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FUEL MELTING TIME FOLLOWING
A LOSS OF COOLANT

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FUEL MELTING TIME FOLLOWING
A LOSS OF COOLANT

INTRODUCTION

A knowledge of the fuel temperature transient following a reactor flow-loss accident is important in reactor confinement studies, since the rate and time of fission product release depend upon this transient. Fuel melting times presented by Ekern, Muraoka, and Zaloudek¹ have been used in past confinement studies. These fuel melting times were based on the solid fuel element design, and the temperature transients were not presented. Furthermore, improved fission product decay heat data and reactor scram transients are now available. Therefore, the ALOC computer code was developed to calculate the fuel temperature transient following an accidental loss of coolant. This report presents fuel temperature transients for assumed accidents as calculated by this code. Operating instructions for the ALOC code are given in the appendix.

SUMMARY

The ALOC computer code has been developed to calculate the fuel temperature transient resulting from a loss of coolant to the Hanford Production Reactors. The code uses temperature dependent thermal properties. It handles both total reactor and single crossheader coolant loss cases. The aluminum fuel jacket can be assumed to either melt or form an uranium aluminum alloy.

The time required for the highest power fuel element in a 2000 kw process tube in a K Reactor to melt after a total loss of coolant was calculated as 2.75 hours, if the possibility of a graphite stored energy release is neglected. A graphite stored energy release may reduce the uranium melting time to about 2.0 hours. For a loss of coolant to a single crossheader, it was found that the uranium would not melt in a 2000 kw process tube in the K Reactors. The time required for fuel jacket and uranium melting is also presented as a function of tube power for a total coolant loss in a K Reactor.

The time for the uranium to melt in a 1350 kw D-Reactor process tube after a total loss of coolant is 8.25 hours. If graphite stored energy releases are considered, the uranium may melt in about 5.7 hours.

In order to further improve the program, data on coolant boilout times and graphite stored energy are required.

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DISCUSSION

Although a reactor shutdown usually results in a substantial reduction in the heat generation rate, the fission products continue to generate a substantial amount of heat for a considerable period of time. This heat must be continuously removed by some means in order to prevent the fuel temperature from increasing beyond a safe value.

When the coolant is lost to an operating reactor, the heat generation rate remains rather high even after the reactor scrams. If this heat is not removed, the fuel will eventually melt. The time at which the fuel element jacket and the uranium melt is important in reactor confinement and containment studies. Furthermore, the uranium temperature transient is of interest, since some fraction of the fission products are released prior to uranium melting. The uranium melting times for a loss of coolant accident were presented by Ekern, Muraoka, and Zaloudek¹ for the old solid fuel used in the Hanford Production Reactors. Since the reactors currently use internally and externally cooled fuel elements (I & E fuel) and the temperature transients were not presented in this previous study, the ALOC (Accidental Loss of Coolant) computer code was developed to calculate the temperature transient of the fuel and other reactor components following a loss of coolant. The ALOC code is written in Fortran IV for both the Univac-1107 and IBM-7090 computers.

Immediately following a flow loss to the operating reactor, the coolant remaining in the process tube will begin to increase in temperature up to its saturation temperature and a reactor scram will be initiated by a pannelit trip in about 0.2 second. In a short period of time, sufficient coolant will have been vaporized to expel the remaining coolant from the process tube. The rate at which this process proceeds depends upon the pre-accident operating tube power and the rate at which the reactor scrams. This boilout time was experimentally determined by Pound and Busselman² for the old solid fuel. Since data for I & E fuel are not available, the data for the solid fuel were used in this study.

After the coolant is completely expelled from the process tube, the fuel temperature increases up to the melting point of the aluminum jacket. The jacket will then slump down and occupy a portion of the annular gap between the fuel and the process tube. When uranium and aluminum are in contact at temperatures greater than 630 C, a diffusional process may occur forming an aluminum-uranium alloy. The melting point of the resulting alloy is greater than that of uranium. Therefore, the ALOC code allows the choice of assuming the fuel jacket either melts or forms an alloy with the uranium fuel. The process tube is assumed to remain centered and in place even if its melting temperature is exceeded. This is a conservative assumption, since, if the tube melts, the fuel would fall and make contact with the graphite, resulting in improved heat transfer between the fuel and graphite.

The heat generation used in the code consists of the two following parts:

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1. Heat generation from delayed neutron fissioning during the reactor scram.
2. Fission product decay heat.

When the reactor is at equilibrium operation, approximately 95% of the heat generated is released in the fuel, while during shutdown only about 86% of the heat is generated in the fuel.³ In order to remain conservative, the code assumes 95% of the heat is generated in the fuel during the entire transient. The heat generated from the delayed neutron fissioning, during the scram, is obtained from the Clumsy I computer code,⁴ a one-dimensional space dependent reactor kinetics code. The power excursion resulting from a total loss of coolant is also obtained from the Clumsy I code. The heat generation from the fission products is obtained from the RIBD computer code.

The effect of a possible graphite stored energy released is not included in the code due to insufficient data on the stored energy content, distribution, spectrum, and rate of release. The possible effect of a stored energy release is discussed qualitatively in a later section.

The possible heat generated by a water-aluminum reaction is neglected. Experimental investigations have shown the reactivity of water on aluminum is essentially negligible for temperatures less than 1150 C.⁵ The kinetic rate data for the uranium-steam reaction are quite poor having variations of over an order of magnitude.⁶ It is calculated that the heat of reaction for K Reactor fuel (using the highest kinetic rate data) would correspond to about 900 Btu/hr for a single fuel element. The heat generated in the highest power fuel element of a 2000 KW process tube is 2.57×10^5 Btu/hr at one hour after the scram. Therefore, neglecting the possible heat generated from the uranium-steam reaction is not serious for all but the lowest powered fuel pieces.

The mathematical model used to simulate the fuel temperature transient in the ALOC code consists of dividing the physical system into concentric circular nodes and solving the resulting differential heat balance equations by a forward difference technique. In the model only radial heat transfer is considered, assuming axial and angular heat flow to be negligible. The code allows the system to be divided into 14 uranium nodes, two cladding nodes, one process tube node, and 16 graphite nodes. A provision for a gas gap between the graphite tube and filler blocks is included. For the coolant loss cases to be discussed, the system was divided as follows:

<u>Node Number</u>	<u>Location</u>
1	Inner Fuel Jacket
2 - 5	Uranium Fuel
6	Outer Fuel Jacket
7	Process Tube
8 - 11	Graphite Tube Block
12 - 15	Graphite Filler Block

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The general differential heat balance equation describing the i th node is:

$$A_i \rho_i \frac{dT_i}{dt} = \frac{T_{i+1} - T_i}{R_{i+1/2}} + \frac{T_i - T_{i-1/2}}{R_{i-1/2}} + H_i$$

Where: A_i = Cross section area of node i
 C_i = Specific heat of material in node i
 H_i = Heat generation rate in node i
 R_i = Thermal resistance from the center of node i to the center of node $i+1$
 T_i = Temperature of node i
 ρ_i = Density of material in node i
 t = Time

The thermal resistance terms are a function of the thermo-physical properties of the materials in nodes i and $i+1$. The thermal resistance between two adjacent nodes of the same material is given by:

$$R_i = \left(\ln \frac{r_{i+1/2}}{r_{i-1/2}} \right) / 2\pi k_i$$

Where: $r_{i+1/2}$ = Radius at the volumetric center of node $i+1$
 $r_{i-1/2}$ = Radius at the volumetric center of node i
 k_i = Thermal conductivity of the material.

The thermal resistance between two adjacent nodes of different materials is:

$$R_i = \left(\ln \frac{r_{i+1/2}}{r_{i-1/2}} \right) / 2\pi k_i + \left(\ln \frac{r_{i+1/2}}{r_{i-1/2}} \right) / 2\pi k_{i+1} + R_c$$

Where: R_c = The contact resistance between the two materials

The thermal resistance between the outer cladding and the process tube is a series resistance of the following:

1. The thermal resistance of the outer half of the jacket, R_j
2. The thermal resistance of the steam gap between the fuel and the tube, R_g
3. The thermal resistance of the inner half of the process tube, R_t

The resistance of the steam gap is comprised of the following parallel combination:

1. The steam resistance, R_s

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2. The resistance of the supporting ribs, a series resistance of a contact resistance and the rib or support resistance, $R_c + R_{rib}$
3. Radiation resistance, R_{rad}

The overall resistance between the fuel and process tube is:

$$R = R_j + R_t + \frac{1}{\frac{1}{R_s} + \frac{1}{R_{rad}} + \frac{1}{R_c + R_{rib}}}$$

The thermal resistance between the process tube and the graphite is also a series - parallel path of the following:

1. Resistance of outer half of process tube, R_t
2. Resistance of gas gap, R_{gas} , in parallel with a radiative resistance, R_{rad}
3. Resistance of the outer half of the innermost graphite tube block node, R_g

The overall thermal resistance is:

$$R = R_t + R_g + \frac{1}{\frac{1}{R_{gas}} + \frac{1}{R_{rad}}}$$

A complex resistance path also occurs between the graphite tube and filler blocks. This thermal resistance term has the same form as for the resistance between the process tube and the graphite tube block.

The equation used for calculating the radiative resistance is:

$$\frac{1}{R_{rad}} = \frac{6.92 \times 10^{-9} A_1}{\frac{1}{\epsilon_1} + \frac{D_1}{D_2} \left(\frac{1}{\epsilon_2} - 1 \right)} \left(\frac{T_1 + T_2}{2} \right)^3$$

Where: A_1 = Area of surface 1
 D_1 = Outer diameter of surface 1
 D_2 = Inner diameter of surface 2
 T_1 = Temperature of surface 1, °R
 T_2 = Temperature of surface 2, °R
 ϵ_1 = Emmissivity of surface 1
 ϵ_2 = Emmissivity of surface 2

Temperature dependent material physical properties are used in the ALOC code.

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In order to simulate the loss of coolant to a single crossheader, the outermost graphite filler block node is assumed to increase to 950 C and then remain at 950 C due to heat removal by the process tubes above and below the stricken crossheader. This is a conservative assumption, since the heat generation rate is rather low by the time the graphite filler block temperature reaches 950 C even at the K Reactors. For a total reactor loss of coolant, the outermost graphite filler block is assumed to be thermally insulated.

The ALOC code has been used to investigate several loss of coolant accidents. The computer running time is quite long (40 minutes on the IBM-7090) for cases where the fuel does not melt in less than four hours. Therefore most of the cases studied were for K Reactor process tubes having powers greater than 1400 KW. Figure I presents the relative heat generation rate as a function of time after the loss of coolant for the K Reactors. If a total loss of coolant occurs, a power excursion results prior to the reactor scram. In order to simulate this, the fuel temperature computed by the Clumsy I code,⁴ at the time at which the power has returned to its pre-accident level, is used as the initial fuel temperature in the ALOC code. It was found that a 100 C to 150 C initial fuel temperature increase does not significantly change the uranium melting times, as long as the pre-accident tube power is below 2400 KW. The reason for this is that the heat capacity of the fuel is only about 2.0% of the total system heat capacity for the K Reactors.

The heat generation rate curve is represented in the program by the following equation:

$$H_{241} = a T^{-b}$$

Where a and b are constants over various time intervals, Table I presents the constants for describing the heat generation curves for B, D, and K Reactors.

The temperature transients of various components located at the center of a 2000 KW K Reactor tube following a total loss of coolant are shown in Figure II. The fuel jacket was assumed to melt in this case.

The time sequence of events is illustrated quite well in Figure II. During the first few seconds, while the reactor is scrambling, the heat generation rate is quite high and consequently the uranium fuel temperature increases quite rapidly up to the melting point of the aluminum jacket (660 C). Meanwhile the graphite tube block temperature increases while the graphite filler block temperature decreases. The process tube temperature increases rapidly and approaches the fuel temperature. After the fuel jacket melts, slumps down, and occupies a portion of the annular space between the fuel and process tube, the thermal resistance decreases, and the rate of uranium temperature increase is reduced. Eventually the process tube temperature increases above the graphite tube block temperature. Up to this point none of the heat generated in the fuel is transferred to the graphite, since the tube temperature is less than the

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TABLE I
CONSTANTS FOR HEAT GENERATION CURVE

K REACTOR			B, D REACTORS		
Time (Hours)	a	b	Time (Hours)	a	b
0-4.16x10 ⁻⁴	5.145x10 ⁻²	.363	5.91x10 ⁻⁴ -6.4x10 ⁻⁴	4.196x10 ⁻¹⁰	2.87
4.16x10 ⁻⁴ -4.995x10 ⁻⁴	2.185x10 ⁻⁵	1.36	6.4x10 ⁻⁴ -7.64x10 ⁻⁴	4.66x10 ⁻¹⁹	5.674
4.995x10 ⁻⁴ -5.83x10 ⁻⁴	7.98x10 ⁻¹³	3.615	7.64x10 ⁻⁴ -1.528x10 ⁻³	1.78x10 ⁻³	.6744
5.83x10 ⁻⁴ -6.25x10 ⁻⁴	1.092x10 ⁻²⁵	7.85	1.528x10 ⁻³ -3.08x10 ⁻²	7.31x10 ⁻³	.457
6.25x10 ⁻⁴ -7.5x10 ⁻⁴	3.400x10 ⁻⁶	1.509	3.08x10 ⁻² -1.25x10 ⁻¹	1.53x10 ⁻²	.225
7.5x10 ⁻⁴ -2.779x10 ⁻³	1.051x16 ⁻²	0.390	1.25x10 ⁻¹ -2.88x16 ⁻¹	1.479x10 ⁻²	.262
2.779x10 ⁻³ -2.779x10 ⁻²	6.780x10 ⁻³	0.465	2.88x10 ⁻¹	1.49x10 ⁻²	.255
2.779x10 ⁻² -2.779x10 ⁻¹	1.455x10 ⁻²	0.268			
2.779x10 ⁻¹	1.341x10 ⁻²	0.291			

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graphite temperature. As time progresses the heat generation rate decreases as shown in Figure I. After the tube temperature increases above the graphite temperature, the rate of increase of the fuel temperature is decreased due to part of the heat generated in the fuel being transferred to the graphite. By the time the fuel temperature has reached 1000 C, the heat generation rate is decreased sufficiently and the temperature difference between the process tube and graphite tube block is large enough to substantially reduce the fuel temperature transient. For this case the uranium melted (1133 C) at about 2.75 hours after the assumed coolant loss accident.

The ALOC code may be used to obtain the time for the fuel to melt at any position in the process tube by adjusting the heat generation rate to fit a cosine power distribution. Figure III is a plot of the fraction of uranium melted as a function of time for K Reactor process tubes following a total loss of coolant. The shape of these curves would be slightly altered if axial heat transfer was considered. However, the time required for 100% of the uranium to melt would be essentially that shown.

The uranium temperature transient for the highest powered fuel element in a K Reactor process tube is shown in Figure IV for various parameters of tube power. The high power fuel cases were run to find the approximate tube power at which the uranium would just start to melt before the large decrease in the uranium temperature transient. It was found that for a 2800 kw tube power, the uranium would melt in about 1.2 hours, whereas a 3000 kw tube would melt in about six minutes.

The uranium fuel temperature transient for a 2000 kw K Reactor tube, following a total loss of coolant, with the assumption that the fuel aluminum jacket completely alloys with the uranium is shown in Figure V. In this case the fuel would melt in about five minutes. However, the fuel jacket would be in the region of 630 C to 660 C for only about one to two seconds and it is calculated that only the innermost one to two mills of the jacket would probably alloy in this time period. Therefore, this situation is unlikely to be encountered in any loss of coolant accident. Furthermore, a movie of a fuel element heated up to simulate a loss of coolant accident, gave visual evidence that this problem of alloy formation is not encountered.

The fuel jacket and uranium melting times for the highest powered fuel element in a K Reactor process tube are shown in Figure VI as a function of tube power.

The loss of coolant to a single crossheader was simulated for a 2000 kw K Reactor tube. The ALOC code predicted that the jacket would melt in less than one minute, but the uranium would not melt. The uranium fuel temperature transient for this case is illustrated by Figure VII. Thus, for the current operating levels, the code predicts no uranium melting for a single crossheader loss of coolant.

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The fuel temperature transient, following a total loss of coolant at D Reactor, was simulated for tube powers of 800 kW and 1350 kW. The uranium fuel transient for the 1350 kW case is shown in Figure VIII. The 800 kW case was terminated after simulation of 2.25 hours, since the low rate of temperature increase would have required excessive computer time.

Since the ALOC code does not allow for a graphite stored energy release, a qualitative discussion of the effect of a stored energy released is warranted. If an appreciable amount of stored energy is released rapidly as the graphite temperature increases, following a loss of coolant, the uranium melting time may be substantially reduced. This depends not only on the total stored energy content and its temperature released spectrum, but also upon the time rate of energy release.

The last graphite stored energy determinations for the Hanford Production Reactors were performed in 1958. Furthermore, there has never been any stored energy measurements on K Reactor graphite. Since 1958 the stored energy content, its distribution and its spectrum has changed in all of the reactors. The stored energy has increased due to the additional irradiation, but increased graphite temperatures have annealed out part of the stored energy. The energy spectrum has shifted, since irradiation damage at higher irradiation temperatures is more difficult to release.

The total stored energy in D Reactor in 1958 was about 650 cal/gm right at the tube bore, and decreased to about 100 cal/gm in the filler block.⁶ Most of the stored energy in the filler block and in the outer regions of the tube block would probably not be released until graphite temperatures exceeding 1100 C are reached.

From the shape of the fuel and graphite temperature transients following a loss of coolant, it can be seen that if sufficient stored energy is released to increase the graphite temperature by 100 to 150 C, the time to uranium melting may be reduced to several minutes instead of several hours. Therefore, in order to estimate this effect the 1958 stored energy data were used for a D Reactor coolant loss case. The 1958 data predicts an average stored energy content in the central core regions of about 16 cal/gm for graphite temperatures less than about 1100 C. Neglecting the time rate of release, 16 cal/gm corresponds to a 46 C graphite temperature increase. For a 1350 kW tube in D Reactor the uranium would not melt until 8.25 hours following a coolant loss, if a stored energy release is neglected. In this case the graphite temperature lags the fuel temperature by 35 C. Therefore, it is assumed that a stored energy release would increase the graphite temperature by 50 C, and the uranium would melt in 5.7 hours.

Assuming that the stored energy distribution at the K Reactors is similar to that in the old reactors for graphite temperatures exceeding 350 C, it is estimated that the stored energy content is about 11.5 cal/gm for graphite temperatures below 1100 C. The release of 11.5 cal/gm would result in about a 28 C graphite temperature increase. If this energy is released in the order of minutes, then the hottest fuel element in a 2000 kW tube would melt in about two hours after the coolant loss compared to 2.75 hours if a stored energy release is neglected.

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In order to quantitatively include a stored energy release in the ALOC code, the total stored energy content, the stored energy distribution, the release spectrum, and the time rate of release for both the old and K Reactors are required. Furthermore, experimental coolant boilout tests for I & E fuel are required to check the value of two seconds which was assumed in the cases investigated.

The ALOC computer code is a useful tool in estimating the time required for the uranium fuel to melt following a loss of coolant to either the entire reactor or to a single crossheader.

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FIGURE I - Shutdown Heat Generation, K Reactor

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FIGURE IV
Fuel Temperature Transients
Following a Loss of Coolant

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FIGURE V

Fuel Temperature Transient for a 2000 KW K-Reactor Tube Following a Loss of Coolant (Uranium-Aluminum Alloy Formed)

Time After Coolant Loss (Minutes)

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TABLE II

TABLE OF NOMENCLATURE

A	= Cross sectional area
a	= Constant for relative heat generation curve
b	= Constant for relative heat generation curve
C_p	= Specific heat
D	= Diameter
H	= Heat generation rate
H_{rel}	= Heat generation relative to operating heat generation
R	= Thermal resistance
r	= Radius
T	= Temperature
ϵ	= Emmisivity
ρ	= Density
t	= Time

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APPENDIX I

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PHYSICAL PROPERTIES

The physical properties used in ALOC are given below:

I. Material Densities

Aluminum

$$\rho = 170 \frac{\text{lb}}{\text{ft}^3}$$

Graphite

$$\rho = 139.5 \frac{\text{lb}}{\text{ft}^3}$$

Uranium

The density and specific heat are combined (ρc_p) and the values used are reported in the specific heat section.

Zirconium

$$\rho = 410 \frac{\text{lb}}{\text{ft}^3}$$

II. Specific Heat

Aluminum

$$c_p = 0.2152 + 3.69 \times 10^{-5} T \frac{\text{BTU}}{\text{lb} \cdot ^\circ\text{F}} \quad T \leq 1215$$

$$c_p = 17.25 \frac{\text{BTU}}{\text{lb} \cdot ^\circ\text{F}}$$

$$1215^\circ\text{F} < T \leq 1225^\circ\text{F}$$

$$c_p = 0.25 \frac{\text{BTU}}{\text{lb} \cdot ^\circ\text{F}}$$

$$T > 1225^\circ\text{F}$$

Graphite

$$c_p = 0.332 - 7.1 \times 10^{-5} T + 1.15 \times 10^{-7} T^2 \frac{\text{BTU}}{\text{lb} \cdot ^\circ\text{F}}$$

Uranium

$$\rho c_p = 32.175 + 17.16 \frac{T}{1000} - 2.152 \left(\frac{T}{1000} \right)^2 + 0.35 \left(\frac{T}{1000} \right)^3 \frac{\text{BTU}}{\text{ft}^3 \cdot ^\circ\text{F}}$$

$$T \leq 1242^\circ\text{F}$$

$$\begin{aligned} C_p &= 616.81 \text{ BTU/ft}^3 \text{ F} \\ C_p &= 438 \text{ BTU/ft}^3 \text{ F} \\ C_p &= 990.1 \text{ BTU/ft}^3 \text{ F} \\ C_p &= 38.35 \text{ BTU/ft}^3 \text{ F} \\ C_p &= 4026.42 \text{ BTU/ft}^3 \text{ F} \\ C_p &= 43.85 \text{ BTU/ft}^3 \text{ F} \end{aligned}$$

$$\begin{aligned} 1242 < T \leq 1252 \text{ F} \\ 1252 < T \leq 1420 \text{ F} \\ 1420 < T \leq 1430 \text{ F} \\ 1430 < T \leq 2064 \text{ F} \\ 2064 < T \leq 2074 \text{ F} \\ T < 2074 \text{ F} \end{aligned}$$

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Zirconium

$$C_p = 0.085 \frac{\text{BTU}}{\text{lb} \cdot \text{F}}$$

III. Thermal Conductivities

Aluminum

$$\begin{aligned} k &= 119.5 + 0.0175 T \text{ BTU/ft}^2 \text{ H} \cdot \text{F} \quad T \leq 1220 \text{ F} \\ k &= -21.5 + 0.0635 T \text{ BTU/ft}^2 \text{ H} \cdot \text{F} \quad T > 1220 \text{ F} \end{aligned}$$

Graphite

$$k = 17.0 \text{ BTU/ft}^2 \text{ H} \cdot \text{F}$$

Uranium

$$k = 14.688 + 3.25 \frac{T}{1000} + 2.8125 \left(\frac{T}{1000} \right)^2 + 0.5209 \left(\frac{T}{1000} \right)^3 \text{ BTU/ft}^2 \text{ H} \cdot \text{F}$$

$$k = 30.0 \text{ BTU/ft}^2 \text{ H} \cdot \text{F} \quad T > 2074$$

Zirconium

$$k = 0.085 \text{ BTU/ft}^2 \text{ H} \cdot \text{F}$$

Gas

$$k = a_g + b_g T$$

Where a_g and b_g are constants depending on the gas composition.

APPENDIX II
INPUT DATA FOR ALOC

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Card 1

<u>Columns</u>	<u>Format</u>	<u>Information</u>
1 - 5	A5	Reactor Type
6 - 10	I5	Case Number

Card 2 Format 9I5

<u>Columns</u>	<u>Information</u>
1 - 5	Number of uranium nodes, max. = 14
6 - 10	Number of graphite tube block nodes; max. = 8
11 - 15	Number of graphite filler block nodes; max. = 8
16 - 20	Total number of graphite nodes; max. = 16
21 - 25	Printout frequency (number of cycles between printout)
26 - 30	1 - Not last case 2 - Last case
31 - 35	1 - Total coolant loss 2 - Single crossheader coolant loss
36 - 40	1 - Aluminum process tube 2 - Zirconium process tube
41 - 45	1 - Uranium - aluminum alloy formed 2 - Fuel jacket melts

Data format for all remaining cards is E10-4

Card 3

1 - 10	Fuel element ID inches
11 - 20	Fuel element OD inches
21 - 30	Tube ID inches
31 - 40	Tube OD inches
41 - 50	Graphite bore inches
51 - 60	Equivalent graphite tube block OD - inches
61 - 70	Equivalent graphite filler block OD - inches

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Card 4

Columns

1 - 80

Card 5

1 - 20

Card 6

1 - 80

Card 7

1 - 20

Card 8

1 - 80

Card 9

1 - 20

Card 10

1 - 10
11 - 20
21 - 30
31 - 40
41 - 50
51 - 60
61 - 70
71 - 80

Card 11

1 - 10
11 - 20
21 - 30
31 - 40

Information

First eight constants for heat generation curve; a

Last two a values

First eight constants for heat generation curve; b

Last two b values

First eight values of time for selection of a & b

Last two values of time for selection of a & b

Boilout time; hours

Maximum simulation time; hours

Initial , for iteration intervals; hours

Fraction of heat generated in graphite

Fuel Length; inches

Not used

Jacket thickness; inches

Power of fuel piece during equilibrium operation;

KW

Not used

Emmisivity of aluminum

Emmisivity of tube

Emmisivity of graphite

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Card 12

Column

1 - 10
11 - 20
21 - 30
31 - 40
41 - 50
51 - 60
61 - 70

Information

Graphite conductivity
Fuel jacket contact coefficient;
Contact area between graphite tube & filler blocks
Graphite - graphite contact coefficient;
Tube rib area per fuel element length; sq. ft.
Tube rib to fuel element contact coefficient;
Equivalent gas gap between tube and filler
blocks; inches

Card 13

1 - 10
11 - 20

1st constant for gas conductivity; ag
2nd constant for gas conductivity; bg

Card 14

1 - 10
11 - 20
21 - 30

Inner jacket temperature at boilout time, °F
Outer jacket temperature at boilout time, °F
Tube temperature at boilout time, °F

Card 15

1 - 80

Temperature of uranium nodes 1 - 8 at boilout
time, °F

Card 15 b

1 - 60

Temperature of uranium nodes 9 - 14 at boilout
time, °F

Card 16

1 - 80

Temperature of graphite nodes 1 - 8 at
boilout time, °F

Card 16 b

1 - 80

Temperature at graphite nodes 9 - 16 at
boilout time, °F

Card 17

1 - 10
11 - 20

Maximum outer jacket T per iteration, °F
Minimum outer jacket T per iteration, °F

Six control cards are required for use of the ALOC code on the Univac-1107. The program is stored on tape U0064. A sample case of input data is presented on the next page for a 2000 KW K Reactor process tube. A total loss of coolant was assumed for this case. If additional information on the ALOC code is desired, contact the author of this report.

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