A DIGITAL COMPUTER PROGRAM FOR
THE EGCR TEMPERATURE AND NEUTRON
TRANSIENT ANALYSIS — ETNATA

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Experimental Gas-Cooled Reactor Operated By
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ABSTRACT

This report describes the use of a digital computer program written to solve the kinetic equations associated with the EGCR core. The neutron kinetic equations are solved with six groups of delayed neutrons. Temperature is calculated, based on the neutron level, and temperature feedback is used. Provisions are made to vary several operating parameters as a function of time. A scram routine is provided so that the reactor may be shut down realistically.

This program will be used for control studies, design of kinetic experiments, and accident analysis.

A DIGITAL COMPUTER PROGRAM FOR THE
EGCR TEMPERATURE AND NEUTRON TRANSIENT ANALYSIS

ETNATA

GENERAL

In evaluating the transient performance of the EGCR plant it is necessary to consider the time-dependent core temperature and neutron level and their effect on each other. In order to arrive at a more complete analysis of the plant, this digital computer program has been written.

One may question the use of a digital computer for this type of analysis since the analog computer has very successfully filled the role of transient analysis in the past. There are several reasons for the present choice of the digital computer:

1. For the type equations that are to be solved--that is, heat transfer and neutron kinetics where the solutions to the first order differential equations are exponentials--the digital computer does an admirable job with a minimum of worry about instability if a few simple rules are observed.
2. Nonlinear coefficients are easily handled on the digital computer.
3. It is very easy to put different data into the digital computer. This allows a multitude of separate cases to be run with a minimum of effort.
4. There is no time lost in scheduling the computer. The use of the analog computer requires that potentiometers be adjusted to represent the input conditions and coefficients of equations. When going from one

type of large problem to the next, many such potentiometers must be adjusted, and it is very easy to make mistakes during this process. Even though the plant complex is already patched for use on the analog computer, it may require several days to put the problem on the computer and get answers.

5. The capacity of the analog computer is taxed in problems as large as the EGCR if any detail is to be gained about the temperature structure in the core. Within reasonable bounds, the digital computer does not have this restriction. The core can be subdivided into very small increments if desired.
6. The digital computer is capable of providing more output data than the analog computer. Usually the analog computer is limited to recording data on a few recorders. The digital computer may be programmed to write out any variable as a function of time. The form of the output data may not be as convenient from the digital as the analog computer.
7. The digital computer program is easier for another person to use. The same program may be successfully shared by several persons for different uses.
8. For extremely fast transients, the frequency limitations of the analog computer may produce erroneous answers unless time scaling is used. This should not be a problem with the digital computer if short time intervals are used.

The digital computer program has limitations that the analog computer does not have.

1. It is not possible for one to get the feel of the plant by manipulating the controls.
2. It is not convenient to tune in controllers using digital techniques.

DESCRIPTION OF THE PROGRAM

Temperature Kinetics

A major portion of this program consists of solving the transient temperature equations in the EGCR core. The method of solution of these equations and the thermal model used are described in reference 2 and will not be repeated here.

Neutron Kinetics

Basically, this program solves the space-independent, time-dependent, neutron kinetics equations. An axial spatial dependence has been introduced through control rod position. Neutron diffusion codes yield flux distribution as a function of rod position. This information is used in this program to make a continuous change in the axial flux distribution as a function of rod position by linear interpolation between tabular values.

Only one fuel channel in the reactor is analyzed. This must be the average power channel so that the correct temperature feedback is obtained. If temperatures in other channels are desired, it is necessary to insert power as a function of time from this program in other existing programs for the calculation.

The neutron kinetics equations have as their input a k_{eff} that is derived from (1) rod worth, which is a function of depth of insertion; (2) doppler effect from fuel and moderator temperatures; and (3) any additional programmed reactivity effects, such as xenon or air.

The neutron kinetics equations that the program solves are the following:

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

$$\frac{dn}{dt} = \frac{\rho - \beta}{\ell} n + \sum_{i=1}^6 \lambda_i C_i + S \quad (2)$$

$$\frac{dC_i}{dt} = \frac{n\beta_i}{\ell} - \lambda_i C_i \quad (3)$$

Where:

k is the multiplication factor.

ρ is the reactivity.

n is the neutron level at a particular point.

ℓ is the effective prompt neutron lifetime.

β is the fraction of all neutrons in the reactor born from delayed neutron precursors.

β_i is the fraction of all neutrons in the reactor born from delayed neutron precursors of the ith kind.

C_i is proportional to the concentration of the ith kind of precursor.

λ_i is the decay constant of the ith kind of precursor.

S is the source term.

The following finite difference equations are solved by the computer:

$$n(t+\Delta t_1) = n(t) + \Delta t_1 \left[\left(\frac{\rho-\beta}{\ell} \right) n(t) + \sum_{i=1}^6 \lambda_i C_i(t) + S \right] \quad (2a)$$

$$C_i(t+\Delta t) = C_i(t) + \Delta t \left[\frac{\beta_i}{\ell} n(t+\Delta t) - \lambda_i C_i(t) \right] \quad (3a)$$

Where:

Δt is the time increment between successive calculations in the main program.

Δt_1 is the time increment between successive calculations of equation (2a).

The maximum time increment between successive calculations of equation 2a is determined in the program from the following:

$$\Delta t_1 \left[\frac{\rho-\beta}{\ell} \right] \lesssim 1/n \quad (2b)$$

To assure stability, n must be greater than one. The value of n is an input to the program, and is determined by the expected time constant of the transient. Typically, a value of five has been used for fast transients.

The temperature coefficient is introduced in the following manner (reference 3):

It is assumed that the average reactor k is given by

$$\bar{k} = \frac{\int_0^H k(z) Q^2(z) dz}{\int_0^H Q^2(z) dz} \quad (4)$$

The change in reactor k in going from temperature T_1 to T_2 is given by

$$\overline{\Delta k}_{T_1 \rightarrow T_2} = \frac{\int_0^H \Delta k_{T_1 \rightarrow T_2}(z) Q^2(z) dz}{\int_0^H Q^2(z) dz} \quad (4a)$$

Where:

$k(z)$ is the local reactor k_{eff} at axial location z .

$Q(z)$ is the power level at axial location z . For instance, power level could be $\frac{\text{Btu at } z \text{ per foot}}{\text{Average Btu per foot for the channel}}$ so that

$$\int_0^H Q(z) dz = 1.$$

H is the height of the reactor.

In this analysis it is desired to find the change in k produced by changing the axial temperature distribution of the core in an arbitrary manner.

The fuel and moderator temperature coefficients are tabulated as

$$\frac{1}{k} \frac{dk}{dT} = C(T) \quad (5)$$

Since the reactor k is hopefully always near unity while the temperature is changing, assume

$$\frac{dk}{dT} = C(T) \quad (6)$$

The local change in k produced in going from T_1 to T_2 is then

$$\Delta k_{T_1 \rightarrow T_2} = \int_{T_1}^{T_2} C(T) dT. \quad (7)$$

If this is substituted in equation 4a, the following is obtained:

$$\overline{\Delta k}_{T_1 \rightarrow T_2} = \frac{\int_0^H \int_{T_1}^{T_2} C(T) dT Q^2(z) dz}{\int_0^H Q^2(z) dz} \quad (8)$$

The equation that is actually solved by the computer is

$$\overline{\Delta k}_{T_1 \rightarrow T_2} = \frac{\sum_{i=1}^{INC} \left(\int_{T_1}^{T_2} C(T) dT \right) Q_i^2 \Delta z_i}{\sum_{i=1}^{INC} Q_i^2 \Delta z_i} \quad (9)$$

where INC is the number of axial increments in the analysis, and Δz_i is the fractional length of the i th increment.

Scram Control

Provisions are made in the program for testing the value of reactor outlet temperature and flux minus flow against operating limits. When one of these parameters exceeds the limit, a scram routine is called. This portion of the program inserts the control rods (after a predetermined time delay) in the reactor. The rate at which the rods are inserted is a function of rod height at scram. Manufacturers' test data (reference 1) were used to obtain rod velocities.

After the scram it is possible to allow the flow rate to change according to an arbitrary curve. The afterheat may also be programmed after the scram.

Parameter Changes That May Be Programmed

The code allows several parameters to be varied arbitrarily as a function of time during the course of execution.

1. The helium flow rate may be varied.
2. The reactor helium inlet temperature may be varied.
3. The reactivity may be changed, with control rods remaining fixed.
4. Control rods may be moved, thereby changing the reactivity.

Other parameters which may be programmed are:

1. Rod position versus cold reactor k_{eff} . This allows various rod patterns to be simulated by supplying the code with different curves of rod position versus reactor k_{eff} .
2. Power distribution versus control rod position. This distribution is normally obtained from neutron diffusion calculations using other codes. This program continuously varies the axial power distribution in response to a table of power distributions for various rod positions.

REFERENCES

1. Bigler, N. J., General Electric Company, letter to R. P. Teissier, August 7, 1963.
2. Keeton, D. C., "FATT: A Transient Temperature Computer Program for the EGCR Core," EGCR-2, February 25, 1964.
3. Preskitt, C. A., "Spatial Weighting Functions for Calculations of Reactivity," ORNL TM-725, November 1, 1963.

APPENDIX A

INPUT DATA FOR THE PROGRAM

DATA INPUT LIST

All input is on a field width of eight digits.

Fortran conversion is indicated following each card.

- I. Input that is normally not changed from case to case unless the number of axial increments is changed.

<u>Field</u>	<u>Variable Name</u>	<u>Definition and Use</u>
<u>1. Card 1. (9I8)</u>		
1-8	INC	The number of axial increments through the reactor. The number and lengths of increments may be chosen arbitrarily.
9-16	IT	The number of steady-state temperature iterations to establish a heat transfer coefficient. Four is a good number to use.
<u>2. Cards 2. (9E8.0)</u>		
	Z(K)	The length, in feet, of the K th reactor axial increment, starting at the bottom of the reactor. Use as many cards as necessary, with nine fields per card.
<u>3. Cards 3. (9E8.0)</u>		
	CL(K)	Correction to the fuel - gas heat transfer coefficient at the K th axial increment. This accounts for flow interrupters. Use as many cards as necessary.

<u>Field</u>	<u>Variable Name</u>	<u>Definition and Use</u>
	<u>4. Cards 4. (9E8.0)</u>	
	CS(K)	Same as CL(K) except for sleeve - fuel gas stream.
	<u>5. Cards 5. (9E8.0)</u>	
	G(L)	Geometric conductance between radial increment L and L+1. The product of this number with axial increment length in feet, and the conductivity of the material yields conductance. This includes 15 numbers.
	<u>6. Cards 6. (9E8.0)</u>	
	R(L)	Fraction of the total heat generated in a material, i.e., fuel, sleeve, or moderator, that is generated in the L th radial increment. For example, R(1) is the ratio of heat generated in the inner one-sixth of the fuel to that generated in all six radial fuel increments. Give 16 numbers.
	<u>7. Card 7. (9E8.0)</u>	
	CONST(K)	Decay constant of the K th kind of delayed neutron precursor.
	<u>8. Card 8. (9E8.0)</u>	
	BETA(K)	Fraction of all neutrons in the reactor which are born from the K th precursor.
	<u>9. Card 9. (9E8.0)</u>	
1-8	AGE	Prompt neutron lifetime
9-16	BFRA(K)	Total prompt neutron fraction. This should be:

$$1.0 - \sum_K \text{BETA}(K)$$

<u>Field</u>	<u>Variable Name</u>	<u>Definition and Use</u>
<u>10. Card 10. (9E8.0)</u>		
1-8	CCGC	Thermal conductance of the clad per foot of fuel length, Btu per second per °F.
9-16	CCP	k/WCp for clad
17-24	RCP	1.0/WCp for clad per foot of fuel length
25-32	CPG	Specific heat of helium. Btu per °F per pound.
33-40	QF	The average Btu per foot of fuel length per second generated in a fuel pin in an average power channel.
41-48	QS	The average Btu per foot of sleeve length per second generated in the average power channel.
49-56	QB	The average Btu per foot of moderator block length per second generated in the average power channel.
57-64	HEGP	Clad to fuel helium gap width in mils.

II. The following cards are stacked for multiple cases. Usually only these variables need changing from case to case.

11. Card 11. (12A6)

1-72	IDENT	Any identification for the case. The first field should be 1.
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12. Card 12. (9E8.0)

1-8	RNTM	Case run time in seconds of reactor operating time.
9-16	DELT	Time increment between successive calculations. A value of 0.15 for the cold reactor, and 0.2 for the hot reactor may be used. For fast transients a smaller value should be used to obtain better detail.

<u>Field</u>	<u>Variable Name</u>	<u>Definition and Use</u>
17-24	FRI	<p>For the critical reactor this is the initial power level. A value of 1.0 is used for the average power channel at rated power.</p> <p>For the subcritical reactor, FRI is the initial subcritical power level. Its value may be calculated using the relation</p> $FRI = \frac{(SOURCE) (AGE) (REACT)}{1.0 - REACT} = \frac{S \ell}{(1-k)/k} = \frac{S \ell}{-\rho}$
25-32	SOURCE	<p>This is the calculated source level for the EGCR using the above relation.</p> <p>For the critical reactor use <u>zero</u>.</p> <p>For the subcritical reactor a value of 0.25×10^6 has been used.</p> <p>Note that for the critical reactor the source <u>must be zero</u>, and for the subcritical reactor a value <u>must be specified</u>.</p>
33-40	WGII	Total initial He flow in the channel in lb per second. A value of about 0.5 is full power flow in the average power channel.
41-48	TGII	Initial He inlet temperature in °F.
<u>13. Card 13. (9E8.0)</u>		
1-8	RD POS	Rod position required to produce the desired reactor k, taking into account temperature and any extra reactivity.

<u>Field</u>	<u>Variable Name</u>	<u>Definition and Use</u>
9-16	RDDIFR	Difference in reactor k between room temperature and operating temperature tied up in withdrawable rods. This is the "hot rod effect."
17-24	RDDIFB	Difference in reactor k between room temperature and operating temperature tied up in fully inserted rods.
25-32	EREA	Initial extra Δk inserted into the reactor. For instance a value of $-.046$ is used for 20 Atm. of air.
33-40	VARI	A constant which determines the number of additional calculations of the flux term over that required for stability (n in equation 2b).

14. Card 14. (9E8.0)

1-8 PTIV A number to decide the method of advancing the print interval. A value of 0.0 causes a printout between equal time increments (PI below). A value of 1.0 gives a printout interval that advances with time.

As an example, if PI is 2.0 and PTIV is 0.0, a printout would occur at 2.0, 4.0, 6.0---nPI, seconds.

IF PI is 2.0 and PTIV is 1.0, a printout would occur at 2.0, 6.0, 12.0, 20.0, 30.0, -----

i-1

$PI \left(i + \sum_{i=1} n_i \right)$ seconds.

<u>Field</u>	<u>Variable Name</u>	<u>Definition and Use</u>
9-16	PI	Initial printout interval in seconds.
17-24	SCPI	Printout interval after a scram occurs (equally spaced).
25-32	RTAS	Case run time after scram in seconds of reactor operating time.
33-40	TTRP	Temperature of outlet gas at which a scram occurs, °F.
41-48	FFTRP	Flux minus flow trip point. The number to use here is based on the following formula:
$\left[\frac{\text{Fractional power level at which scram is desired}}{\text{scram is desired}} \right] - \left[\frac{\text{Flow at which scram is desired}}{\text{full power flow rate (WGS)}} \right] = \text{FFTRP}$		
Normally a value of 0.1 is used here.		
49-56	SCDEL	Scram delay in seconds. This is the delay to account for instrument lag between the time that scram conditions occur and the rods start to move.
57-64	WGS	Full power flow rate in the channel. A value of 0.5 is used for the average power channel.

15. Card 15. (918)

1-40	NTBL 1 to NTBL 5	Total number of entries in each of tables one to five.
41-48	NTBL 6	Number of control rod positions included in TBL 6.
49-64	NTBL 7 NTBL 8	Total number of entries in each of tables 7 and 8. Note that a number must be given here even though the entry is zero.

<u>Field</u>	<u>Variable Name</u>	<u>Definition and Use</u>
	<u>16. Cards 16. (9E8.0)</u>	
	TBL1(K)	This table causes the program to change the gas flow rate at a prescribed rate. The table alternates between a time and a flow change rate during which the first number is effective. The number following the time is the change rate in pounds per second. As many cards may be used as necessary, each card up to the last must be completely filled with nine numbers. The total entries must be equal to NTBLI on card 15. A change rate must be included for the maximum case run time even though it is zero.
	<u>17. Cards 17. (9E8.0)</u>	
	TBL2(K)	Same as above except for inlet temperature. The change rate is in °F per second.
	<u>18. Cards 18. (9E8.0)</u>	
	TBL3(K)	Same as cards 14 except for extra reactor k_{eff} . The change rate in this case is in Δk per second.
	<u>19. Cards 19. (9E8.0)</u>	
	TBL4(K)	Rod position versus reactor cold-clean k_{eff} . The first number is the rod withdrawal in inches. The next number is the reactor cold-clean k_{eff} . The table should start at 0.0 inches withdrawal.
	<u>20. Cards 20. (9E8.0)</u>	
	TBL5(K)	Same as TBL1 except time versus rod position change rate. The change rate is in inches per second.

<u>Field</u>	<u>Variable Name</u>	<u>Definition and Use</u>
<u>21. Cards 21. (9E8.0)</u>		
	TBL6(I,K)	This table contains the axial power distribution for various rod positions. The first field of the first card contains the rod position. This is followed by a list of power distributions (the ratio of Btu per foot at K, to the average Btu per foot for the entire channel). More than one card may be used for each rod position, but each new rod position must start on a new card. The total number of entries for each rod position is (INC +1). A value should be given for zero rod withdrawal.
<u>22. Cards 22. (9E8.0)</u>		
	TBL7(K)	This table produces a change in reactor flow rate when the rods start to move after a scram. The change rate is in pounds per second per second. This allows blower rampdown or coastdown after scram. The first field is the time after scram during which the change rate is effective. The next field is the change rate.
<u>23. Cards 23. (9E8.0)</u>		
	TBL8(K)	This table allows afterheat to be included after a scram. The first field should be 0.0. The second field is the reactor power level (in fractional parts of full power) at which the afterheat dominates. After these two fields, a power change rate is given. The first digit is the time after scram during which the change rate is effective, the next

<u>Field</u>	<u>Variable Name</u>	<u>Definition and Use</u>
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field is the change rate in fractional parts of full power per second.

The power distribution used for afterheat is that of the zero-inch rod withdrawal.

OUTPUT FROM THE PROGRAM

All output should be self-explanatory. The following points may be helpful:

1. All time units are in seconds in reactor time.
2. All temperatures are in °F.
3. The power level is in fractional parts of full power in the average power channel.
4. The flow is in pounds per second.
5. Axial location one is at the bottom of the reactor.

APPENDIX B

SIMPLIFIED FLOW DIAGRAM OF THE PROGRAM

MAIN PROGRAM

SUBPROGRAM