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KINETICS OF THE LOS ALAMOS
POWER REACTOR EXPERIMENT (LAPRE)

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~~REACTORS - RESEARCH AND POWER~~

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ABSTRACT

A theoretical study of the kinetics of a model of the Los Alamos Power Reactor Experiment (LAPRE) has been made through integration of the dynamic equations with the IBM 701 computer. The stability is investigated under various conditions of power demand, rod-induced reactivity changes, and other conditions especially applicable to LAPRE. The results are given in graphical form, along with conclusions as to appropriate conditions of operation.

ACKNOWLEDGMENT

We wish to thank George I. Bell and P. J. Bendt for helpful discussions and criticisms.

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1. Introduction

The theory of reactor dynamics has been investigated by many workers. A recent review article by Goertzel¹ summarizes current knowledge and gives an extensive bibliography of the subject.

Quite often useful and satisfactory information can be obtained from the solutions of the linearized kinetic equations, or by means of various simplifying assumptions. This has usually, but by no means always, been the procedure for investigating kinetic behavior of a reactor. Analog computers, too, are quite often used in such studies.

In the case of LAPRE, however, we have felt that its somewhat novel design, in which gross expansion of fuel plays so great a part, deserved a complete numerical treatment. The availability of a high speed digital computer has, of course, entered into our decision.

The calculation described in this paper is designed to predict the dynamic behavior of LAPRE, a homogeneous power-prototype reactor. A detailed description of the reactor is given in LAMS-1611. Basically, LAPRE consists of an upright cylinder divided into three regions: a lower storage volume, maintained subcritical by neutron absorbers; a central reactive volume wherein power is generated; and an upper region in which vapor accumulates. The upper region is separated from the central region by a steel baffle, designed to direct the flow of circulating fuel. Between the central and lower regions are the coils of a heat exchanger, within which water is converted to supercritical steam.

Two programs of calculation are described. The first, Program I, is the more complete in terms of variations of the parameters which are taken into account. Program II is a simplified problem with more limited usefulness. Chronologically, the order is reversed, in that Program II was the first to be coded. We have, however, felt that the simplifications can be more easily seen if the more complex computation is given first.

2. Formulation of the Problem. Program I

We have, of necessity, based our calculations on a somewhat simplified model. The geometry which is most nearly approximated is that of a right circular cylinder, as indicated in Fig. 1. In this model the area of the cylinder is uniform; hence the volume is readily expressed in terms of the height. (Although heat exchanger coils are shown in the drawing, their volume should be ignored.)

The baffle is in fact thick enough to act as an effectively infinite tamper. Thus the criticality is affected by the proximity of the upper surface of the fuel to the baffle. As the upper surface of the fuel approaches the baffle, reactivity increases because of the increase in effective extrapolation length, which leads to a smaller buckling.

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Fig. 1. Idealized model of LAPRE. Arrows indicate the direction of fuel circulation.

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Fuel expands within the reactor as temperature increases in a manner which is determined by the initial filling. The fuel is a solution of UO_3 in excess H_3PO_4 . It is known that the ionic components of the fuel can be distilled differentially, with the result that the hydrogen to uranium ratio is changed as temperature varies.

There are other effects of varied importance. Those effects and parameters which have been taken into account in some approximation in the calculation are variations of:

1. Solution density with temperature
2. Hydrogen and oxygen content of solution with temperature and solution volume
3. Microscopic cross sections with neutron temperature
4. Buckling with filling and tamping conditions
5. Fermi age with density, neutron temperature, and hydrogen content of solution
6. Filling with time
7. Control rod withdrawal with time
8. Circulation of fuel.

One is interested in the parameters of power, neutron level, temperature, solution level, and power removal, as functions of time. Initial values of these parameters are, of course, specified as starting conditions.

We have used a space independent approximation, with modified Fermi age, to describe our cylindrical reactor. The kinetic equations are:

$$\frac{dN}{dt} = f(\nu \sum_f N_v + \alpha N_v + \Gamma \sum_i \gamma_i \lambda_i C_i) e^{-B^2 \tau} - \left(\frac{\lambda_t B^2}{3} + \Sigma_a + \Sigma_r \right) N_v$$

$$\frac{d\Gamma}{dt} = \frac{P - R}{C}$$

$$\frac{dC_i}{dt} = -\lambda_i C_i + \beta_i \sum_f N_v$$

$$P = E_f \sum_f N_v + p_o$$

where i runs from 1 to 6, corresponding to 6 delayed neutron emitters, and where:

- N = total number of thermal neutrons in reactor
 f = first flight correction

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ν	= number of neutrons per fission
Σ_f	= macroscopic fission cross section
v	= thermal neutron velocity
α	= coupling coefficient to reservoir as neutron source
Γ	= effectiveness of delayed neutrons relative to fission neutrons
C_i	= number of delayed neutron precursors in group i
γ_i	= circulation coefficient for C_i
λ_i	= decay constant of group C_i
B^2	= buckling
τ	= modified Fermi age
λ_t	= thermal transport mean free path
Σ_a	= macroscopic absorption cross section, including fission
Σ_r	= control rod effective macroscopic absorption cross section
T	= temperature
P	= power
R	= power removal
C	= appropriate heat capacity
β_i	= abundance of production of delayed neutron precursor C_i in fission
E_f	= energy liberated per fission
p_0	= equilibrium delayed β - γ power, taken equal to ϵP_0 , where P_0 is the initial power

The assumption is always made that for $t < 0$, the reactor is in equilibrium. One may then specify an initial period, $\mathcal{I}$, for the reactor, and calculate Σ_r . It should be noted that $\mathcal{I}$ is not the equilibrium reactor period as given by the inhour equation, but is a true transient period.

The starting conditions imposed on the system are usually represented by the discontinuity of some parameter at $t = 0$. Such discontinuities are, of course, not physically realizable, and one expects that the various excursions which result will tend to be exaggerated in the calculations as compared to those of the actual system.

The following table indicates the variations in the parameters which were taken into account in some degree of approximation as the physical conditions changed. The nature of these approximations is discussed below.

The constants need little discussion. f , the so-called first-flight correction, was taken as 0.88, the value given by Bendt² for LAPRE in his hydrogen slowing-down theory. The initial period of the reactor implicitly specifies Σ_r , which was calculated by the machine. α was

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Variation taken into account with variation of:

Parameter	No Variation	Solution Density	Water Concentration	Thermal Neutron Energy	Filling	Circulation	Values of Constants
f	X						0.88
ν	X						2.50
Σ_f		X		X			
v				X			2.2×10^5 cm/sec at 20°C
α	X						→ 0
Γ	X						1.1
γ_i						X	
λ_i	X						0.0125, 0.0315, 0.154, 0.457, 1.613, 13.89
B^2		X	X	X	X		
τ		X	X	X			
λ_t		X	X	X			
Σ_a		X	X	X			
Σ_r	X						Calculated by machine
C	X						1.5×10^5 joule (°C) ⁻¹
β_i	X						6.2×10^{-4} , 4.2×10^{-3} , 5.3×10^{-3} , 6.0×10^{-3} , 2.1×10^{-3} , 6.2×10^{-4}
E_f	X						2.96×10^{-11} joule (fission) ⁻¹
P_0	X						ϵP_0
ϵ	X						0.0448

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varied over wide limits in one problem and was finally set equal to zero, on the basis that it influenced the transient response of the reactor only slightly. C was taken as 1.5×10^5 joule $^{\circ}\text{C}^{-1}$, approximately the heat capacity of 40 liters of water. ϵ was estimated as 0.0448. The other constants were given the values shown in the table.

The variations of other parameters are described most easily by stating the actual relations used in the calculation. The subscript $_0$ denotes the condition at 20°C .

$$\Sigma_f = \Sigma_{f0} \chi(T) \frac{v_0}{v}$$

where

$$\chi = \frac{\rho(T)}{\rho_0}$$

is a quartic least squares fit to the observed variation of density with temperature.

$$v = v_0 \left(\frac{T}{T_0} \right)^{1/2}$$

The γ_i are intended to approximate the effect of circulation on delayed neutrons. For delayed neutron emitters with half lives short compared to circulation cycle time, $\gamma_i \rightarrow 1$, while for those of long half life, $\gamma_i \rightarrow \frac{V_R}{V_T}$ where V_R and V_T are the reacting and total fuel volumes.

$$B^2 = \left(\frac{\pi}{H + \delta_U + \delta_L} \right)^2 + \left(\frac{2.405}{R + \delta_L} \right)^2$$

where

$$\delta_L = 2.75\lambda_{tr};$$

and

$$\delta_U = 0.71\lambda_{tr}$$

for solution more than 10 cm below the top baffle;

$$\delta_U = 2.75\lambda_{tr}$$

for solution in contact with the top baffle; and

$$\delta_U = 0.71\lambda_{tr} + 0.204\lambda_{tr} (X - 10)$$

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in the intermediate range, where X is the distance from baffle to liquid surface in cm. The variation in transport mean free path with temperature and neutron energy is given below. λ_{tr} was taken to be the transport mean free path for thermal neutrons. This choice is conservative, in that the increase of extrapolation length with increased temperature is greatest in this case. The increased extrapolation length leads to decreased buckling, hence increased reactivity with increasing temperature. Thus this selection adds an effective positive component to the temperature coefficient of reactivity which is exaggerated with respect to that which will exist in LAPRE.

The liquid height is defined by the initial filling, H_0 , and the temperature as

$$H_L(T) = H_0 \lambda^{-1}$$

Then the height of the reacting volume

$$H = H_L - H_1$$

is given by subtraction of a constant, H_1 , corresponding to the height of the lower, poisoned volume.

In the calculation of τ and λ_{tr} , account was taken of the evaporation loss of water from the fuel solution, as well as the effects of changing density and neutron energy with temperature.

We write (after Bendt²):

$$\tau = \frac{\ell_n \left(\frac{T_1}{T} \right)}{3 \Sigma'_{tr} (\Sigma'_a + \xi \Sigma'_{sc}) + B^2}$$

where T_1 is chosen such that for $T = 293^\circ A$, $\ell_n \left(\frac{T_1}{T} \right) = 17$

$$\Sigma'_a = \sum_i \rho_i(T) \sigma'_{a_i}$$

$$\xi \Sigma'_{sc} = \sum_i \rho_i(T) \xi \sigma'_{sc_i}$$

$$\Sigma'_{tr} = (\lambda_{tr})^{-1} = \sum_i \rho_i(T) \sigma'_{tr_i}$$

where primes indicate that the cross sections are those appropriate to fission neutrons. Summations on i are made for the four components H, O, P, U.

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In agreement with qualitative experimental observations, it is assumed that the only component leaving the solution at elevated temperature is water. It is convenient to express the densities of the components of the fuel in term of molarity. Thus with the indices, H, O, P, U, and W (water), we have the empirical relation:

$$M_W(T) = M_W(20^\circ\text{C}) \text{ for } T < 100^\circ\text{C}$$

$$M_W(T) = M_W(20^\circ\text{C}) - 6.58 \times 10^{-7} \frac{V_v}{V_L} (T - 100)^3 \text{ for } T \geq 100^\circ\text{C}$$

where V_v and V_L are the vapor and liquid volumes at temperature T , respectively. The constant 6.58×10^{-7} is derived from measurements of the vapor pressure of water by J. R. Phillips.

Then, for a given concentration of H_3PO_4 , M_P , and of uranium, M_U , we assume:

$$M_P(T) = \text{const.}$$

$$M_U(T) = \text{const.}$$

$$M_H = 2 M_W + 3 M_P$$

$$M_O = M_W + 4 M_P$$

All absorption cross sections were taken to vary as $1/v$. All scattering cross sections were taken constant with energy except that of hydrogen, for which a linear variation with temperature (over a limited range) was assumed.

For both the fast and slow neutron absorption cross sections, the variation of hydrogen concentration with temperature was neglected as being a second order effect when uranium absorption is considered. Thus we had simply

$$\Sigma_a = \Sigma_{a_0} \chi$$

$$\Sigma'_a = \Sigma_{a_0} \chi$$

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2.1 Doppler Broadening

The effective broadening of resonance widths due to thermal motion of the fuel has been neglected in our calculations. Its exact treatment is, indeed, quite involved. Goertzel¹ has pointed out, however, that for the Brookhaven and Oak Ridge reactors, the Doppler broadening leads to a negative temperature coefficient of reactivity.

In LAPRE, the large H U ratio (~ 200) insures that the effect of Doppler broadening will be small, inasmuch as the ratio of uranium absorption to hydrogen slowing down cross sections is small. Furthermore, the increase in capture/fission ratio for uranium with increasing neutron energy leads to the conclusion that the effect gives a negative temperature coefficient of reactivity. The neglect of Doppler broadening is, therefore, probably justified.

2.2 Xenon Burnout

At the fluxes which are anticipated in LAPRE, $1 - 2 \times 10^{13}$ n/cm² sec, the steady state Xe poisoning will give a reduction in reactivity $\Delta k \cong -.03$.³ The production of Xe¹³⁵ through the Te¹³⁵ $\rightarrow$ I¹³⁵ chain is essentially governed by the I half life of 6.7 hr. Hence a surge of neutron flux will tend to burn up Xe¹³⁵ which is not immediately replenished, leading to autocatalysis, as is well known. An estimate of the time required to burn out half the poison at a flux of 10^{15} n/cm² sec, a factor of 100 over normal flux, gives a time of the order of 200 sec. Since the calculated excursions of the reactor are normally finished in times of the order of 10 sec, the neglect of Xe¹³⁵ burnout appears to be justified.

2.3 Method of Solution

The eight coupled differential equations were integrated by the Runge-Kutta method. The accuracy was checked by a modified Simpson's Rule integration, after every fourth time step. The time interval was regulated continually in accordance with the deviation of the Runge-Kutta and Simpson's Rule integrations.

2.4 Modification of the Calculations. Time Delay.

In a particular problem, that of start-up of the reactor, it was realized that the effect of circulation on the temperature in the upper and lower sections of LAPRE was not being taken into account to a sufficient extent. In this calculation, therefore, the expansion of the fuel was separated into two parts that of the upper region and that of the lower. We then write, for the height, H, of the critical region,

$$H = \frac{1}{\chi_U} (H_0 - \chi_L H_L)$$

where H_0 is the initial filling, H_L is the height of the lower region, and χ_U and χ_L are defined as functions of upper and lower region temperatures. The difference in the χ 's is that

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the temperature of the lower region is assumed to lag that of the upper region by 0.5 sec, i. e.,

$$T_L(t) = T_U(t - 0.5 \text{ sec})$$

This is, of course, a somewhat crude approximation, since the two regions will expand with rather different expansion functions when they are at different temperatures.

Under this procedure the feature of time-interval regulation in the calculation must be discarded. It is necessary to run at uniform time intervals, storing enough past temperature information so that $T(t - 0.5 \text{ sec})$ is always available. This was done by setting $\Delta t = 0.0025 \text{ sec}$ and storing 200 values of T .

The calculations done in this way are labelled Time Delay.

3. Simplified Problem. Program II

The preceding description is of the calculation in its final form. Problems of this magnitude normally evolve to some extent after their initial programming. Initially the program was considerably simpler in that no account was taken of the effects of loss of water from the fuel, variation of cross sections with neutron temperature, and the modifications of hydrogen slowing down theory (first flight correction, modified Fermi age). These corrections are, indeed, not essential if wide excursions of reactor temperature are not expected. One other simplification was made in that density variation with temperature was described by an exponential factor, i. e.,

$$\chi = \exp [-K(T - T_0)]$$

instead of by the much more accurate quartic polynomial fit. In the simplified program the value of K chosen was somewhat smaller than the appropriate value, since the exponential curve was made to fit the experimentally determined variation of density with temperature at 20°C and 430°C , instead of in the region of 430°C . K was too small by about a factor of 2. This does not, however, invalidate the calculation, since this choice gave a smaller temperature coefficient of reactivity than actually exists. In fact, the reactor will be more stable than is predicted by the calculations.

The only equation affected by these changes is that for N , in which v is taken constant, and, in our previous notation,

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$$\begin{aligned}f &= 1 \\ \Sigma_f &= \Sigma_{fo} X \\ \tau &= \tau_o X^{-2} \\ \lambda_t &= \lambda_{to} X^{-1} \\ \Sigma_a &= \Sigma_{ao} X,\end{aligned}$$

all other quantities being taken as constant with temperature.

3.1 Coding

The calculations were coded for the IBM 701 computer in the "dual" coding system.⁴ Dual coding supplies a floating decimal interpretation scheme in which the numerical calculations are carried out, thereby eliminating the problem of scaling. At the same time all of the flexibility of fixed point programming is available to the coder for address modification and control of the course of the problem. Running time for a problem ranged from 15 minutes to, in a few cases, one hour. Program I took almost exactly twice as long in terms of machine time as did Program II; one time step for Program II required about 1.5 sec of machine time.

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4. Results

4.1 Power Demand. Program II

$$P_0 = 10^2 \text{ w}$$

$$T_0 = 430^\circ\text{C}$$

$$R = 10^6 \text{ w}$$

$$\alpha = 0$$

$$\mathcal{T} = 10^{10} \text{ sec}$$

$$\gamma_i = 0.4 = (\nu)^{-1}, i = 1, 2, \dots, 5$$

$$\gamma_6 = 0.2$$

Variations of power and temperature are plotted versus time. Power withdrawn is suddenly increased from 100 w to 10^6 w.

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4.2 Power Demand. Program II

$$P_o = 10^2 \text{ w}$$

$$T_o = 430^\circ\text{C}$$

$$R = 10^6 \text{ w}$$

$$\alpha = 3 \times 10^{-2}$$

$$\mathcal{I} = 10^{10} \text{ sec}$$

$$\gamma_i = \text{as above}$$

$$\beta_i = 0 \text{ (no production of delayed neutrons)}$$

Variations of power and temperature are plotted versus time. These curves demonstrate the stabilizing influence of delayed neutrons quite effectively.

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4.3 Reactivity Increase. Program II

$$P_o = 10^6 \text{ w}$$

$$T_o = 430^\circ\text{C}$$

$$R = 10^6 \text{ w}$$

$$\alpha = 3 \times 10^{-2}$$

$$\mathcal{I} = 10^{-2} \text{ sec}$$

$$\gamma_i = \text{as above}$$

The situation shown here is that of sudden removal of a control rod. Note that at operating temperature the control rod acts as a thermostat, rather than as a level control.

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4.4 Reactivity Decrease. Program II.

$$P_o = 10^6 \text{ w}$$

$$T_o = 430^\circ\text{C}$$

$$R = 10^6 \text{ w}$$

$$\alpha = 3 \times 10^{-2}$$

$$\lambda = -10^{-2} \text{ sec}$$

$$\gamma_i = \text{as above}$$

Sudden insertion of a control rod is shown. Compare with problem 4.3.

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4.5 Demand Decrease. Program II.

$$P_o = 10^6 \text{ w}$$

$$T_o = 430^\circ\text{C}$$

$$R = 10^{-10} \text{ w}$$

$$\alpha = 3 \times 10^{-2}$$

$$\lambda = 10^{10} \text{ sec}$$

$$\gamma_i = \text{as above}$$

Sudden failure of feed water pump or circulating pump. It should be remembered that $\epsilon p_o = 4.5 \times 10^4 \text{ w}$ of power generation is due to delayed β - γ emitters. The remainder is due to continued multiplication of delayed neutrons.

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4.6 Simultaneous Demand and Reactivity Decrease. Program II.

$$P_o = 10^6 \text{ w}$$

$$T_o = 430^\circ\text{C}$$

$$R = 10^{-10} \text{ w}$$

$$\alpha = 3 \times 10^{-3}$$

$$\lambda = -10^{-2} \text{ sec}$$

$$\gamma_i = \text{as above}$$

This problem is essentially the same as 4.5 but with the added condition that reactivity is decreased somewhat by the insertion of control rod when power removal fails. Note that multiplication should be decreased to a considerably greater extent if delayed neutron power is to be made comparable to delayed β - γ power.

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4.7 Demand Decrease. Program II.

$$P_o = 2 \times 10^6 \text{ w}$$

$$T_o = 430^\circ\text{C}$$

$$R = 1 \times 10^6 \text{ w}$$

$$\alpha = 3 \times 10^{-2}$$

$$\mathcal{I} = 10^{10} \text{ sec}$$

$$\gamma_i = \text{as above}$$

In this case the power demand is reduced only by a factor of 2. Note that the reactor can follow such a transient quite well.

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4.8 Extreme Power Demand. Program II.

$$P_o = 10^{-3} \text{ w}$$

$$T_o = 430^{\circ}\text{C}$$

$$R = 10^6 \text{ w}$$

$$\alpha = 3 \times 10^{-2}$$

$$\tau = 10^{10} \text{ sec}$$

$$\gamma_i = \text{as above}$$

This calculation is intended merely to show the moderate overshoot even when power is increased a factor of 10^9 . This is not, of course, a practical case.

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4.9 Startup (no Time Delay). Program I

$$P_o = 10^5 \text{ w}$$

$$T_o = 20^\circ\text{C}$$

$$R = 10^{-10} \text{ w}$$

$$\alpha = 0$$

$$\mathcal{I} = 10 \text{ sec}$$

$$\gamma_i = 1, i = 1, 2, \dots, 6$$

Power is plotted on two graphs, the second showing the peak on an expanded time scale. The notch in the peak is a result of the increased reactivity as the upper surface of the fuel approaches the baffle.

The violence of this power surge is quite evidently intolerable; its existence is, however, due to a lack of refinement of the program and not to an eccentricity of the reactor. The fault lies in not allowing for the finite circulation time of the fuel. This has been done in the succeeding calculations. It must be emphasized that, for the reasons given above, calculation 4.9 is entirely unrealistic.

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4.10 Startup (Time Delay). Program I.

$$P_o = 10^6 \text{ w}$$

$$T_o = 20^\circ\text{C}$$

$$R = 10^{-10} \text{ w}$$

$$\alpha = 0$$

$$\mathcal{I} = 10 \text{ sec}$$

$$\gamma_i = 1$$

This problem is essentially that of 4.9, with the exception that circulation effects on temperature of the upper and lower sections of the reactor are now approximated. Note that initially the power starts to decay, on account of an inherent negative temperature coefficient of reactivity associated with the upper section. The main surge of power begins only after hot fuel from the upper section has been circulated into the lower sections (in 0.5 sec), causing expansion of additional uranium into the reactive region.

This calculation further illustrates that level safeties would have sufficient time to operate if they were set in the vicinity of 1.5 Mw on startup.

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4.11 Startup (Time Delay). Program I.

$$P_o = 10^5 \text{ w}$$

$$T_o = 20^\circ\text{C}$$

$$R = 10^{-10} \text{ w}$$

$$\alpha = 0$$

$$\mathcal{J} = 10 \text{ sec}$$

$$\gamma_i = 0.5 \quad i = 1, 2, 3, 4$$

$$\gamma_i = 1.0 \quad i = 5, 6$$

For comments see 4.10. The delayed neutron precursors are reduced here to simulate circulation effects. Also plotted is the height of fuel in the reactor region as a function of time.

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4.12 Startup (Time Delay), Fast Period, Program I.

$$P_o = 10^5 \text{ w}$$

$$T_o = 20^\circ\text{C}$$

$$R = 10^{-10} \text{ w}$$

$$\alpha = 0$$

$$\lambda = 10^{-2} \text{ sec}$$

$$\gamma_i = 0.5, i = 1, 2, 3, 4$$

$$\gamma_i = 1.0, i = 5, 6$$

The effect of a sudden increase in reactivity on startup is shown. No appreciable effect is observed over that shown in calculations 4.10 and 4.11.

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4.13 Startup (Time Delay) with Fuel Addition at the Rate of 3 liter/min. Program I.

$$P_0 = 10^5 \text{ w}$$

$$T_0 = 20^\circ\text{C}$$

$$R = 10^{-10} \text{ w}$$

$$\alpha = 0$$

$$\mathcal{L} = 10 \text{ sec}$$

$$\gamma_i = 0.5 \quad i = 1, 2, 3, 4$$

$$\gamma_i = 1.0 \quad i = 5, 6$$

The oscillatory behavior is presumably due to the existence of temperature coefficients of reactivity which are positive for the reactor in the overall sense, but negative in the reacting region. The rather rapid addition of reactivity permits excursions of power about a mean level which is coupled to the circulation cycle.

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4.14 Startup (Time Delay) with Removal of One Control Rod at Maximum Rate (36 mk/min).
Program I.

$$P_o = 10^5 \text{ w}$$

$$T_o = 20^\circ\text{C}$$

$$R = 10^{-10} \text{ w}$$

$$\alpha = 0$$

$$\lambda = 10 \text{ sec}$$

$$\gamma_i = 0.5 \quad i = 1, 2, 3, 4$$

$$\gamma_i = 1.0 \quad i = 5, 6$$

The rate of reactivity increase is much less than in the previous case (Problem 4.13).

4.15 Emergency Shutdown with Removal of Fuel at Various Rates (1, 2, and 5 liters/sec).
Program I.

$$P_o = 2 \times 10^6 \text{ w}$$

$$T_o = 430^\circ\text{C}$$

$$R = 10^{-10} \text{ w}$$

$$\alpha = 0$$

$$\tau = 10^{10} \text{ sec}$$

$$\gamma_i = 0.5 \quad i = 1, 2, 3, 4$$

$$\gamma_i = 1.0 \quad i = 5, 6$$

Upper curves in all figures correspond to removal rates of 1 liter/sec; lower curves to 5 liters/sec; central curves to 2 liters/sec. Perhaps the most interesting curves are those of liquid height with time, since an ultimate decrease in height indicates that the removal rate is adequate to relieve pressure in the vessel. The cusp in power at 0.87 sec for the 5 liters/sec dump is due to the decreased tamping as the upper fuel surface leaves the baffle.

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5.1 Conclusions

The results of Program II (Problems 4.1 through 4.8) indicate the high degree of stability of the reactor under normal operating conditions. The constant volume operation of the reactor region at power, together with the presence of the lower reservoir, leads to an enhanced negative temperature coefficient of reactivity (since the geometrical buckling is constant), thus contributing to the very stable operation.

In the startup problems, 4.9-4.14, the indication is that if the reactor is brought up to power with the fuel circulating pump running, and without the use of control rods (i.e., in case of control rod failure), the surge which occurs will be excessive. Problems 4.13 and 4.14, with the accidental addition of reactivity, do not show essential variations from 4.9-4.12. The conclusion to be drawn from this series is that advantage should be taken of the inherent negative temperature coefficient of reactivity of the upper region during startup, with gradual heating of the lower volume occurring through convection rather than through forced circulation. It is apparent that the period of the reactor is limited by the rate at which fuel can be exchanged between the two regions. Hence period and level safeties could be activated in sufficient time to avoid the power surge. For engineering reasons, principally for uniform heating of the pressure vessel, it will be desirable to start up at a power of about 100 kw. An adequate level safety, set to trip at 1 Mw, will provide sufficient control of the surge problem.

Problem 4.15, dealing with the problem of circulating pump failure at high power, was used as the basis for design of a line which will dump solution to a separate tank in such an eventuality. Current design of the dump line is to carry 15 liters/sec, which appears to be entirely adequate.

6. References

- 1 G. Goertzel, The Reactor Handbook, Technical Information Service, Atomic Energy Commission, First Edition, Chapter 1.6, June 1953.
- 2 P. J. Bendt, LA-1581.
- 3 Glasstone and Edlund, The Elements of Nuclear Reactor Theory, D. Van Nostrand Company, Inc., New York, 1952, p. 334.
- 4 S. I. Schlesinger and B. Carlson, LA-1573.

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