temperature increment for calcula- tions before disintegration begins	°F
19	DTIME	Time increment for calculations after disintegration starts	sec
20	DENS	Fuel density	lb/in ³
21	TMG	Total gm-moles of H ₂ gas available in fuel	gm-moles

<u>Location</u>	<u>FORTTRAN Symbol</u>	<u>Description</u>	<u>Units</u>
22	ETMG	Energy required to dissociate TMG gm-moles of H ₂	Mw
23	VF1	Core void fraction	--
24	VF2	Fuel void fraction	--

3. Data Output

The output from the ENRECO code consists of the following data:

- 1) Input data edit
- 2) Einstein oscillator parameters (deck 9W-146 only)
- 3) Power (Mw), fuel temperature (°F), gas temperature (°F), maximum fuel temperature before disintegration begins (°F), energy (Mw-sec), reactivity (\$), as a function of time
- 4) Inverse period (sec⁻¹), inner radius (in.), outer radius change (in.), acceleration mass (lb), and pressure (psi), as a function of time.

In addition, the extrapolated system boundary is printed as REXT. A skipped space in the output list indicates the start of fuel disintegration (see sample cases), for deck 9W-146; no space is skipped when deck 9W-145 is used, but the start of disintegration can be found by locating the time point for which Gas Temp = T_{dis} (see sample cases).

Before the start of fuel disintegration, the column labeled Gas Temp. contains maximum fuel temperature.

4. Sample Case

The input and output for a sample case are shown on the following pages. The sample case is the SNAPTRAN-2 test, run on deck 9W-145, using parameters shown in Table 3-1 in Part 3 of this report.

The case shown consumed 8.2 sec of load time and 18.8 sec of execution time.

FORTRAN FIXED 10 DIGIT DECIMAL DATA

DECK NO. DATA PROGRAMMER W. B. GREEN DATE 5/10/66 PAGE 1 of 2 JOB NO.

NUMBER	IDENTIFICATION	DESCRIPTION DO NOT KEY PUNCH
1 S N A P T R A N - 2 *		SAMPLE CASE - DECK 9W-145
13 Q U A D R A T I C C		
25 S U B P *		TITLE CARD
37		
49	73 80	
61	1	
1	1	This card contains data for Locations 1, 2, 3, 4, 5
13 1 . - 8		$P_0 = 10^{-8}$ Mw
25 7 . 0		$T_0 = 70^\circ\text{F}$
37 4 . 9 3		$\rho_0 = \$4.93$
49 2 . 5 - 3	73 80	$\alpha_T = - \$0.0025/^\circ\text{F}$
61 1 0	2	$\alpha_R = - \$10./\text{in.}$
1	2 4	This card contains data for Locations 4+2 = 6, 7, 8, 9, 10
13 1 9 3 2 .		$T_{\text{exp}} = 1932^\circ\text{F}$
25 8 . - 3		$\beta = 0.008$
37 6 . 5 - 6		$l = 6.5 \mu\text{sec}$
49 1 . 8 6	73 80	Peak/avg fission density = 1.86
61 1 . 5 + 4	3	Initial pressure = 1.5×10^4 psi
1	1 1	
13 1 1 2		Fuel wt = 112 lb
25 1 4 0		Acceleration mall = 140 lb
37 9 . 3 8 - 5		$C_p(T) = 9.38 \times 10^{-5} (1 + 3.36 \times 10^{-4} T + 3.43 \times 10^{-7} T^2)$
49 3 . 3 6 - 4	73 80	Mw-sec/lb°F
61 3 . 4 3 - 7	4	

FORTRAN FIXED 10 DIGIT DECIMAL DATA

DECK NO DATA PROGRAMMER W. B. GREEN DATE 5/10/66 PAGE 2 of 2 JOB NO. _____

	NUMBER	IDENTIFICATION	DESCRIPTION DO NOT KEY PUNCH
1	1 6		
13	1 0 .		Every 10 th point will be printed before disintegration starts
25	1 0 0 .		Every 100 th point will be printed after disintegration starts
37	5 .		$\Delta T = 5^{\circ}F$ before disintegration starts
49	1 . - 7	73 80	$\Delta t = 0.1 \mu\text{sec}$ after disintegration starts
61	. 2 0 2	5	Fuel density = 0.202 lb/in.^3
1	1 2 1		1 in Col. 1 $\longrightarrow$ LAST CARD OF CASE
13	4 6 0 .		460 gm-moles of H_2 available
25	7 3 . 5		73.5 Mw required to dissociate 460 gm-moles of H_2
37	. 1		Core void fraction = 0.1
49	. 0 1	73 80	Fuel void fraction = 0.01
61		6	
1			
13			
25			
37			
49		73 80	
61			
1			
13			
25			
37			
49		73 80	
61			

SNAPTRAN-2 * QUADRATIC C SUB P *

INPUT DATA

ADDRESS

1	10.00000E-09	7.00000E 01	4.93000E 00	2.50000E-03	1.00000E 01	1.93200E 03
7	8.00000E-03	6.50000E-06	1.86000E 00	1.50000E 04	1.12000E 02	1.40000E 02
13	9.38000E-05	3.36000E-04	3.43000E-07	1.00000E 01	1.00000E 02	5.00000E 00
19	10.00000E-08	2.02000E-01	4.60000E 02	7.35000E 01	1.00000E-01	10.00000E-03

REXT= 0.63236E 01

ENERGY KICKOFF

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SNAPTRAN-2 * QUADRATIC C SUB P *

TIME (SEC.)	POWER (M W)	FUEL TEMP. (DEG. F)	GAS TEMP.* (DEG. F)	ENERGY (MW-SEC)	REACTIVITY (DOLLARS)
5.45537E-03	2.58375E 03	1.20000E 02	1.61668E 02	5.43711E-01	4.80500E 00
5.60444E-03	5.13435E 03	1.70000E 02	2.52129E 02	1.09841E 00	4.68000E 00
5.69413E-03	7.65245E 03	2.20000E 02	3.40800E 02	1.66499E 00	4.55500E 00
5.75963E-03	1.01382E 04	2.70000E 02	4.27628E 02	2.24437E 00	4.43000E 00
5.81199E-03	1.25915E 04	3.20000E 02	5.12610E 02	2.83743E 00	4.30500E 00
5.85610E-03	1.50116E 04	3.70000E 02	5.95784E 02	3.44508E 00	4.18000E 00
5.89461E-03	1.73976E 04	4.20000E 02	6.77211E 02	4.06823E 00	4.05500E 00
5.92907E-03	1.97479E 04	4.70000E 02	7.56974E 02	4.70776E 00	3.93000E 00
5.96051E-03	2.20607E 04	5.20000E 02	8.35165E 02	5.36459E 00	3.80500E 00
5.98962E-03	2.43338E 04	5.70000E 02	9.11883E 02	6.03961E 00	3.68000E 00
6.01691E-03	2.65643E 04	6.20000E 02	9.87227E 02	6.73373E 00	3.55500E 00
6.04273E-03	2.87492E 04	6.70000E 02	1.06130E 03	7.44784E 00	3.43000E 00
6.06738E-03	3.08850E 04	7.20000E 02	1.13419E 03	8.18285E 00	3.30500E 00
6.09109E-03	3.29677E 04	7.70000E 02	1.20599E 03	8.93965E 00	3.18000E 00
6.11403E-03	3.49929E 04	8.20000E 02	1.27679E 03	9.71916E 00	3.05500E 00
6.13635E-03	3.69559E 04	8.70000E 02	1.34667E 03	1.05223E 01	2.93000E 00
6.15819E-03	3.88514E 04	9.20000E 02	1.41570E 03	1.13499E 01	2.80500E 00
6.17964E-03	4.06739E 04	9.70000E 02	1.48397E 03	1.22029E 01	2.68000E 00
6.20080E-03	4.24173E 04	1.02000E 03	1.55152E 03	1.30822E 01	2.55500E 00
6.22176E-03	4.40751E 04	1.07000E 03	1.61842E 03	1.39887E 01	2.43000E 00
6.24259E-03	4.56406E 04	1.12000E 03	1.68473E 03	1.49233E 01	2.30500E 00
6.26336E-03	4.71064E 04	1.17000E 03	1.75049E 03	1.58869E 01	2.18000E 00
6.28415E-03	4.84649E 04	1.22000E 03	1.81575E 03	1.68804E 01	2.05500E 00
6.30501E-03	4.97078E 04	1.27000E 03	1.88056E 03	1.79047E 01	1.93000E 00
6.32180E-03	5.06134E 04	1.31000E 03	1.93210E 03	1.87470E 01	1.83000E 00
6.33179E-03	5.11330E 04	1.33205E 03	1.41635E 03	1.92557E 01	1.82999E 00
6.34179E-03	5.16230E 04	1.35519E 03	1.43908E 03	1.97695E 01	1.77473E 00
6.35178E-03	5.20804E 04	1.37778E 03	1.44792E 03	2.02880E 01	1.71633E 00
6.36178E-03	5.25047E 04	1.39974E 03	1.45084E 03	2.08109E 01	1.65848E 00
6.37177E-03	5.28954E 04	1.42103E 03	1.44987E 03	2.13379E 01	1.60079E 00
6.38177E-03	5.32516E 04	1.44163E 03	1.44572E 03	2.18687E 01	1.54263E 00
6.39176E-03	5.35720E 04	1.46153E 03	1.43879E 03	2.24028E 01	1.48323E 00
6.40175E-03	5.38549E 04	1.48072E 03	1.42939E 03	2.29400E 01	1.42171E 00
6.41175E-03	5.40979E 04	1.49921E 03	1.41787E 03	2.34797E 01	1.35708E 00
6.42174E-03	5.42980E 04	1.51700E 03	1.40459E 03	2.40218E 01	1.28834E 00
6.43174E-03	5.44517E 04	1.53410E 03	1.38995E 03	2.45656E 01	1.21446E 00
6.44173E-03	5.45548E 04	1.55052E 03	1.37435E 03	2.51107E 01	1.13441E 00
6.45173E-03	5.46024E 04	1.56629E 03	1.35814E 03	2.56567E 01	1.04721E 00
6.46172E-03	5.45892E 04	1.58147E 03	1.34184E 03	2.62056E 01	9.51901E-01
6.47171E-03	5.45095E 04	1.59594E 03	1.32521E 03	2.67513E 01	8.47413E-01

* This column contains maximum fuel temperature before disintegration starts. Disintegration starts when $T_{\max} = 1932^{\circ}\text{F}$ (arrow). All data for $t > 6.3218$ msec in this column are gas temperatures.

6.48171E-03	5.43573E 04	1.60978F 03	1.30871E 03	2.72957E 01	7.33299E-01
6.49170E-03	5.41264E 04	1.62301E 03	1.29249E 03	2.78383E 01	6.08607E-01
6.50170E-03	5.38107E 04	1.63564E 03	1.27663E 03	2.83780E 01	4.72633E-01
6.51169E-03	5.34039E 04	1.64769E 03	1.26116E 03	2.89142E 01	3.24744E-01
6.52169E-03	5.29001E 04	1.65917E 03	1.24610E 03	2.94458E 01	1.64367E-01
6.53168E-03	5.22941E 04	1.67008E 03	1.23142E 03	2.99718E 01	-9.02183E-03
6.54167E-03	5.15809E 04	1.68045E 03	1.21708E 03	3.04913E 01	-1.95857E-01
6.55167E-03	5.07567E 04	1.69029E 03	1.20305E 03	3.10031E 01	-3.96532E-01
6.56166E-03	4.98185E 04	1.69960E 03	1.18925E 03	3.15060E 01	-6.11379E-01
6.57166E-03	4.87645E 04	1.70841E 03	1.17563E 03	3.19990E 01	-8.40664E-01
6.58165E-03	4.75943E 04	1.71671E 03	1.16213E 03	3.24809E 01	-1.08460E 00
6.59165E-03	4.63091E 04	1.72454E 03	1.14869E 03	3.29505E 01	-1.34333E 00
6.60164E-03	4.49115E 04	1.73189E 03	1.13525E 03	3.34066E 01	-1.61695E 00
6.61163E-03	4.34059E 04	1.73878E 03	1.12175E 03	3.38483E 01	-1.90548E 00
6.62163E-03	4.17985E 04	1.74523E 03	1.10815E 03	3.42744E 01	-2.20889E 00
6.63162E-03	4.00972E 04	1.75124E 03	1.09438E 03	3.46839E 01	-2.52710E 00
6.64162E-03	3.83116E 04	1.75684E 03	1.08043E 03	3.50760E 01	-2.85994E 00
6.65161E-03	3.64527E 04	1.76204E 03	1.06623E 03	3.54498E 01	-3.20722E 00
6.66161E-03	3.45333E 04	1.76685E 03	1.05177E 03	3.58048E 01	-3.56867E 00
6.67160E-03	3.25671E 04	1.77129E 03	1.03702E 03	3.61403E 01	-3.94399E 00
6.68159E-03	3.05689E 04	1.77537E 03	1.02195E 03	3.64500E 01	-4.33283E 00
6.69159E-03	2.85541E 04	1.77911E 03	1.00656E 03	3.67516E 01	-4.73478E 00
6.70158E-03	2.65386E 04	1.78253E 03	9.90829E 02	3.70270E 01	-5.14941E 00
6.71158E-03	2.45380E 04	1.78564E 03	9.74760E 02	3.72823E 01	-5.57626E 00
6.72157E-03	2.25677E 04	1.78846E 03	9.58356E 02	3.75178E 01	-6.01484E 00
6.73156E-03	2.06426E 04	1.79100E 03	9.41626E 02	3.77338E 01	-6.46464E 00
6.74156E-03	1.87763E 04	1.79329E 03	9.24584E 02	3.79308E 01	-6.92512E 00
6.75155E-03	1.69812E 04	1.79534E 03	9.07252E 02	3.81095E 01	-7.39577E 00
6.76155E-03	1.52681E 04	1.79717E 03	8.89655E 02	3.82707E 01	-7.87674E 00
6.77154E-03	1.36462E 04	1.79879E 03	8.71822E 02	3.84151E 01	-8.36540E 00
6.78154E-03	1.21228E 04	1.80022E 03	8.53789E 02	3.85439E 01	-8.86332E 00
6.79153E-03	1.07031E 04	1.80147E 03	8.35594E 02	3.86579E 01	-9.36930E 00
6.80152E-03	9.39057E 03	1.80257E 03	8.17280E 02	3.87582E 01	-9.88284E 00
6.81152E-03	8.18675E 03	1.80353E 03	7.98886E 02	3.88460E 01	-1.04034E 01
6.82151E-03	7.09140E 03	1.80435E 03	7.80460E 02	3.89223E 01	-1.09307E 01
6.83151E-03	6.10264E 03	1.80506E 03	7.62045E 02	3.89881E 01	-1.14641E 01
6.84150E-03	5.21720E 03	1.80566E 03	7.43688E 02	3.90446E 01	-1.20033E 01
6.85150E-03	4.43058E 03	1.80618E 03	7.25433E 02	3.90928E 01	-1.25479E 01
6.86149E-03	3.73731E 03	1.80661E 03	7.07321E 02	3.91335E 01	-1.30976E 01
6.87148E-03	3.13118E 03	1.80697E 03	6.89392E 02	3.91678E 01	-1.36519E 01
6.88148E-03	2.60545E 03	1.80728E 03	6.71683E 02	3.91964E 01	-1.42107E 01
6.89147E-03	2.15308E 03	1.80753E 03	6.54227E 02	3.92201E 01	-1.47735E 01
6.89597E-03	1.97154E 03	1.80762E 03	6.46463E 02	3.92293E 01	-1.50284E 01

SNAPTRAN-2 * QUADRATIC C SUB P *

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TIME (SEC.)	INVERSE PERIOD (1/SEC.)	INNER RADIUS (IN)	OUTER RADIUS (IN)	ACCEL. MASS (LB)	PRESSURE (PSI)
5.45537E-03	4.68308E 03	0.	0.	0.	0.
5.60444E-03	4.52923E 03	0.	0.	0.	0.
5.69413E-03	4.37538E 03	0.	0.	0.	0.
5.75963E-03	4.22154E 03	0.	0.	0.	0.
5.81199E-03	4.06769E 03	0.	0.	0.	0.
5.85610E-03	3.91385E 03	0.	0.	0.	0.
5.89461E-03	3.76000E 03	0.	0.	0.	0.
5.92907E-03	3.60615E 03	0.	0.	0.	0.
5.96051E-03	3.45231E 03	0.	0.	0.	0.
5.98962E-03	3.29846E 03	0.	0.	0.	0.
6.01691E-03	3.14462E 03	0.	0.	0.	0.
6.04273E-03	2.99077E 03	0.	0.	0.	0.
6.06738E-03	2.83692E 03	0.	0.	0.	0.
6.09109E-03	2.68308E 03	0.	0.	0.	0.
6.11403E-03	2.52923E 03	0.	0.	0.	0.
6.13635E-03	2.37538E 03	0.	0.	0.	0.
6.15819E-03	2.22154E 03	0.	0.	0.	0.
6.17964E-03	2.06769E 03	0.	0.	0.	0.
6.20080E-03	1.91385E 03	0.	0.	0.	0.
6.22176E-03	1.76000E 03	0.	0.	0.	0.
6.24259E-03	1.60615E 03	0.	0.	0.	0.
6.26336E-03	1.45231E 03	0.	0.	0.	0.
6.28415E-03	1.29846E 03	0.	0.	0.	0.
6.30501E-03	1.14462E 03	0.	0.	0.	0.
6.32180E-03	1.02154E 03	0.	0.	0.	0.
6.33179E-03	1.02152E 03	9.27413E-01	1.23772E-06	1.39386E 02	3.75258E 03
6.34179E-03	9.53511E 02	1.30041E 00	1.52915E-05	1.38308E 02	6.50575E 03
6.35178E-03	8.81632E 02	1.57917E 00	6.98552E-05	1.36971E 02	9.21291E 03
6.36178E-03	8.10443E 02	1.80807E 00	2.08453E-04	1.35453E 02	1.18208E 04
6.37177E-03	7.39434E 02	2.00449E 00	4.89526E-04	1.33805E 02	1.42717E 04
6.38177E-03	6.67853E 02	2.17744E 00	9.84942E-04	1.32059E 02	1.65081E 04
6.39176E-03	5.94748E 02	2.33232E 00	1.77791E-03	1.30241E 02	1.84803E 04
6.40175E-03	5.19023E 02	2.47269E 00	2.96043E-03	1.28371E 02	2.01522E 04
6.41175E-03	4.39486E 02	2.60104E 00	4.63044E-03	1.26464E 02	2.15039E 04
6.42174E-03	3.54886E 02	2.71922E 00	6.88894E-03	1.24534E 02	2.25338E 04
6.43174E-03	2.63954E 02	2.82860E 00	9.83735E-03	1.22591E 02	2.32555E 04
6.44173E-03	1.65433E 02	2.93030E 00	1.35753E-02	1.20645E 02	2.36954E 04
6.45173E-03	5.81080E 01	3.02519E 00	1.81987E-02	1.18704E 02	2.38873E 04
6.46172E-03	-5.91983E 01	3.11439E 00	2.37994E-02	1.16764E 02	2.38820E 04
6.47171E-03	-1.87799E 02	3.19759E 00	3.04633E-02	1.14851E 02	2.36854E 04

6.48171E-03	-3.28247E 02	3.27573E 00	3.82694E-02	1.12962E 02	2.33477E 04
6.49170E-03	-4.81715E 02	3.34923E 00	4.72907E-02	1.11101E 02	2.28998E 04
6.50170E-03	-6.49067E 02	3.41844E 00	5.75937E-02	1.09272E 02	2.23679E 04
6.51169E-03	-8.31085E 02	3.48367E 00	6.92385E-02	1.07479E 02	2.17736E 04
6.52169E-03	-1.02848E 03	3.54517E 00	8.22790E-02	1.05727E 02	2.11341E 04
6.53168E-03	-1.24187E 03	3.60316E 00	9.67630E-02	1.04017E 02	2.04636E 04
6.54167E-03	-1.47182E 03	3.65784E 00	1.12732E-01	1.02354E 02	1.97727E 04
6.55167E-03	-1.71881E 03	3.70939E 00	1.30223E-01	1.00739E 02	1.90701E 04
6.56166E-03	-1.98324E 03	3.75796E 00	1.49264E-01	9.91769E 01	1.83622E 04
6.57166E-03	-2.26543E 03	3.80369E 00	1.69880E-01	9.76685E 01	1.76542E 04
6.58165E-03	-2.56566E 03	3.84670E 00	1.92088E-01	9.62162E 01	1.69499E 04
6.59165E-03	-2.88410E 03	3.88711E 00	2.15901E-01	9.48220E 01	1.62523E 04
6.60164E-03	-3.22086E 03	3.92501E 00	2.41323E-01	9.34873E 01	1.55640E 04
6.61163E-03	-3.57598E 03	3.96052E 00	2.68354E-01	9.22135E 01	1.48868E 04
6.62163E-03	-3.94941E 03	3.99372E 00	2.96989E-01	9.10017E 01	1.42224E 04
6.63162E-03	-4.34104E 03	4.02470E 00	3.27213E-01	8.98526E 01	1.35719E 04
6.64162E-03	-4.75069E 03	4.05354E 00	3.59008E-01	8.87667E 01	1.29366E 04
6.65161E-03	-5.17811E 03	4.08033E 00	3.92352E-01	8.77441E 01	1.23174E 04
6.66161E-03	-5.62298E 03	4.10515E 00	4.27213E-01	8.67848E 01	1.17151E 04
6.67160E-03	-6.08491E 03	4.12808E 00	4.63558E-01	8.58881E 01	1.11304E 04
6.68159E-03	-6.56348E 03	4.14920E 00	5.01346E-01	8.50535E 01	1.05641E 04
6.69159E-03	-7.05819E 03	4.16858E 00	5.40535E-01	8.42797E 01	1.00166E 04
6.70158E-03	-7.56851E 03	4.18632E 00	5.81076E-01	8.35654E 01	9.48850E 03
6.71158E-03	-8.09386E 03	4.20249E 00	6.22920E-01	8.29090E 01	8.98012E 03
6.72157E-03	-8.63365E 03	4.21717E 00	6.66012E-01	8.23085E 01	8.49176E 03
6.73156E-03	-9.18725E 03	4.23045E 00	7.10298E-01	8.17618E 01	8.02366E 03
6.74156E-03	-9.75400E 03	4.24241E 00	7.55722E-01	8.12664E 01	7.57591E 03
6.75155E-03	-1.03333E 04	4.25313E 00	8.02224E-01	8.08199E 01	7.14853E 03
6.76155E-03	-1.09244E 04	4.26271E 00	8.49749E-01	8.04195E 01	6.74143E 03
6.77154E-03	-1.15266E 04	4.27121E 00	8.98237E-01	8.00623E 01	6.35443E 03
6.78154E-03	-1.21395E 04	4.27872E 00	9.47633E-01	7.97454E 01	5.98727E 03
6.79153E-03	-1.27622E 04	4.28533E 00	9.97882E-01	7.94658E 01	5.63958E 03
6.80152E-03	-1.33943E 04	4.29111E 00	1.04893E 00	7.92207E 01	5.31095E 03
6.81152E-03	-1.40350E 04	4.29613E 00	1.10072E 00	7.90069E 01	5.00083E 03
6.82151E-03	-1.46839E 04	4.30048E 00	1.15322E 00	7.88216E 01	4.70867E 03
6.83151E-03	-1.53404E 04	4.30421E 00	1.20636E 00	7.86620E 01	4.43383E 03
6.84150E-03	-1.60041E 04	4.30741E 00	1.26011E 00	7.85254E 01	4.17564E 03
6.85150E-03	-1.66744E 04	4.31012E 00	1.31442E 00	7.84092E 01	3.93338E 03
6.86149E-03	-1.73509E 04	4.31241E 00	1.36926E 00	7.83110E 01	3.70631E 03
6.87148E-03	-1.80331E 04	4.31433E 00	1.42460E 00	7.82285E 01	3.49369E 03
6.88148E-03	-1.87208E 04	4.31593E 00	1.48038E 00	7.81597E 01	3.29474E 03
6.89147E-03	-1.94136E 04	4.31726E 00	1.53660E 00	7.81027E 01	3.10871E 03
6.89597E-03	-1.97273E 04	4.31777E 00	1.56202E 00	7.80805E 01	3.02900E 03