UNCLASSIFIED

NAA-SR-9646

COPY

70 PAGES

UNCLASSIFIED

MASTER

RESPONSE OF THE SNAP 8 REACTOR
DURING AUTOMATIC STARTUP
(Title Unclassified)

AEC Research and Development Report

~~RESTRICTED DATA~~

This document contains restricted data as defined in the Atomic Energy Act of 1954. Its transmittal or the disclosure of its contents in any manner to an unauthorized person is prohibited.

This document contains Confidential-Restricted Data relating to civilian applications of atomic energy.

UNCLASSIFIED
GROUP 1
Excluded from automatic downgrading and declassification

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.

~~CONFIDENTIAL~~

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

Printed in USA

Price \$1.30

Available from the

U. S. Atomic Energy Commission
Technical Information Extension,
P. O. Box 1001
Oak Ridge, Tennessee.

Please direct to the same address inquiries covering the procurement of other classified AEC reports.

UNCLASSIFIED
~~CONFIDENTIAL~~

Exempt from CCRP Re-review Requirements
(per 7/22/82 Duff/Caudle memorandum) HA 7/04

NAA-SR-9646
SNAP REACTOR,
SNAP PROGRAM
M-3679 (35th Ed.)

Classification cancelled (or changed to)

UNCLASSIFIED

by authority of

Letter 4/17/73
Bram-Feldman Dir of Clean. Work

by

GG

DTIE, date

5/24/73

DC

RESPONSE OF THE SNAP 8 REACTOR
DURING AUTOMATIC STARTUP
(Title Unclassified)

By

S. BIRKEN

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

~~RESTRICTED DATA~~

This document contains restricted data as defined in the Atomic Energy Act of 1954. Its transmittal or the disclosure of its contents in any manner to an unauthorized person is prohibited.

This document contains Confidential-Restricted Data relating to civilian applications of atomic energy.

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.
P.O. BOX 309 CANOGA PARK, CALIFORNIA

UNCLASSIFIED
~~CONFIDENTIAL~~

CONTRACT: AT(11-1)-GEN-8
ISSUED: SEPTEMBER 24, 1964

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED
GG

DISTRIBUTION

SYSTEMS FOR NUCLEAR AUXILIARY POWER (SNAP)-REACTOR SNAP PROGRAM M-3679(35th Ed.)

	No. of Copies
Aerojet-General Corporation (NASA)	6
Aerojet-General Corporation, Sacramento	1
Aerojet-General Nucleonics	1
Aeronautical Systems Division	2
Aerospace Corporation	1
Aerospace Test Wing (AFSC)	1
Air Force Surgeon General	1
Air Force Weapons Laboratory	2
AlResearch Manufacturing Company, Phoenix	1
Army Ballistic Research Laboratories	1
Army Missile Command	1
Army Nuclear Defense Laboratory	1
ARO, Inc.	1
Air University Library	1
Argonne National Laboratory	1
Army Combat Developments Command	1
Astropower, Inc	1
Avco Corporation	1
Battelle Memorial Institute	1
Bendix Corporation (AF)	1
Brookhaven National Laboratory	1
Bureau of Naval Weapons	2
Bureau of Ships	2
Bureau of Yards and Docks	1
California Patent Group	1
Central Intelligence Agency	1
Chicago Patent Group	1
Defense Atomic Support Agency, Sandia	1
Department of the Army	1
Director of Defense Research and Engineering (OAP)	1
Edgerton, Germeshausen and Grier, Inc , Goleta	1
Foreign Technology Division (AFSC)	1
General Atomic Division	1
General Dynamics/Astronautics (AF)	1
General Dynamics/Fort Worth	1
General Electric Company, Cincinnati	1
General Electric Company (FPD)	2
General Electric Company (MSVD)	1
General Electric Company, Richland	2
General Electric Company, San Jose	1
General Electric Company, San Jose (AF)	1
General Technologies Corporation	1
Institute for Defense Analysis	1
Ion Physics Corporation	1
Jet Propulsion Laboratory	2
Johns Hopkins University (APL)	1
Lockheed-Georgia Company	1
Lockheed Missiles and Space Company	1
Los Alamos Scientific Laboratory	1
Martin-Marietta Corporation, Denver	1
Monsanto Dayton Laboratory	1
Mound Laboratory	1
NASA Ames Research Center	1
NASA Goddard Space Flight Center	2
NASA Langley Research Center	1
NASA Lewis Research Center	4
NASA Manned Spacecraft Center	1
NASA Marshall Space Flight Center	1
NASA Scientific and Technical Information Facility	3
National Aeronautics and Space Administration, Washington	2
NASA Western Operations Office	1
Naval Air Development Center	1
Naval Ordnance Laboratory	2
Naval Postgraduate School	1
Naval Radiological Defense Laboratory	1
Naval Research Laboratory	2
Naval Underwater Ordnance Station	1
Navy Marine Engineering Laboratory	1
New York Operations Office	1
New York Operations Office, Canal Project Office	1
North American Aviation, Inc , Downey	1
Nuclear Metals, Inc	1
Office of Naval Research	2
Office of the Assistant General Counsel for Patents (AEC)	1
Office of the Chief of Engineers	1
Office of the Chief of Naval Operations	3
Office of the Chief of Naval Operations (OP-03EG)	2
Office of the Chief of Transportation	1
Phillips Petroleum Company (NRTS)	4
Pratt and Whitney Aircraft Division	1
Pratt and Whitney Aircraft Division (NASA)	1
Rand Corporation	1
Republic Aviation Corporation	1
Sandia Corporation	1
School of Aerospace Medicine	1
Union Carbide Corporation (ORNL)	8
USAF Headquarters	1
University of California, Livermore	1
Westinghouse Electric Corporation (NASA)	1
Westinghouse Electric Corporation, Lima (AF)	1
Division of Technical Information Extension	40
AI Library (Includes 2 copies to CPAO, 2 copies to AEC, Washington, 2 copies to COO)	120

CONTENTS

	Page
Abstract	7
I. Introduction	9
A. Description of SNAP 8	9
1. Fuel Elements	9
2. Thermal Power	10
3. Temperature	10
4. Endurance	10
5. Control System	10
6. Startup	12
7. Shielding	12
8. Configuration	12
9. Reliability	12
B. SNAP 8 Startup Scheme	13
C. Limitations of Startup Rates	13
D. Method of Analyzing Nuclear System Startup	15
II. Results of the Startup Analysis	17
A. Maximum Fuel Temperature Rates	17
B. Effect of Additional Drum Rates	19
C. Effect of Source Size	21
D. Effects of Temperature Coefficients	21
E. Effect of Startup Flow Rate	22
F. Effect of Ramps in Coolant Flow	23
Appendix I. Nuclear Kinetics Equations and Constants	27
A. Digital Equations	27
B. Analog Equations	28
C. Values of Constants	29
Appendix II. Heat Transfer Equations and Constants	31
A. Introduction	31
B. Fuel Equations	31
1. Temperature	33
2. Normalized Power	33
3. Heat Generation Rate	33

CONTENTS

	Page
4. Specific Heats	33
5. Fuel-to-Coolant Heat Transfer Coefficient	33
6. Mass	34
C. Core Coolant Equations	34
1. Coolant Mass	34
2. NaK Flow Rate	35
3. Specific Heat	35
4. Fuel-to-Coolant Heat Transfer Coefficient	35
D. Inlet Coolant Equations	35
1. Transport Delays	35
2. Temperature Losses	37
E. Upper and Lower Grid Plate Equations	38
1. Mass	39
2. Specific Heat and Conductivity	39
3. Heat Transfer Areas and Path Lengths	39
F. Auxiliary Heat Exchanger (AHX) Equations	41
G. Intermediate Heat Exchanger (IHX) Equations	42
H. Hot Channel Equations	42
Appendix III. Analog Computer Diagrams and Potentiometer Settings	45
Appendix IV. Symbols and Subscripts	63
Appendix V. Nomenclature	67
References	69

FIGURES

1. Artist's Concept of SNAP 8 Electrical Generating System During a Mars Fly-By Mission	8
2. SNAP 8 Fuel Element Cross Section	10
3. SNAP 8 Nuclear System	11
4. SNAP 8 Startup Sequence	14

FIGURES

	Page
5. Rate of Change of Maximum Fuel Temperature During the Initial Startup Transient	18
6. Time for Outlet Coolant to Reach Deadband Temperature	18
7. Effects of Source, Additional Drum Rates, and Higher Allowable Fuel Transients	20
8. Allowable Exposure to Startup Source	20
9. Excess Reactivity Due to Control Drums	30
10. Fuel Heat Generation Rate	32
11. Heat Capacity of the Fuel	32
12. Overall Fuel to Coolant Heat Transfer Coefficient	34
13. Schematic of Temperature Losses and Transport Delays	36
14. Heat Capacity of Grid Plates	40
15. Conductivity of Grid Plates	40
16. Ratio of Grid Plate Conductivity to Heat Capacity	40
17. Analog Computer Circuitry Symbols	50
18. Kinetics and Reactivity Feedback Simulation	51
19. Typical Fuel and Coolant Node Simulation	52
20. Transport Delay and Primary Loop Heat Loss Simulation	53
21. Auxiliary Heat Exchanger Simulation	54
22. Intermediate Heat Exchanger (IHX) Simulation	55
23. Controller Simulation	56
24. Console B Simulation	59
25. Console C Simulation	61

ACKNOWLEDGMENT

The assistance of L. N. Wimberly was extremely valuable in setting up and operating the analog computers for the startup studies. The support furnished by D. L. DeHaven in completing some of the parameter studies helped shorten the time necessary for the preparation of material for this report.

ABSTRACT

To determine the response of the SNAP 8 nuclear system to automatic startup, the system was simulated using both digital and analog computer models. Parameter studies were made to determine the effects of startup source size, NaK coolant flow, temperature coefficients, and additional drum rates on both fuel temperature transients and startup time.

The detailed results of the parameter studies are presented. The analog model and digital program are discussed in detail, including circuit diagrams, equations, and constants used in the simulation.

4-10-64

7568-01025

Figure 1. Artist's Concept of SNAP 8 Electrical Generating System
During a Mars Fly-By Mission

I. INTRODUCTION

Maximum permissible startup rates for the SNAP 8 ground test and flight nuclear systems have been studied. It was necessary to examine (1) a suitable startup scheme which will result in a reliable, simple control system; (2) limiting the transients in the system so no damage can occur to the reactor and its associated components; (3) keeping the startup time to a minimum so as to have a low battery requirement during the startup phase.

The study covers preheating the nuclear system to rated temperature at the startup coolant flow rates, but does not include the startup of the power conversion system¹(PCS).

A. DESCRIPTION OF SNAP 8

SNAP 8 is a nuclear powerplant which will produce approximately 35 kw of electric power output for use in spacecraft. The system, which is being developed jointly by NASA and AEC, employs a nuclear system, being developed by Atomics International under contract to AEC, as the heat source for a mercury Rankine cycle power conversion system, being developed by Aerojet - General Corporation under contract to NASA (Figure 1).

The SNAP 8 nuclear system consists of a shielded compact nuclear reactor moderated by zirconium hydride and cooled by liquid NaK-78. The reactor is controlled by movable beryllium reflectors which control the amount of leakage flux by their position.

The performance requirements of SNAP 8 are as follows:

1. Fuel Elements

The SNAP 8 reactor core contains 211 fuel elements of Zr-10.5 wt % U alloy hydrided to a hydrogen content of 6×10^{22} hydrogen atoms/cc of $(U-Zr)H_x$. The elements are clad in 10-mil-thick Hastelloy-N tubing coated internally with a hydrogen permeation barrier. A radial gap between the fuel material and the cladding is used to accommodate radiation swelling, differential expansion, etc., of the Zr-U hydride. A schematic of the fuel element is shown in Figure 2.

5-7-64

7568-01058

Figure 2. SNAP 8 Fuel Element Cross Section

2. Thermal Power

The SNAP 8 nuclear system is designed to transfer 600 thermal kw to the NaK coolant. The system may be operated at any constant power level desired from approximately 10 to 100% of rated power.

3. Temperature

The nuclear system is designed for a NaK coolant outlet temperature of $1300 \pm 30^\circ\text{F}$. The coolant inlet temperature is nominally 1100°F .

4. Endurance

The nuclear system is designed for 10,000 hour endurance.

5. Control System

The radial neutron reflector contains six movable semicylindrical sections or drums as shown in Figure 3. Reactor control is accomplished by rotation of these drums toward or away from the core to increase or decrease the fraction of leakage neutrons reflected back to the core.

Three of the six reflector drums are designated as startup drums and will be rotated to the "full-in" position by springs. The remaining three drums are control drums, and will be rotated toward or away from the core by geared-down stepper motors.

Control of the SNAP 8 nuclear system is based on its outlet coolant temperature. This temperature is maintained at approximately 1300°F by an on-off digital acting control system which leaves a deadband of from $1270 \pm 10^\circ\text{F}$ to $1330 \pm 10^\circ\text{F}$. When the outlet coolant NaK temperature is within the deadband, the controller does not change the position of the control drums. Whenever

7-22-63

7568-1217-A

Figure 3. SNAP 8 Nuclear System

NAA-SR-9646

the measured outlet coolant temperature goes outside the control deadband, the control system steps the control drums 0.52 ± 0.10 degrees in the proper direction to return the outlet coolant temperature to within the deadband. If a single control drum step is not sufficient, the control system continues to provide steps at discrete time intervals until the outlet coolant temperature remains within the deadband.

6. Startup

The complete nuclear and power conversion system (PCS) startup is accomplished with the use of nonnuclear instrumentation. Nuclear startup is accomplished by varying the control drum insertion rate. Three rates are used: a rapid pre-critical rate, a slow through-critical rate, and a fast rate to bring the nuclear system to within an operating temperature deadband. The first rate is initiated by a ground startup command, the second by control drum angle, and the third by primary coolant temperature. Finally, the outlet coolant temperature is used to control the nuclear system during long-term operation.

7. Shielding

The reference design for the SNAP 8 shield consists of a conical lithium hydride shield and a cylindrical tungsten-alloy shield. This shield is designed to limit the integrated dose at the PCS to 5×10^{12} nvt from fast neutrons and 5×10^7 rads (carbon) from gammas for 10,000 hours of reactor operation at 600 kw. It is also designed to limit the integrated dose at the 20-ft-diameter, 40-ft-distant payload to 10^{11} nvt (fast neutrons) and 10^6 rads (C) for the same operating conditions.

8. Configuration

The reactor and shield are designed to be launched with the reactor above the shield. The relative positions of reactor and shield to the PCS are depicted in Figure 3.

9. Reliability

A tentative reliability goal of approximately 96% has been set for the reactor and its associated shield and controls to survive launch, achieve automatic startup, and operate for 10,000 hr. This design reliability figure includes an allowance for micrometeoroid protection.

B. SNAP 8 STARTUP SCHEME

Nuclear system startup is initiated by a signal to the startup programmer located in the payload compartment of the spacecraft. The startup programmer automatically sequences the disarming of the reactor launch safety systems and energizes the control drum motors to drive the control drums in at a preset rate to take the reactor to criticality and to rated operating temperature. NaK coolant flow (at least 20% of rated flow) must be maintained through the core during startup. The startup sequence, illustrated schematically in Figure 4, proceeds as follows: the environmental cover is ejected overboard. The six control-drum lockout pins are released. Three startup drums are spring driven to the full-in position after startup is initiated. Simultaneously, the three control drums are stepped inward fairly rapidly until the reactor is approximately \$0.50 subcritical. At this point, the stepping rate is slowed down by a signal to the programmer from a preset position-sensing switch on the control-drum shafts. The control drums continue stepping inward until the NaK temperature reaches 1270°F, approximately 3 hours after initiation of startup. The closing of the lower deadband temperature-sensor switch deactivates the startup programmer, and the control system is switched to the long-term temperature control mode.

During temperature control, for the 10,000-hour life of the system, if the low or high temperature (1270 and 1330°F) switches are closed, the control drums are stepped approximately 1/2° every 4 minutes until the NaK temperature is within the control band.

C. LIMITATIONS OF STARTUP RATES

The startup rates are limited by the transients they impose on the nuclear system. The first limitation reached is the rate of change of the fuel temperature. This limit is a maximum fuel temperature rate of 150°F/min.

Other limitations on the startup rates may be imposed by the safety criteria and coolant scram points; however, at present the only limitation which determines the minimum allowable startup time is the maximum fuel temperature rate.

The preceding discussion on the limitations of the startup rate assumes a fixed startup power level. In the initial phase of investigating the startup time,

5-5-64

7568-80240D

Figure 4. SNAP 8 Startup Sequence

a survey was made using different startup source powers. The lower the source power, the shorter the reactor period at the time when sensible heat begins to be generated, and, consequently, the larger the initial transient. Since the startup rates can be increased if the shutdown power is increased, it can be concluded that the startup rates are limited indirectly by the shutdown source levels.

The overall fuel-to-coolant heat transfer coefficient limits the startup time because of its influence on the rate of change of fuel temperature. The overall fuel-to-coolant heat transfer coefficient is determined by the location of the fuel within the cladding and the thermal conductivity of the gas in the resulting gap. For a low heat transfer coefficient, heat cannot readily be removed by the coolant. This results in a large rate of change of fuel temperature. Because of the uncertainty in the positioning of the fuel rod within an element, there is an uncertainty in the overall fuel-to-coolant heat transfer coefficient which, in turn, results in uncertainties in the prediction of startup rates or minimum startup time.

D. METHOD OF ANALYZING NUCLEAR SYSTEM STARTUP

The startup problem may be divided into two separate phases. In the first phase, the reactor power increases from source level ($\rho = -50\%$) to the point where sensible heat is generated. In the second phase, sensible heat is produced and the various reactivity feedbacks, through temperature coefficients, affect the kinetic behavior of the reactor.

There is a portion of the actual problem which is neglected in the kinetic analysis since it does not affect the final results, that is, the time during which the three startup drums are spring-driven into the full-in position and the control drums move to the $\rho = -50\%$ position. Since the core is still subcritical, it is assumed that after the drums rotate in, the core is at a steady-state power level due to the external Po-Be neutron source and any subcritical multiplication. It was calculated that the core power will increase by a factor of about 200 from the condition when the six drums are full-out to the point at which the core is 50% subcritical.

The first portion of the problem (i. e. , no sensible heat generation) can be solved using the digital computer. At present, the AIREK III code² is being

used. The code solves the time-dependent core kinetic equations, and the output of the program gives the initial values for the second portion of the problem, which is solved using the analog computer.

The analog model simulates the nuclear and thermal characteristics of the SNAP 8 primary loop. The nuclear portion of the model contains the standard nuclear kinetic equations with six groups of delayed neutrons. The thermal representation of the core utilizes a technique of taking the physical system and breaking it up into small coupled nodes.³ The thermal representation of the remainder of the primary loop includes simulation of the piping losses, heat exchangers, and transport delays. In addition, the control system of the reactor is simulated.

The equations and detailed methods used for the solution of the startup problem are described in the Appendices I through V.

II. RESULTS OF THE STARTUP ANALYSIS

Using the AIREK III digital computer code² and analog simulation described in Appendices I through III of this report, a study was made of the startup behavior of SNAP 8. The variables studied were initial subcritical power level, effects of limiting fuel temperature rates, NaK flow, additional control drum rate, and temperature coefficients of reactivity. In addition, the effect of the flow ramp from 20 to 50% of rated flow prior to PCS startup was investigated.

For the purposes of these analyses, the startup time is defined as the time for the control drum movements to bring the nuclear system from the 50-cents-subcritical condition to the condition at which the outlet coolant temperature reaches the lower deadband setpoint. The time it would take the control drums to move from their full-out position to the 50-cents-subcritical position (approximately 30 minutes) is not considered as a variable at this time. After reaching deadband temperatures, some time is necessary for stabilization in the deadband as well as for increasing the NaK flow prior to PCS startup. This additional time is considered in this section of the report so that total startup time from startup signal to the end of preheat may be determined.

A. MAXIMUM FUEL TEMPERATURE RATES

At present, the maximum fuel temperature rate is assumed to be limited to 150°F/min. This limitation is due to the allowable rate of change of fuel element ceramic (hydrogen barrier) temperature. It was assumed, in this study, that the fuel could be touching the ceramic and, consequently, the rate of change of fuel and ceramic temperatures could be the same. For any given source, this maximum allowable rate dictates the value of the control drum stepping interval during startup. The analog simulation gives the maximum fuel temperature rate for the average as well as maximum channel and node. The maximum rate of change of fuel temperature was surveyed for varying source sizes and control drum rates, and is shown in Figure 5. The power levels shown correspond to the subcritical reactivity of -50 cents for source sizes between 0.68 and 680 curies.

It is obvious that a relaxation of the present requirement of 150°F/min to larger rates of change could considerably reduce startup time. Figure 6 shows the startup time if there were no restrictions on rates of change of fuel

5-4-64

7568-01017

Figure 5. Rate of Change of Maximum Fuel Temperature During the Initial Startup Transient

5-4-64

7568-01043

Figure 6. Time for Outlet Coolant to Reach Deadband Temperature

temperature (assuming also that no other restrictions would limit control drum rates). Using the present limitation of $150^{\circ}\text{F}/\text{min}$ ($2.5^{\circ}\text{F}/\text{sec}$) and a source power level of 0.017 watts at -50 cents, the allowable drum rate is 90 seconds/step. From Figure 5, the startup time is 9600 seconds, or 2.7 hours. If the fuel temperature limitation were doubled to $300^{\circ}\text{F}/\text{min}$, the allowable drum rate for the same source would be approximately 50 seconds/step. The startup time would be about 5200 seconds, or about 1.5 hours.

Fuel elements are being tested to determine the maximum allowable rates and it is anticipated that the $150^{\circ}\text{F}/\text{min}$ transient limitation may be relaxed in the near future. All indications are that the SNAP 8 ceramic coatings can withstand higher transients.

B. EFFECT OF ADDITIONAL DRUM RATES

At present, there are three control drum rates. The first rate is a fast rate that moves the control drums from the full-out position to a position such that the drums are at a reactivity of -50 cents. The second rate is a slow through-critical rate, at which the drums move from their 50-cent-subcritical positions to their operating-coolant-temperature positions. The third, or final, rate is achieved when the outlet coolant temperature reaches the operating deadband. This is the long-term control rate which is used during PCS startup as well as during the remainder of the operating lifetime of the nuclear system.

Since there is a restriction on the through-critical rate imposed by the allowable fuel or ceramic temperature rates, it was thought that a fourth rate could be used after the initial startup transient to reduce startup time. This rate would be faster than the through-critical rate since feedback through the negative temperature coefficients or reactivity would limit the transient. The effect of such an additional rate is shown in Figures 6 and 7. The restriction on the additional rate during this study was that the maximum fuel temperature rates be limited to $150^{\circ}\text{F}/\text{minute}$.

For a through-critical rate of 90 seconds/drum step (i.e., $150^{\circ}\text{F}/\text{min}$ fuel transient) and a 30 second/drum step intermediate rate initiated approximately three or four minutes after the startup transient, the savings in startup time would be about one hour.

5-4-64

7568-01015

Figure 7. Effects of Source, Additional Drum Rates, and Higher Allowable Fuel Transients

4-6-64

7568-01016

Figure 8. Allowable Exposure to Startup Source

C. EFFECT OF SOURCE SIZE

The initial subcritical power level (50¢ subcritical) at the time the through-critical rate is initiated determines the fuel temperature transient. For any fixed through-critical drum rate, use of a smaller source would result in a shorter nuclear period at sensible heat which, in turn, would cause a higher fuel temperature transient. Since the transient limits the allowable drum rates (and the minimum startup times), the startup time can be reduced by using a source as large as is considered reasonable and within the limitations of safety.

For SNAP 8, a startup Po-Be source could be located on the outlet NaK pipe above the reactor core. Calculations show that the unmultiplied source strength is about 1 microwatt/curie. However, data extrapolated from S8ER indicate that the unmultiplied source could be as high as 16 microwatts/curie. For the purposes of the startup study, the lower value of 1 microwatt/curie was used. For the subcritical reactivity of -50 cents, the corresponding multiplication factor is $k_{\text{eff}} = 0.996$. Therefore, the per-curie power level is $0.996 / (1 - 0.996) \approx 0.00025$ watts/curie.

The dose rate to personnel exposed to the source prior to launch determines the allowable source size. It is expected that the exposure time could be as short as a few minutes. Using the maximum permissible dose rates established for workers in the field of nuclear energy, the allowable times of exposure for personnel 3 feet or more from the source⁴ are shown in Figure 8.

D. EFFECTS OF TEMPERATURE COEFFICIENTS

Because experimental verification of the temperature coefficients of reactivity has not been achieved at this date, a survey was made varying the temperature coefficients over their anticipated range to determine the effects on startup.

Before the analysis was started, a representative "base case" was chosen. This was the case of a drum rate of 90 seconds/drum step with a 0.017-watt startup source power level. For the base case, the rate of change of fuel temperature was the present limitation of 150°F/min.

The lower grid plate temperature coefficient with a nominal value of -0.04¢/°F was varied over the range of -0.04 to 0.06¢/°F. Increasing the magnitude of the lower grid plate coefficient did not affect the fuel transients or the startup time.

The upper grid plate temperature coefficient of $-0.06\text{¢}/^{\circ}\text{F}$ was varied to $-0.05\text{¢}/^{\circ}\text{F}$. Decreasing the magnitude of the upper grid plate coefficient from -0.06 to $-0.05\text{¢}/^{\circ}\text{F}$ results in an increase of the fuel temperature transient by 15%. Other than an increase in the coolant and fuel temperature rates by an equivalent percent, no other effects were observed.

The fuel temperature coefficient used in all of the analyses in this report, with the exception of this investigation of effects of fuel temperature coefficients, was $-0.05\text{¢}/^{\circ}\text{F}$ over the complete temperature range. For the purposes of this study of temperature coefficient effects, the fuel temperature coefficient was varied linearly from ambient to operating temperature. Three straight-line variations were studied. For fuel temperatures of 68 and 1300°F , the corresponding sets of fuel temperature coefficients used were -0.006 and $-0.04\text{¢}/^{\circ}\text{F}$, -0.016 and $-0.05\text{¢}/^{\circ}\text{F}$, -0.036 and $-0.07\text{¢}/^{\circ}\text{F}$. In all cases of varying fuel temperature coefficients, the rates of change of fuel temperature were higher than the rates of the base case because of the lower temperature coefficients. For the $-0.006\text{¢}/^{\circ}\text{F}$ case, the increase in maximum fuel temperature rate was 45%; for the $-0.016\text{¢}/^{\circ}\text{F}$ coefficient, the increase was 33%; and for the $-0.036\text{¢}/^{\circ}\text{F}$ coefficient, the increase was 11%. Consequently, since the rates of change of fuel temperature increase with decreasing negative fuel temperature coefficient, the through-critical control drum rate would have to be slower. This could increase the startup times discussed in Sections II-B and II-C. On the other hand, the integrated reactivity loss due to a temperature-dependent fuel temperature coefficient would be less than the constant value.

The time saved from sensible heat to the lower deadband temperature, because of a smaller cold-to-hot reactivity loss in the case where the temperature coefficient varied between $-0.006\text{¢}/^{\circ}\text{F}$ at 70°F and $-0.04\text{¢}/^{\circ}\text{F}$ at 1300°F , was 1100 seconds. Consequently, although the through-critical rate must be decreased because of the higher fuel transient, the smaller reactivity loss will tend to offset, at least in part, the effect of the smaller temperature coefficient.

E. EFFECT OF STARTUP FLOW RATE

During all the startup cases discussed in previous sections, the NaK flow in the primary loop was 20% of rated flow. Perturbations were made from this 20% value to determine if this was the optimum flow. The base case was the

90 sec/drum step case that corresponds to a fuel temperature transient of 150°F/min. The results were as follows.

<u>% of Rated Flow</u>	<u>Rate of Change of Fuel Temperature (°F/sec)</u>
5	3.0
10	2.5
15	2.4
20	2.5
30	2.8
40	3.0
50	3.2

It appears that there are two competing effects which change the fuel temperature transient. First, as the flow decreases, the heat transferred from fuel to coolant decreases so that the rate of change of fuel temperature increases. The other effect is that, as flow decreases, the increase in fuel temperature increases the negative reactivity feedback through both the fuel temperature coefficient and upper grid plate coefficient. This results in a decrease in core power which reduces the rate of change of fuel temperature. Consequently, there is an optimum flow during startup, with regard to the first fuel transient. The optimum is between 10 and 20% of rated flow.

F. EFFECT OF RAMPS IN COOLANT FLOW

At some time near the end of the nuclear system startup, the primary NaK flow will be increased from startup flow rates to 50% of rated flow to prepare for PCS startup. A survey was made to determine the necessary time for this ramp increase of NaK flow. The lower deadband temperature sensor was used to provide a signal for the initiation of the flow ramp. The following table summarizes the survey.

<u>Time for 20-50% Flow Ramp (min)</u>	<u>Rate of Change of Outlet Coolant Temperature (°F/sec)</u>	<u>Rate of Change of Inlet Coolant Temperature (°F/sec)</u>
1	-3.0	2.5
2	-2.0	1.7
3	-1.8	1.4

Consequently, the total time for the flow ramp should be greater than 1-1/2 minutes in order that the fuel temperature does not exceed transient limitations.

BLANK

APPENDICES

BLANK

APPENDIX I

NUCLEAR KINETICS EQUATIONS AND CONSTANTS

A. DIGITAL EQUATIONS

The standard kinetic equations are

$$\dot{n} = \frac{n(\delta k_e - k_e \beta)}{\ell^*} + \sum_{i=1}^6 \lambda_i C_i + S \quad , \quad \dots(1)$$

$$\dot{C}_i = \frac{k_e \beta_i n}{\ell^*} - \lambda_i C_i \quad , \quad \dots(2)$$

where

n = neutron flux or power

C = delayed neutron precursor concentration

S = source term

ℓ^* = neutron lifetime

k_e = effective multiplication factor

$\delta k_e = k_e - 1$

β = effective delayed neutron fraction

λ = decay constant of delayed neutron precursor

subscript i = i^{th} delayed neutron precursor group.

Dividing the numerator and denominator of the first term on the right side in both equations by k_e , and defining

$$\rho = \frac{\delta k_e}{k_e} \text{ and } \ell = \frac{\ell^*}{k_e} \quad ,$$

results in the following equations used in the AIREK code:

$$\dot{n} = \frac{n(\rho - \beta)}{\ell} + \sum_{i=1}^6 \lambda_i C_i + S \quad , \quad \dots(3)$$

It should be pointed out that the assumption made in the code is that the ratio ℓ^*/k_e is assumed constant. The code does not assume $\ell^* = \ell$ although Cohen⁵ calls ℓ the "neutron lifetime." For many cases (e.g., reactor with control rods) the assumption that ℓ^*/k_e is a constant is good. Computer runs were made with the lifetime changed by a factor of 10 and the difference in power and delay neutron precursor concentrations was small at the time of sensible heat generation.

B. ANALOG EQUATIONS

The nuclear kinetics analog equations are similar to Equations 1 and 2 but are in a slightly different form. In Equation 1 the source term is left out since it is negligible at the time of sensible heat generation. The power and the delayed precursors in Equations 1 and 2 are normalized to a reference state n_r and c_r

$$\frac{d}{dt} \left[\frac{n(t)}{n_r} \right] = \frac{\rho(t) - \beta}{\ell} \frac{n(t)}{n_r} + \sum_{i=1}^6 \lambda_i \frac{c_i(t)}{n_r} , \quad \dots (5)$$

$$\frac{d}{dt} \left[\frac{c_i(t)}{c_{ir}} \right] = \frac{\beta_i}{\ell} \frac{n(t)}{c_{ir}} - \lambda_i \frac{c_i(t)}{c_{ir}} . \quad \dots (6)$$

At the steady-state reference level the left side of Equation 6 is zero and

$$n_r = \frac{\lambda_i c_{ir}}{\beta_i / \ell} .$$

Substituting for n_r in the last term of Equation 5 and the first term of Equation 6 and letting

$$N = \frac{n(t)}{n_r} \text{ and } C_i = \frac{c_i(t)}{c_{ir}} ,$$

$$\frac{dN}{dt} = \frac{\rho(t) - \beta}{\ell} N + \sum_{i=1}^6 C_i \frac{\beta_i}{\ell} ,$$

$$\frac{dC_i}{dt} = \lambda_i N - \lambda_i C_i .$$

Equation 5 may be manipulated to put it in a more useful form by factoring out β/ℓ and letting $\rho(\text{reactivity})/\beta = \rho(\text{dollars})$,

$$\frac{dN}{dt} = \frac{\beta}{\ell} \left[\rho(t)N - N + \sum_{i=1}^6 \frac{\beta_i}{\beta} C_i \right] . \quad \dots(7)$$

Equations 6 and 7 are used for the solution of the kinetic behavior of the core on the analog computer. The initial values of the precursors, power level, and reactivity are obtained from the digital model. The changeover point is about 500 watts. This allows the analog computer to cover up to 500-kw core power. In all cases run, the energy release at the changeover point was less than 0.5 Mw-sec (which corresponds to about a 10°F rise in coolant temperature).

C. VALUES OF CONSTANTS

The following constants were used in the kinetic analysis:

$\ell = 8.1 \times 10^{-6} \text{ sec}$	
$\beta = 0.0077$	
$\alpha_1 = \beta_1/\beta = 0.03297$	$\lambda_1 = 0.0124 \text{ sec}^{-1}$
$\alpha_2 = 0.2194$	$\lambda_2 = 0.0305 \text{ sec}^{-1}$
$\alpha_3 = 0.1960$	$\lambda_3 = 0.111 \text{ sec}^{-1}$
$\alpha_4 = 0.3947$	$\lambda_4 = 0.301 \text{ sec}^{-1}$
$\alpha_5 = 0.1150$	$\lambda_5 = 1.13 \text{ sec}^{-1}$
$\alpha_6 = 0.04193$	$\lambda_6 = 3.00 \text{ sec}^{-1}$

The reactivity introduced by the control drums (ρ_D) is shown in Figure 9.

5-4-64

7568-01018

Figure 9. Excess Reactivity
Due to Control Drums

APPENDIX II

HEAT TRANSFER EQUATIONS AND CONSTANTS

A. INTRODUCTION

The differential equations for heat transfer in the core can be written in a series of axial nodal equations. The model used in this study contains a fuel entrance node and five axial core nodes. It has been shown³ that this model gives correct average fuel temperatures and eliminates the anomalous dip in outlet coolant temperature. The coolant transport delays in the piping and the lags in the boiler are taken into account. The upper and lower grid heating is considered to be due entirely to the coolant. Axial heat transfer from the fuel elements to the grid plates and from one fuel node to an adjacent fuel node is neglected.

B. FUEL EQUATIONS

The fuel equations are

$$\dot{T}_{fe}(t) = \frac{G_e}{M_{fe}C_f} N(t) - \frac{UA_e}{M_{fe}C_f} [T_{fe}(t) - T_{cin}(t)] \quad , \quad \dots (8)$$

$$\dot{T}_{fi}(t) = \frac{G_i}{M_{fi}C_f} N(t) - \frac{UA_i}{M_{fi}C_f} [T_{fi}(t) - T_{ci}(t)] \quad , \quad \dots (9)$$

where

subscript i = "ith" node (i = 1, 2, 3, 4, or 5)

subscript e = entrance node

subscript f = fuel

subscript in = inlet

subscript c = coolant

T = temperature (°F)

M = mass (pounds)

C = specific heat (Btu/lb-°F)

Figure 10. Fuel Heat Generation Rate

3-24-64

7568-01002

Figure 11. Heat Capacity of the Fuel

4-21-64

7568-01032

A = heat transfer area (ft²)

U = heat transfer coefficient (Btu/sec-ft²-°F)

N = ratio of reactor power to rated power

G = heat generation rate at rated power (Btu/sec).

1. Temperature

Initially all temperatures in the system are set at 70°F.

2. Normalized Power

The reference power level is considered to be 500 kw. The analog computer was scaled such that 500 kw power was equal to 100 volts.

3. Heat Generation Rate

The heat generation rates used (Figure 10) were as follows (474 Btu/sec = 500 kw):

$$\left. \begin{array}{l} G_e = 5.593 \text{ Btu/sec} \\ G_1 = 55.93 \text{ Btu/sec} \\ G_2 = 109.98 \text{ Btu/sec} \\ G_3 = 131.79 \text{ Btu/sec} \\ G_4 = 114.25 \text{ Btu/sec} \\ G_5 = 62.10 \text{ Btu/sec} \end{array} \right\} \quad G = \sum_{i=1}^5 G_i = 474 \text{ Btu/sec}$$

4. Specific Heats

The specific heat of the fuel was considered to be the same as that of ZrH shown in Figure 11. The function was fitted with a line and the equation was used in the analog model. The equation is $C_f = 0.0906 + 9.4 \times 10^{-5} \bar{T}_f$ where $\bar{T}_f$ is the average fuel temperature:

$$\bar{T}_f = 1/6 \left(T_{fe} + \sum_{i=1}^5 T_{fi} \right) .$$

5. Fuel-to-Coolant Heat Transfer Coefficient

The heat transfer coefficient used for the average core and maximum channel simulation is shown in Figure 12. The maximum or hot channel simulation is discussed in Section H of this appendix.

4-15-64

7568-01019

Figure 12. Overall Fuel to Coolant Heat Transfer Coefficient

6. Mass

The fuel mass used was 33.4 lb for each of the five fuel nodes and 3.34 lb for the entrance node.

C. CORE COOLANT EQUATIONS

The coolant equations are

$$\dot{T}_{ci}(t) = \frac{UA_i}{M_{ci}C_c} [T_{fi}(t) - T_{ci}(t)] - \frac{w}{M_{ci}} [T_{ci}(t) - T_{c(i-1)}(t)], \dots (10)$$

$$T_{co}(t) = T_{c5}(t), \dots (11)$$

$$\bar{T}_c(t) = \frac{T_{cin}(t) + T_{co}(t)}{2}, \dots (12)$$

where the notation is the same as in the previous section, with the following additions:

subscript o = outlet from reactor

subscript i-1 = inlet to reactor when i=1

w = mass flow rate of coolant (lb/sec).

1. Coolant Mass

The variation of coolant density with temperature was taken into account by fitting an equation for a straight line. The equation used was

$$M_{ci} = 1.008 - 1.4 \times 10^{-4} \bar{T}_c \dots (13)$$

2. NaK Flow Rate

The coolant mass flow rate was assumed to be 20% of rated flow with corrections for pumping rate due to changes in NaK density during the preheat phase. The equation used in the simulation was:

$$W_r(50\%) = 5.127 - 0.000578T_{p_3},$$

where

$W_r(50\%)$ = flow at 50% rated NaK flow (lb/sec)

T_{p_3} = auxiliary heat exchanger outlet temperature ($^{\circ}\text{F}$).

For 20% flow, all values in the above equation were reduced by a factor of 2.5.

3. Specific Heat

The specific heat for the coolant (C_c) used in the analog calculations was 0.209 Btu/lb- $^{\circ}\text{F}$.

4. Fuel-to-Coolant Heat Transfer Coefficient

Figure 12 shows the fuel-to-coolant heat transfer coefficient.

D. INLET COOLANT EQUATIONS

The inlet coolant equation considers the delay in transport of the coolant in the piping, the temperature loss in the pipes, the lag in the heat exchanger, and effects of the heat exchanger on coolant temperature. Figure 13 is a schematic of the pipe delays and temperature losses.

1. Transport Delays

Delays due to coolant transport from core outlet to the IHX and from the IHX to the core inlet were considered. The delays are dependent upon flow rates at any time. The lag from the core outlet to the IHX is $(t - \tau_1)$ where $\tau_1 = M_1/W$. M_1 is the mass of coolant in the piping between the core outlet and IHX and W is the flow rate. Similarly, for the transport delay between the IHX and core inlet, the delay is $(t - \tau_2)$ where $\tau_2 = M_2/W$ and $M_1 + M_2$ equals the total NaK mass in the piping, heat exchangers, and parasitic load resistors.

4-15-64

7568-01036

Figure 13. Schematic of Temperature Losses and Transport Delays

2. Temperature Losses

Effects of heat losses from the core vessel and piping were considered. The core vessel heat losses at 50% NaK flow were calculated and are as follows:

$\bar{T}_c$ (°F)	$f(\bar{T}_c)^*$ (°F)
70	0
195	0.0498
315	0.1346
350	0.1680
455	0.2970
525	0.4113
595	0.5534
700	0.8249
875	1.4722
1050	2.246
1195 to 2000	3.5

*Temperature drop at 50%
NaK flow

The temperature drop due to losses from the core vessel is

$$T_{L1} = \frac{f(\bar{T}_c)}{2W/W_r} \quad \dots (14)$$

where W/W_r is the fraction of rated flow.

Piping heat losses were combined into two groups: the first consisted of all losses between the core outlet and the IHX inlet, and the second between the IHX and the core inlet. The total piping heat losses were assumed to vary linearly between 70 and 1200°F. At 70°F, the piping heat loss was set equal to 0; and at 1200°F, 5.357°F (50% flow). The losses in each section of the primary loop were assumed to be proportional to the NaK mass in that section. Using the data in the previous table, a straight-line function was used for the temperature drop due to heat losses.

Therefore, the equation used for the total loss is:

$$T_{L_\ell} = \frac{0.00237 \bar{T}_c - 0.166}{W/W_r} . \quad \dots(15)$$

The total loop temperature loss, T_{L_ℓ} , is distributed as follows. The loss in the outlet to IHX piping is

$$T_{L_2} = \frac{M_1}{M_1 + M_2} T_{L_\ell} , \quad \dots(16)$$

and the loss from the IHX to the core inlet is

$$T_{L_3} = \frac{M_2}{M_1 + M_2} T_{L_\ell} , \quad \dots(17)$$

where M_1 is the NaK mass between the core outlet and IHX and M_2 is the NaK mass between the IHX and core inlet.

E. UPPER AND LOWER GRID PLATE EQUATIONS

The upper and lower grid equations used were

$$\dot{T}_{ug}(t) = \frac{K_{ug}}{M_{ug} C_{ug}} \left[\frac{A}{L} \right]_{ug} \left[T_{co}(t) - T_{ug}(t) \right] , \quad \dots(18)$$

$$\dot{T}_{lg}(t) = \frac{K_{lg}}{M_{lg} C_{lg}} \left[\frac{A}{L} \right]_{lg} \left[T_{cin}(t) - T_{lg}(t) \right] , \quad \dots(19)$$

where the notation is the same as that previously described, with the following additions

subscripts u = upper

l = lower

g = grid

1 = radial distance or heat transfer area as indicated

2 = axial distance or heat transfer area as indicated

A = area (ft²)

K = conductivity (Btu/sec-ft-°F)

L = length (ft²).

1. Mass

The mass of the upper grid was assumed to be 5.9 lb; the lower grid, 6.5 lb.

2. Specific Heat and Conductivity

The specific heat and conductivity of the upper and lower grid were calculated and are plotted in Figures 14 and 15. Since both the conductivity and specific heat increased with increasing temperature, a graph was made of the ratio of K/C for both the upper and lower grids (Figure 16). By using this technique, it was possible to assume that the K/C value for each grid remained constant. Since the data were calculated, not experimental, it was assumed that a constant value of K/C is sufficient for analysis purposes. The increased complexity of simulating the K/C function on the analog does not seem worthwhile at this time since these physical properties are not well known. The values of K/C chosen were 2.247×10^{-2} lb/sec-ft for the upper grid and 1.822×10^{-2} lb/sec-ft for the lower grid.

3. Heat Transfer Areas and Path Lengths

The values used in the analog calculation for heat transfer areas and path lengths were as follows:

$$\left[\frac{A}{L} \right]_{lg} = 121.6 \text{ ft} ,$$

$$\left[\frac{A}{L} \right]_{ug} = 135.6 \text{ ft} .$$

Figure 14. Heat Capacity of Grid Plates

4-21-64

7568-01033

Figure 15. Conductivity of Grid Plates

4-21-64

7568-01034

Figure 16. Ratio of Grid Plate Conductivity to Heat Capacity

4-21-64

7568-01035

NAA-SR-9646

F. AUXILIARY HEAT EXCHANGER (AHX) EQUATIONS

The AHX was simulated by a three-node counterflow model as shown in the following diagram.

The equations are:

$$\dot{T}_{pi}(t) = \frac{UA_{AHXi}}{(MC)_{pi}} (T_{sj} - T_{pi}) + \frac{WC_p}{(MC)_{pi}} (T_{pi-1} - T_{pi}) , \quad \dots (20)$$

$$\dot{T}_{sj}(t) = \frac{UA_{AHXi}}{(MC)_{si}} (T_{pi} - T_{sj}) + \frac{W}{W_r} \frac{W_{rs} C_s}{(MC)_{si}} (T_{s,j-1} - T_{sj}) , \dots (21)$$

where the subscripts are defined as follows:

p = primary NaK side of the auxiliary heat exchanger

s = secondary NaK side of the auxiliary heat exchanger

i = ith node of primary side of auxiliary heat exchanger

j = jth node of secondary side of auxiliary heat exchanger

r = reference value primary loop

rs = reference value AHX loop.

The constants were as follows:

$$(UA)_{AHXi} = 0.0162 \text{ Btu/sec-}^\circ\text{F}$$

$$(MC)_{pi} = 0.569 \text{ Btu/}^\circ\text{F}$$

$$(MC)_{si} = 0.811 \text{ Btu/}^\circ\text{F}$$

$$C_p = 0.21 \text{ Btu/lb-}^\circ\text{F}$$

$$C_s = 0.218 \text{ Btu/lb-}^\circ\text{F}$$

$$W_{rs} = 0.533 \text{ lb/sec (at 50\% primary NaK flow).}$$

The inlet NaK temperature on the secondary side of the auxiliary heat exchanger was held constant at 70°F during the startup problem.

G. INTERMEDIATE HEAT EXCHANGER (IHX) EQUATIONS

During the nuclear startup, the secondary side of the intermediate heat exchanger does not contain coolant. Consequently, the IHX was treated as a large thermal mass. The equations used were:

$$\dot{T}_{Ai} = \frac{(UA)_{IHX}}{(MC)_{IHX}} (T_{IHXi} - T_{Ai}) \quad , \quad \dots (22)$$

$$\dot{T}_{IHXi} = \frac{(UA)_{IHX}}{(MC)_{IHX}} (T_{Ai} - T_{IHXi}) + \frac{WC_c}{(MC)_c} (T_{IHXi-1} - T_{IHXi}) \quad , \quad \dots (23)$$

where

T_{Ai} = average IHX metal temperature in ith node

T_{IHXi} = average NaK temperature in ith node

UA_{IHX} = overall NaK-to-heat exchanger metal heat transfer coefficient.

When i-1 equals zero, the NaK temperature is that of the inlet coolant temperature going into the primary side of the IHX.

The constants used were

$$(MC)_c = 7.77 \text{ Btu/}^\circ\text{F}$$

$$(MC)_{AHX} = 55 \text{ Btu/}^\circ\text{F}$$

$$(UA)_{AHX} = 31.5 \text{ Btu/sec-}^\circ\text{F.}$$

H. HOT CHANNEL EQUATIONS

In order to determine maximum fuel temperatures and maximum rates of change of temperature, it was necessary to simulate a channel that would have the highest value of temperature and/or rate of change of temperature. The equations for this hot channel are similar to the core thermal representation described in Sections B and C of Appendix II.

However, the constants used reflect the conditions which lead to higher temperatures and rates of change of temperature than the average channel described previously.

The equations used for the hot channel simulation are as follows:

$$\dot{T}_{fhi} = \frac{1.436G_i}{M_{fhi}C_{fh}} N - \frac{UA_{hi}}{M_{fhi}C_{fh}} (T_{fhi} - T_{chi}) , \quad \dots (24)$$

$$\dot{T}_{fhe} = \frac{1.436G_e}{M_{fhe}C_{fh}} N - \frac{UA_{he}}{M_{fhe}C_{fh}} (T_{fhe} - T_{cin}) , \quad \dots (25)$$

$$\dot{T}_{chi} = \frac{UA_{hi}}{M_{chi}C_c} (T_{fhi} - T_{chi}) - \frac{1.134W}{M_{chi}} (T_{chi} - T_{ch,i-1}) , \quad \dots (26)$$

where the subscript h refers to the hot channel. The values of the constants used are the same as the average channel with the following exception.

The 1.436 value is the product of the peak-to-average radial power distribution (1.33) and an allowance for errors in prediction of this peak-to-average ratio (1.08). The total allowance is then (1.08)(1.33) = 1.436. However, the peak-to-average flow ratio for the hot channel is about 1.26. Allowing a 10% error for the analytical prediction, the flow in the hot channel is (1.26)(0.9) = 1.134 of the average flow. The fuel-to-coolant heat transfer coefficient for the hot channel is shown in Figure 12. The assumption made to determine UA was that the fuel rod is centered within the ceramic, thus reducing fuel to coolant heat transfer at low temperatures. This results in higher fuel temperatures as well as higher rates of change of fuel temperature.

BLANK

APPENDIX III

ANALOG COMPUTER DIAGRAMS

This appendix contains (a) a list which summarizes the equations used in the simulation of startup and (b) a list of all the analog circuits used in the simulation. Analog computer circuitry diagrams and symbols are shown in Figures 17 through 25.

BLANK

SUMMARY OF SYSTEM EQUATIONS

1.
$$\frac{d}{dt} N = \frac{\beta}{\ell} \left(\rho_T N - N + \sum_{i=1}^6 \frac{\beta_i}{\beta} C_i \right)$$
2.
$$\frac{d}{dt} C_i = \lambda_i N - \lambda_i C_i$$
3.
$$\rho_T = \rho_D + \alpha_{ug} \Delta T_{ug} + \alpha_{lg} \Delta T_{lg} + \alpha_f \Delta \bar{T}_f$$
4.
$$\bar{T}_f = \frac{T_{fe} + \sum_{i=1}^5 T_{fi}}{6}$$
5.
$$\frac{d}{dt} T_{ug} = \left(\frac{K}{MC} \right)_{ug} \left(\frac{A}{\ell} \right)_{ug} (T_{c5} - T_{ug})$$
6.
$$\frac{d}{dt} T_{lg} = \left(\frac{K}{MC} \right)_{lg} \left(\frac{A}{\ell} \right)_{lg} (T_{cin} - T_{lg})$$

Average Channel

7.
$$\frac{d}{dt} T_{fi} = \frac{G_i}{M_{fi} C_f} N - \frac{UA_i}{M_{fi} C_f} (T_{fi} - T_{ci})$$
8.
$$\frac{d}{dt} T_{fe} = \frac{G_e}{M_{fe} C_f} N - \frac{UA_e}{M_{fe} C_f} (T_{fe} - T_{cin})$$
9.
$$\frac{d}{dt} T_{ci} = \frac{UA_i}{M_{ci} C_c} (T_{fi} - T_{ci}) - \frac{W}{M_{ci}} (T_{ci} - T_{c,i-1})$$
10.
$$M_{ci} = 1.008 - 1.4 \times 10^{-4} \bar{T}_c$$

$$11. \quad C_f = 0.0906 + 6.16 \times 10^{-5} \bar{T}_f$$

$$12. \quad \bar{T}_c = \frac{T_{cin} + T_{c5}}{2}$$

Hot Channel

$$13. \quad \frac{d}{dt} T_{fhi} = \frac{1.437 G_i}{M_{fhi} C_{fh}} N - \frac{UA_{hi}}{M_{fhi} C_{fh}} (T_{fhi} - T_{chi})$$

$$14. \quad \frac{d}{dt} T_{fhe} = \frac{1.437 G_e}{M_{fhe} C_{fh}} N - \frac{UA_{he}}{M_{fhe} C_{fh}} (T_{fhe} - T_{cin})$$

$$15. \quad \frac{d}{dt} T_{chi} = \frac{UA_{hi}}{M_{chi} C_c} (T_{fhi} - T_{chi}) - \frac{1.134 W}{M_{chi}} (T_{chi} - T_{chi-1})$$

$$16. \quad M_{chi} = 1.008 - 1.4 \times 10^{-4} \bar{T}_{ch}$$

$$17. \quad C_{fh} = 0.0906 + 6.16 \times 10^{-5} \bar{T}_{fh}$$

$$18. \quad \bar{T}_{ch} = \frac{T_{cin} + T_{ch5}}{2}$$

$$19. \quad \bar{T}_{fh} = \frac{T_{fhe} + \sum_1^5 T_{fhi}}{6}$$

Primary Loop

$$20. \quad (a) \quad W = \frac{W}{W_r} \cdot W_r$$

$$(b) \quad W_r(\text{at } 50\%) = 5.127 - 0.000578 T_{p3}$$

$$21. \quad T_{L1} = \frac{f(\bar{T}_c)}{W/W_r}$$

$$22. \quad T_{IHXin} = (T_{c5} - T_{L1})(t - \tau_1) - T_{L2}$$

$$23. \quad T_{cin} = (T_{p3} - T_{L3})(t - \tau_2)$$

$$24. \quad T_{Ll} = \frac{(0.00474\bar{T}_c - 0.332)}{W/W_r} \quad , \text{ for } W/W_r = 50\%$$

$$25. \quad T_{L2} = \frac{M_1}{M_1 + M_2} T_{Ll}$$

$$26. \quad T_{L3} = \frac{M_2}{M_1 - M_2} T_{Ll}$$

$$27. \quad UA_i = f(\bar{T}_f)$$

Auxiliary Heat Exchanger

$$28. \quad \frac{d}{dt} T_{pi} = \frac{UA_{AHXi}}{(MC)_{pi}} (T_{sj} - T_{pi}) + \frac{WC_p}{(MC)_{pi}} (T_{pi-1} - T_{pi})$$

$$29. \quad \frac{d}{dt} T_{sj} = \frac{UA_{AHXi}}{(MC)_{si}} (T_{pi} - T_{sj}) + \frac{W}{W_r} \cdot \frac{W_{rs} C_s}{(MC)_{si}} (T_{sj-1} - T_{sj})$$

Intermediate Heat Exchanger

$$30. \quad \frac{d}{dt} T_{ai} = \frac{(UA)_{IHX}}{(MC)_{IHX}} (T_{IHXi} - T_{Ai})$$

$$31. \quad \frac{d}{dt} T_{IHXi} = \frac{(UA)_{IHX}}{(MC)_c} (T_{ai} - T_{IHXi}) + \frac{WC_c}{(MC)_c} (T_{IHXi-1} - T_{IHXi})$$

NAME	SYMBOL	MATHEMATICAL REPRESENTATION
1. AMPLIFIER		$E_0 = -(E_1 + 10E_2)$
2. INTEGRATOR		$E_0 = -\int (E_1 + 10E_2) dt$
3. POTENTIOMETER (POT)		$E_0 = aE_1$ $a = \text{CONSTANT } 0 < a < 1.0$
4. SERVOMECHANISM (MULTIPLICATION)		$E_0 = E_1 E_2 / 100; E_1, E_2 \pm$
	$E_0 = E_1 E_2 / 100; E_2 \pm E_1 + \text{ONLY}$
	(DIVISION)
	$E_0 = -100 \left(\frac{2E_1}{10E_2} \right); E_1 \pm, E_2 + \text{ONLY}$
5. ELECTRONIC MULTIPLIER		$E_0 = -E_1 E_2 / 100$
6. HIGH GAIN AMPLIFIER		$E_0 = -10^6 (\text{approx.}) E_1$ NEVER USED AS REPRESENTED HERE; ALWAYS HAS FEEDBACK PATH FROM OUTPUT TO INPUT.
7. INITIAL CONDITION ON INTEGRATOR		$E_0, \text{ AT } t = 0, = -(100)a$ $a = \text{CONSTANT } 0 < a < 1.0$

3-30-64

7568-0986

Figure 17. Analog Computer Circuitry Symbols

4-15-64

7568-01040

Figure 18. Kinetics and Reactivity Feedback Simulation

7568-01042

Figure 19. Typical Fuel and Coolant Node Simulation

Figure 20. Transport Delay and Primary Loop Heat Loss Simulation

4-15-64

7568-01038

Figure 21. Auxiliary Heat Exchanger Simulation

CONSOLE C

4-15-64

7568-01037

Figure 22. Intermediate Heat Exchanger (IHX) Simulation

4-15-64

7568-01041

Figure 23. Controller Simulation

BLANK

CONSOLE B POTENTIOMETER SETTINGS

P - Series	Set	Potentiometer Number	Set	Q - Series	P - Series	Set	Potentiometer Number	Set	Q - Series
Description				Description	Description				Description
$10\lambda_1$	1240	00	1240	$10\lambda_1$	$20 \Delta \frac{W}{W_r} \times \frac{1}{t_{\text{ramp}}}$	-	25	-	-
$10\lambda_2$	3050	01	3050	$10\lambda_2$	-	-	26	8055	$\frac{100 n}{20 \times 19.66 \times 10^{-1} M_1}$
$10C_1$	-	02	-	$10C_2$	-	-	27	2192	$\frac{G_3}{18 M_{f_i}}$
$10C_3$	-	03	-	$10C_4$	-	-	29	0930	$\frac{G_e}{18 M_{f_e}}$
$10C_5$	-	04	-	$10C_6$	$\frac{1}{7.5 C_c}$	6380	30	-	-
λ_3	1110	05	1110	λ_3	$\frac{100 n}{15 \times 19.66 \times 10^{-1} M_2}$	8454	32	1901	$\frac{G_4}{18 M_{f_i}}$
λ_4	3010	06	3010	λ_4	biased readout	6250	33	-	-
$20 \frac{\beta_1}{\beta}$	6600	07	-	limit A52, flow ramp	controller adjustment	0514	34	6250	biased readout
$2 \frac{\beta_3}{\beta}$	3920	08	7900	$2 \frac{\beta_4}{\beta}$	-	6000	35	0474	$10^{-1} \times 0.474$
$2 \frac{\beta_5}{\beta}$	2300	09	8380	$20 \frac{\beta_6}{\beta}$	-	-	36	0017	$10^{-1} \times 0.0166$
$0.5 \lambda_5$	5650	10	5650	$0.5 \lambda_5$	-	-	37	1033	$\frac{G_5}{18 M_{f_i}}$
$0.2 \lambda_6$	6000	11	6000	$0.2 \lambda_6$	-	-	39	0350	$\frac{T_{\text{initial}}}{2000}, T_{\text{initial}} = 70^\circ\text{F}$
$400 \alpha_{\text{ug}} \left(\frac{\$}{^\circ\text{F}} \right)$	2400	12	-	balance A12 to zero with $\rho_D = \text{zero}$	IC A46 - ϕ	-	40	6780	$\frac{10}{14.745}$
$2 \frac{\beta_2}{\beta}$	4380	13	-	-	$\frac{T_{\text{UDB}}}{2000}$	6600	41	6653	$\frac{20}{9 M_{f_e}}$
$10 N_o$	-	15	4750	$\frac{\beta}{2000l}$	$\frac{T_{\text{LDB}}}{2000}$	6400	42	-	controller adjustment
$\frac{K}{MC} \left(\frac{A}{L} \right)_{\text{ug}}$	5164	16	5164	$\frac{K}{MC} \left(\frac{A}{L} \right)_{\text{ug}}$	-	-	43	0250	controller
$2 \frac{W}{W_r}$ at $t = 0$	-	17	0930	$\frac{G_1}{18 M_{f_i}}$	-	0500	44	6500	$\frac{T_{\text{MDB}}}{2000}$
$\frac{K}{MC} \frac{A}{L} l_g$	3409	18	3409	$\frac{K}{MC} \frac{A}{L} l_g$	IC A45 controller	-	45	0667	$\frac{1}{\tau_M}$
$400 \alpha_{lg} \left(\frac{\$}{^\circ\text{F}} \right)$	1600	20	5040	$\frac{19.66}{200} W r_{T=70^\circ, 50\% \text{ flow}}$	-	5000	46	0667	$\frac{1}{\tau_M}$
10v to kinetic IC pots	1000	21	2276	$19.66 \times 20 \times 0.000578$	controller	-	47	-	step amplitude
-	-	22	1829	$\frac{G_2}{18 M_{f_i}}$	first step rate	-	48	5000	controller
$\frac{200}{9 M_{f_i}}$	6653	23	-	-	second step rate	-	49	5000	controller
-	6000	24	5000	-	long-term control step rate	-

Figure 24. Console B Simulation

CONSOLE C POTENTIOMETER SETTINGS

P - Series	Set	Potentiometer Number	Set	Q - Series	P - Series	Set	Potentiometer Number	Set	Q - Series
Description				Description	Description				Description
-	-	02	2405	$\frac{1.437 G_1}{10 M_{f_i}}$	-	-	32	4913	$\frac{1.437 G_4}{10 M_{f_i}}$
3 x 0.0906	2718	05	3696	3 x 20 x 0.00616	$\frac{10 UA_i}{(MC)_{P_i}} \times \frac{T_{s_o}}{2000}$	0100	34	2847	$\frac{10 UA_i}{(MC)_{P_i}}$
0.5 x 1.008	5040	06	2800	100 x 20 x 0.00014	$\frac{10 UA_i}{(MC)_{P_i}}$	2847	35	2847	$\frac{10 UA_i}{(MC)_{P_i}}$
$\frac{100 C_c}{19.66 (MC)_c}$	1368	07	1368	$\frac{100 C_c}{19.66 (MC)_c}$	$\frac{100 C_c}{19.66 (MC)_c}$	1368	36	1368	$\frac{100 C_c}{19.66 (MC)_c}$
$\frac{(UA)_B}{10 (MC)_c}$	4054	08	7500	$\frac{75}{100}$	$\frac{10 UA_i}{(MC)_{P_i}}$	2847	37	2847	$\frac{10 UA_i}{(MC)_{P_i}}$
$\frac{M_1}{10(M_1 + M_2)}$	0260	09	1500	$(10^{-2} \times 3 \times 5)$	$\frac{10 UA_i}{(MC)_{P_i}}$	-	38	2405	$\frac{1.437 G_e}{10 M_{f_e}}$
-	-	12	4729	$\frac{1.437 G_2}{10 M_{f_i}}$	-	-	39	1877	$\frac{10 C_p}{19.66 (MC)_{P_i}}$
$\frac{20}{M_{f_i} M_1}$	5988	14	6380	$\frac{1}{7.5 C_c}$	$\frac{10 C_p}{19.66 (MC)_{P_i}}$	1877	40	-	-
$\frac{(UA)_B}{10 (MC)_c}$	4054	15	-	-	$\frac{M_2}{10(M_1 + M_2)}$	0740	41	1998	$\frac{10 UA_i}{(MC)_{s_i}}$
$\frac{(UA)_B}{(MC)_B}$	5727	16	-	-	$\frac{10 UA_i}{(MC)_{s_i}}$	1998	42	1998	$\frac{10 UA_i}{(MC)_{s_i}}$
$\frac{(UA)_B}{(MC)_B}$	5727	17	-	-	$\frac{10 UA_i}{(MC)_{s_i}}$	1998	44	-	-
-	6000	19	-	-	$1.134 \times \frac{10}{14.745}$	7690	46	1877	$\frac{10 C_p}{19.66 (MC)_{P_i}}$
-	-	22	5668	$\frac{1.437 G_3}{10 M_{f_i}}$	$\frac{10 C_p}{19.66 (MC)_{P_i}}$	1877	47	1430	$\frac{C_s W r_s \text{ (at 50\% flow)}}{(MC)_{s_i}}$
$\frac{2}{M_{f_e}}$	5988	24	-	-	$\frac{C_s W r_s \text{ (at 50\% flow)}}{(MC)_{s_i}}$	1430	48	1877	$\frac{10 C_p}{19.66 (MC)_{P_i}}$
$\frac{100 C_c}{19.66 (MC)_c}$	1368	25	1368	$\frac{100 C_c}{19.66 (MC)_c}$	$\frac{10 C_p}{19.66 (MC)_{P_i}}$	1877	49	0500	$\frac{C_s W r_s \text{ (at 50\% flow)} \times \frac{T_{s_o}}{200}}{(MC)_{s_i}}$
$\frac{(UA)_B}{10 (MC)_c}$	4054	26	-	-	$\frac{1.437 G_5}{10 M_{f_i}}$	-			
$\frac{(UA)_B}{(MC)_B}$	5727	27	-	-					
-	-	28	2671

Figure 25. Console C Simulation

BLANK

APPENDIX IV

SYMBOLS AND SUBSCRIPTS

SYMBOLS

A = area (ft ²)	ln = natural (napierian) logarithm
A = amplifier	M = mass (lb)
A = analog console "A"	n = neutron level
AHX = auxiliary heat exchanger	O = a constant based on initial conditions
B = analog console "B"	P = power (kw)
C = delayed neutron precursor concentration	PCS = power conversion system
C = specific heat at constant pressure (Btu/lb-°F)	PLR = parasitic load resistor
C = analog console "C"	Q = heat flow rate (Btu/sec)
D = diameter (ft)	T = temperature (°F)
f = fraction factor	$\dot{T}$ = time rate of change of temperature (°F/sec)
FG = function generator	t = time (seconds)
G = heat generation rate (Btu/sec)	t = thickness (ft)
g = acceleration due to gravity (ft/sec ²)	U = overall heat transfer coefficient (Btu/sec-ft ² -°F)
H = heat of vaporization (Btu/lb)	UDB = upper deadband setting (°F)
h = heat transfer coefficient (Btu/sec-ft ² -°F)	V = volume (ft ³)
IC = initial condition on analog integration	w = flow rate (lb/sec)
IHX = intermediate heat exchanger	α = temperature coefficient of reactivity (\$/°F)
K = thermal conductivity (Btu/sec-ft-°F)	β = effective delayed neutron fraction
k = multiplication factor	δ or Δ = change of variable from initial value (used with symbol for variable)
L = length (ft)	λ = decay constant for delayed neutrons (seconds ⁻¹)
LDB = lower deadband setting (°F)	ρ = reactivity (dollars)
(LMTD) = log mean temperature difference	τ = time constant or delay time (seconds)
ℓ = effective mean prompt neutron lifetime	τ = period (seconds)

BLANK

SUBSCRIPTS

A = average IHX metal temperature	j = j th auxiliary heat exchanger node
AHX = auxiliary heat exchanger	IHX = intermediate (NaK-Hg) heat exchanger
AHX ₁ = 1 th node of auxiliary heat exchanger simulation	IHX ₁ = 1 th node of intermediate heat exchanger simulation
b = boiler	g = lower grid plate
c = reactor coolant (NaK)	h = hot channel
c ₁ = 1 th coolant node	Ll = heat loss in primary loop
clad = fuel element cladding	max = peak or maximum value during transient
core = reactor core	o = outlet
D = control drums	p = primary side of heat exchanger
e = entrance node (core heat transfer)	r = reference
f = reactor fuel	s = secondary side of heat exchanger
fe = entrance fuel node	T = total
fi = 1 th fuel node	ug = upper grid plate
i = 1 th delayed neutron precursor group	0 = (zero) initial value
i = 1 th heat transfer node	1 = first core heat transfer node

BLANK

APPENDIX V NOMENCLATURE

A_{fe} = heat transfer area for the entrance fuel node (ft ²)	IHX = intermediate (NaK-Hg) heat exchanger	$T_{\ell 1}$ = temperature drop due to vessel heat losses	$\left \dot{\bar{T}}_f \right _{\max}$ = maximum rate of change of core average fuel temperature (°F/min)
A_1 = heat transfer area between fuel and coolant (outside surface area of fuel cladding) for the 1 th axial node (ft ²)	$K_{\ell g}$ = thermal conductivity of the lower grid plate (Btu/sec-ft-°F)	$T_{\ell 2}$ = temperature drop due to piping losses from core outlet to IHX	$\left \dot{T}_{f4} \right _{\max}$ = maximum rate of change of fuel temperature in the fourth axial node (°F/min)
AHX = auxiliary (NaK-NaK) heat exchanger	K_{ML} = thermal conductivity of liquid mercury (Btu/hr-ft-°F)	$T_{\ell 3}$ = temperature drop due to heat losses from IHX to core inlet	$\dot{T}_{f1}$ = time rate of change of fuel temperature in the 1 th axial node (°F/sec)
$\left(\frac{A}{L} \right)_{\ell g}$ = ratio of effective heat conduction area to effective heat conduction path length for lower grid plate (ft)	K_{ug} = thermal conductivity of the upper grid plate (Btu/sec-ft-°F)	T_{cin} = core inlet coolant temperature (°F)	$\dot{T}_{\ell g}$ = time rate of change of temperature of lower grid plate (°F/sec)
$\left(\frac{A}{L} \right)_{ug}$ = ratio of effective heat conduction area to effective heat conduction path length for upper grid plate (ft)	ℓ = effective prompt neutron lifetime (sec)	T_{c1} = coolant temperature in the 1 th axial node (°F)	$\dot{T}_{ug}$ = time rate of change of temperature of upper grid plate (°F/sec)
C_c = heat capacity of reactor coolant (Btu/lb-°F)	$\ln$ = natural (napierian) logarithm	$T_{c, 1-1}$ = temperature of the NaK coolant in the node previous to the 1 th node	U_{IHX} = overall heat transfer coefficient for the boiler section (Btu/sec-ft ² -°F)
C_f = heat capacity of fuel (Btu/lb-°F)	LDB = lower temperature control dead-band setting (°F)	$\bar{T}_f$ = core average fuel temperature (°F)	UDB = upper temperature control dead-band setting (°F)
C_1 = concentration of delayed neutron precursors in the 1 th group	M_{c1} = mass of coolant in the 1 th axial node (lb)	T_{fe} = fuel temperature in the entrance fuel node (°F)	U_{fc} = overall heat transfer coefficient from reactor fuel to coolant (Btu/sec-ft ² -°F)
C_{IHX} = heat capacity of metal and NaK in the intermediate heat exchanger (Btu/lb-°F)	M_{fe} = mass of fuel in the entrance fuel node (lb)	T_{f1} = fuel temperature in the 1 th axial node (°F)	W_r = NaK coolant flow rate (lb/sec) at 100% flow
$C_{\ell g}$ = heat capacity of lower grid plate (Btu/lb-°F)	M_1 = NaK mass from core outlet to IHX	$T_{\ell g}$ = temperature of the reactor lower grid plate (°F)	α_f = temperature coefficient of reactivity for the reactor fuel (dollars/°F)
C_{ug} = heat capacity of upper grid plate (Btu/lb-°F)	M_2 = NaK mass from IHX to core inlet	$T_{L\ell}$ = total temperature drop in primary NaK loop (°F)	$\alpha_{\ell g}$ = temperature coefficient of reactivity for the reactor lower grid plate (dollars/°F)
$\frac{d}{dt}$ = derivative with respect to real time (sec ⁻¹)	M_{f1} = mass of fuel in the 1 th axial node (lb)	T_{L1} = temperature loss due to reactor vessel heat loss	α_{ug} = temperature coefficient of reactivity for the reactor upper grid plate (dollars/°F)
$\frac{d}{d\tau}$ = derivative with respect to analog time (sec ⁻¹)	M_{IHX} = mass of NaK and metal in the intermediate heat exchanger (lb)	T_{L2} = temperature drop between core outlet and IHX (°F)	β = effective fraction of delayed neutron precursors
f = indication that one variable is a function, f , of another variable	$M_{\ell g}$ = mass of lower grid plate (lb)	T_{L3} = temperature drop between IHX and core inlet (°F)	β_1 = effective fraction of delayed neutron precursors in the 1 th group
FG = function generator	M_{ug} = mass of upper grid plate (lb)	T_{p1} = temperature of NaK primary side of IHX in 1 th node	$\delta k_e = k_e - 1$ = excess multiplication factor
G_e = heat generation rate in the entrance fuel node (Btu/sec)	n = neutron level	T_s = temperature of the NaK coolant as measured by the temperature sensor (°F)	λ_1 = decay constant for delayed neutrons in the 1 th group (1/sec)
G_1 = heat generation rate in the 1 th axial node (Btu/sec)	PCS = power conversion system	T_{sj} = temperature of NaK secondary side of IHX in j th node	ρ_D = drum reactivity (dollars)
G = heat generation rate in fuel	PLR = parasitic load resistor	T_{ug} = temperature of the reactor upper grid plate (°F)	ρ_T = total reactivity (dollars)
$= \sum_{i=1}^6 G_i$ (Btu/sec)	$P_{\max}$ = maximum power (kwt)	$\dot{T}$ = time rate of change of temperature (°F/sec)	τ_M = time constant of temperature sensor (sec)
h_{IHX} = convective heat transfer coefficient for NaK in the IHX (Btu/sec-ft ² -°F)	P_o = initial power (kwt)	$\left \dot{T}_{c5} \right _{\max}$ = maximum rate of change of core outlet NaK temperature (°F/min)	τ_1 = coolant transport delay time for core outlet to intermediate heat exchanger (sec)
	T_{A1} = average metal temperature in 1 th node of IHX	$\left \dot{T}_{cin} \right _{\max}$ = maximum rate of change of core inlet NaK temperature (°F/min)	τ_2 = coolant transport delay time for IHX reactor core inlet (sec)
	T_{c1} = temperature of coolant in the first axial node (°F)	T_{c1} = time rate of change of coolant temperature in the 1 th axial node (°F/sec)	
	T_{c5} = temperature of coolant in the fifth axial node (°F)		
	T_{c5max} = maximum core outlet NaK temperature (°F)		
	$T_{L\ell}$ = temperature loss in primary loop		

BLANK

-
- REFERENCES**
1. C. J. Code, Jr., "Analysis of the Response of the SNAP 8 Nuclear System During Startup of the Power Conversion System," NAA-SR-9626 (CRD), June 8, 1964
 2. L. R. Blue and M. Hoffman, "Generalized Program for the Numerical Solution of Space Independent Reactor Kinetics Equations," AMTD-131, January 1963
 3. D. G. Mason and R. W. Winson, "A Reactor Transient Heat Transfer Model," NAA-SR-7938, March 1, 1964
 4. Federal Regulations, Atomic Energy, Standards for Protection Against Radiation, 10CFR-20, October 15, 1962
 5. E. R. Cohen, "Some Topics in Reactor Kinetics," Proceedings of the Second International Conference on the Peaceful Uses of Atomic Energy (Geneva, Switzerland), September 1958, p 629