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MALFUNCTION ANALYSIS OF A CONCEPTUAL SPACE POWER FAST-SPECTRUM REACTOR

by John A. Peoples
Lewis Research Center
Cleveland, Ohio 44135

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16. Abstract

Four types of reactor malfunctions were examined: (1) control drum run-in, (2) reduction of coolant flow, (3) reduction of coolant inlet temperature, and (4) complete loss of coolant. The results show that this reactor concept exhibited an inherent safety factor due to its relatively slow response to each of these malfunctions. With the reactor at cold critical, reactivity insertions should be limited to a safe margin below 9.7 ¢/sec to avoid a prompt critical condition. Normal coolant velocity per channel (3.95 ft/sec or 1.20 m/sec) can be reduced to almost 10 percent of design before fuel temperatures approach 2500° R (1389 K). A step decrease of 210° R (117 K) in the normal coolant inlet temperature (2100° R or 1667 K) still provides a safety margin of almost 40 sec before fuel temperatures reach 2500° R (1389 K). Six sec are required following a complete loss of coolant accident for fuel temperatures to reach 2500° R (1389 K).

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MALFUNCTION ANALYSIS OF A CONCEPTUAL SPACE POWER FAST-SPECTRUM REACTOR

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SUMMARY

This investigation studies the dynamic response of a fast-spectrum lithium cooled nuclear reactor to four potential malfunctions. These malfunctions are (1) insertion of reactivity due to a run-in of the fueled control drums, (2) decrease in coolant flow rate, (3) decrease in coolant inlet temperature, and (4) complete loss of primary coolant.

The results show that this reactor concept exhibits an inherent safety factor due to its relatively slow response to each of these malfunctions. The ramp reactivity insertion accident due to the run-in of the fueled control drums does not present a serious problem provided the insertion rates are limited to a safe margin below 9.7 cents per second. The operational safety can further be enhanced by limiting the drum run-in malfunction to one, two, or three drums thus inserting only a fraction of the total reactivity available at any one time. Primary coolant flow rates can be reduced to almost 10 percent of design before fuel temperatures approach 2500° R (1389 K). At zero flow rate 8.0 seconds elapse before fuel temperatures reach 2500° R (1389 K) and about 30.0 seconds to reach 2900° R (1611 K). Coolant inlet temperatures can be reduced by as much as 210° R (117 K) and still maintain a relatively safe time margin (40 sec) to respond to the malfunction. The study of the complete loss of coolant accident shows that at least 6 seconds elapse before fuel temperatures reach 2500° R (1389 K) and 550 seconds before they approach the melting temperature.

From these data it appears that the reactor exhibits enough of a safety margin such that through careful design of the primary loop and reactor components most of these malfunctions can be completely avoided.

INTRODUCTION

Nuclear reactors must be designed in such a way as to ensure safe and reliable operation. This criterion is even more severe when applied to the design of a space power system. This type of reactor must operate unattended for long periods of time in the environment of space and therefore must be designed in such a way that certain accidents are not possible or that the system has the inherent capability to combat the most likely malfunctions.

The type of malfunction encountered is dependent on the design of the reactor and various components making up the primary loop. A concept currently being studied consists of a fast-spectrum reactor cooled by lithium and controlled by rotating fueled drums located on the periphery of the core. The reactor is then one member of the primary loop composed of the core, pump, heat exchanger, and appropriate plumbing for the lithium coolant. Based on a system of this type, several potential malfunctions are possible.

This report discusses the four types of primary loop malfunctions that are considered to have the severest consequences if they occur during the operating lifetime of the space power system:

(1) Insertion of reactivity due to a run-in of the fueled control drums

(2) Decrease in coolant flow rate

(3) Decrease in coolant inlet temperature

(4) Complete loss of coolant from the core (major rupture of primary loop)

The purpose of this report is to examine analytically the response of this particular reactor to these malfunctions and to determine

(1) The severity of the disturbance, that is, whether any damage limits have been exceeded

(2) The times required to reach these damage limits

The results of this study will not only offer some insight into the response characteristics of the system but will also provide information necessary to the design of the control system. The times available to respond to a particular malfunction will dictate the dynamic demands on the control drum actuators.

The initial preliminary accident analysis on this type of auxiliary power reactor was reported in reference 1. Since that time, the reactor concept has been updated and the primary loop defined in more detail. This study was made assuming a closed primary loop. Coolant transport times due to changes in coolant flow rate and variations in heat exchanger operation were taken into consideration in these calculations. Also, the afterheat power curves required for the analysis of the loss-of-coolant malfunction were updated and modified to represent more nearly the current reactor configuration.

DESCRIPTION OF REACTOR AND PRIMARY LOOP

The reactor design used for this analysis is a uranium nitride fueled, fast-spectrum, lithium-cooled reactor. At steady-state design conditions, the reactor would operate at

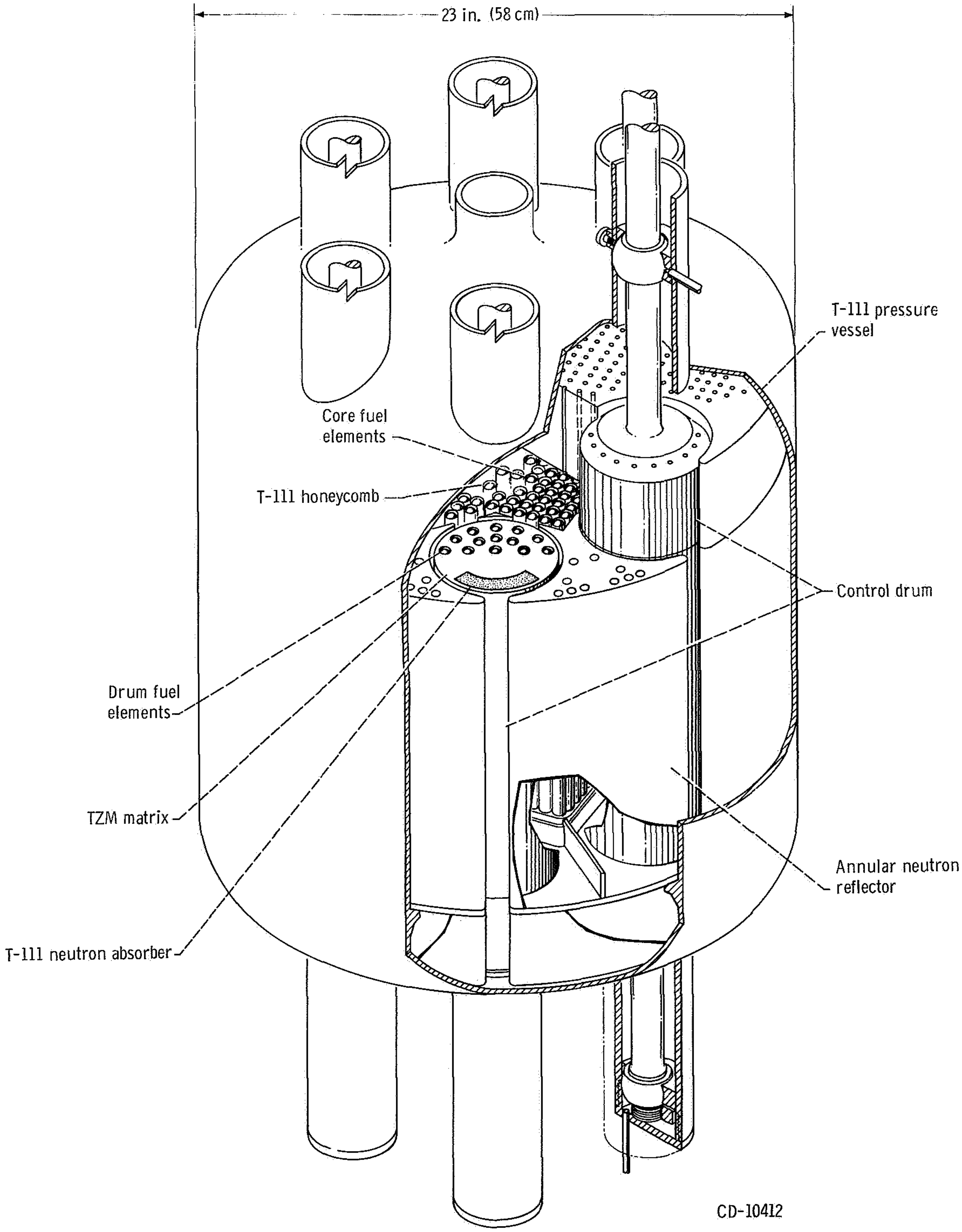

Figure 1. - Space power fast-spectrum reactor.

a thermal power level of 2.17 megawatts (7.4×10^6 Btu/hr). The reactor is shown in figure 1, and details of the pertinent reactor data are given in appendix A. The fuel consists of uranium nitride fully enriched in the U^{235} isotope. The fuel is then clad in tungsten and T-111 (tantalum - 8 percent tungsten - 2 percent hafnium) and placed in a T-111 honeycomb structure. A thin tungsten liner is present between the uranium nitride and the T-111 clad. A 0.040-inch- (0.1-cm-) thick coolant passage annulus exists between the honeycomb structure and the fuel pin.

An annular neutron reflector composed of TZM (molybdenum - 0.5 percent titanium - 0.08 percent zirconium) surrounds the reactor core. Embedded in the neutron reflector are six rotating fueled control drums. The control drums are essentially TZM, with fuel pins lining one side of the drum and an arc of neutron absorber (T-111) situated on the opposite side of the drum. Reactivity control is gained by rotating the drum in such a way as to move the fuel in closer to the core or farther away. The drum configuration position shown in figure 1 would provide maximum reactivity.

The core and fueled control drums are cooled by flowing liquid lithium. At steady-state design conditions, the total coolant flow is 20.7 pounds per second (9.4 kg/sec). The primary loop, consisting of the reactor, electromagnetic pump, and heat exchanger, is shown in figure 2.

For this particular reactor, it was previously determined (private communication with E. Lantz of Lewis) that the temperature defect amounted to about 2.70 percent $\Delta K/K$ and the fuel depletion to about 1.60 percent $\Delta K/K$, thus giving a total excess reactivity of 4.30 percent $\Delta K/K$. Therefore, at the start of full-power life, the control system holds 1.60 percent $\Delta K/K$ and in the cold critical condition holds 4.30 percent $\Delta K/K$.

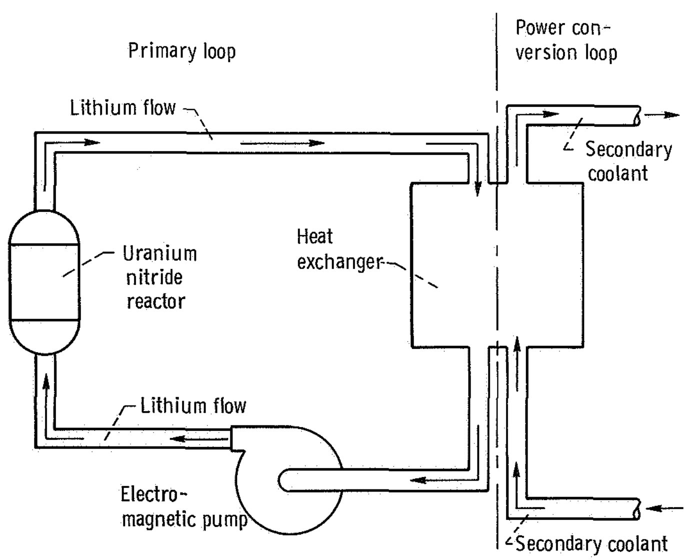

Figure 2. - Primary loop of space power fast-spectrum reactor.

A further assumption used in this study is that the uranium nitride fuel elements had a steady-state and a transient-temperature damage limit. Throughout this report, these fuel element damage limits will be used as the benchmarks in judging the severity of the malfunction and in defining the times available for taking corrective action to combat the accident. These limits are defined as

$$T_{steady\ state} = 2500^{\circ}\ R\ (1389\ K)$$

$$T_{transient} = 2900^{\circ}\ R\ (1611\ K)$$

It should be pointed out that these values for the damage limits are estimates and are considered to be quite conservative. They could change considerably as more information is gathered on the performance of these particular fuel elements. Obviously, should the value of the steady-state damage limit increase, the times available to respond to each of these malfunctions will correspondingly increase.

COMPUTER PROGRAMS

Two analytical programs were used in this study of reactor malfunctions. The two digital programs were FORE and CSMP. The FORE program was used exclusively in the study of the fueled-control-drum-run-in malfunction and the CSMP program for the remaining flow-oriented reactor accidents.

The FORE program was written by P. Greebler, D. B. Sherer, and N. H. Walton (ref. 2) for the purpose of understanding the dynamic response of fast reactors to programmed reactivity insertions specified as one or more ramps. Since the fueled-control-drum-run-in malfunction is essentially large insertions of reactivity at various ramp rates, this program was ideally suited to a parametric examination of this type of problem. The program calculates reactor power, fuel, coolant, clad, and structure temperatures as a function of time in response to the various programmed reactivity insertions.

The CSMP program was developed by IBM (ref. 3). The code is a program designed to facilitate the digital simulation of continuous processes on large-scale digital machines. The dynamic characteristics of the reactor design and primary loop previously discussed herein were simulated by this program. The program calculates reactor power, fuel, clad, and coolant temperatures of the core as well as heat exchanger temperatures as a function of time for either changes in reactivity or perturbations in coolant flow rate or temperature. Provisions are also available to change coolant transport times in the primary loop to correspond with variations in coolant flow rate. This program was therefore used exclusively in malfunctions that were flow oriented, namely, the decrease in coolant flow rate, the drop in coolant inlet temperature, and the complete loss of coolant.

RESULTS AND DISCUSSION

Control Drum Run-In

An operator error or an electrical malfunction could be the cause of a control drum runaway. In either case, if the fueled control drums are permitted to run in, thus inserting reactivity at a rapid rate, the consequences to the reactor could be quite serious.

The behavior of the reactor system in the event of such a malfunction is dependent on three parameters: (1) the rate at which the reactivity is inserted, (2) the total quantity of reactivity inserted, and (3) the time during the reactor run cycle that the drum runaway occurred.

Full-power drum malfunction. - A series of drum run-in calculations was made with the reactor at full-power design conditions and operating in an open-loop mode. At the start of full-power lifetime, the fueled control drums were assumed to hold 1.6 percent $\Delta K/K$ ($2.42). The initial calculations inserted the entire 2.42 dollars at various rates ranging from 1.21 to 0.11 dollar per second. In relation to the actual control drum movement, this case would represent all six control drums rotating fuel into the core in unison (ganged).

As a typical case, figure 3 shows the results of an 0.11-dollar-per-second reactivity insertion over a reactor operating time of 14 seconds. The reactor power, fuel element temperature, and lithium coolant temperature all continue a steady rise as a result of the control drum run-in. The fuel element temperature reaches the steady-state damage limit in 5.5 seconds and the transient limit in about 8.8 seconds after the start of the control drum malfunction.

Table I shows the results of the other calculations made for various reactivity insertion rates with the reactor at full power. The times listed in the table are the times taken to reach the various damage limits after the initiation of the fuel drum insertion (drum roll-in starts at $t = 0$).

It is difficult at this time to envision the proper drum orientation at the start of full power life for this reactor. However, if we were to assume that the six fueled control drums were at the same angular position at the beginning of the power run, then the case in which a total of 0.533 percent $\Delta K/K$ was inserted could correspond to two drums running in and the case of 0.267 percent $\Delta K/K$ would be comparable to a single drum run-in. As we can see from table I both of these cases are considerably safer than the ganged case. This is a strong point for having the fueled control drums operate in an independent mode.

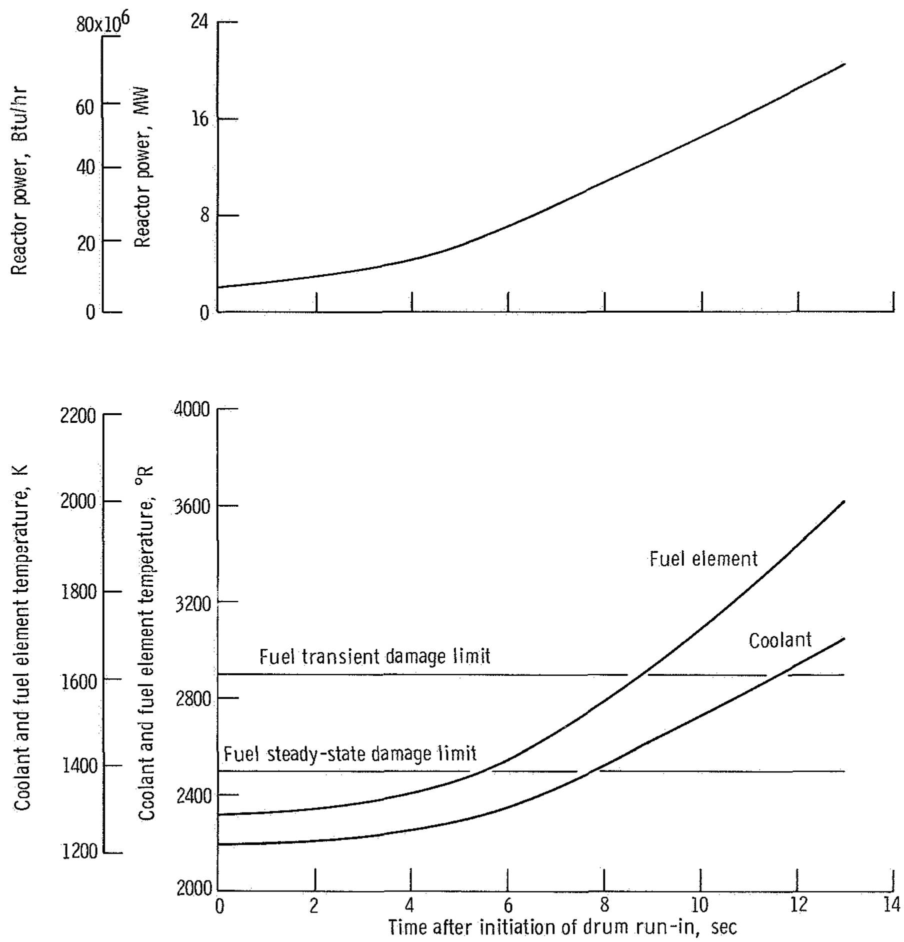

Figure 3. - Reactor power and fuel element and coolant temperatures as function of time after initiation of drum run-in at 2° per second or at reactivity insertion of 11 cents per second for total of 2.42 dollars (reactor at full power).

TABLE I. - TIME REQUIRED FOR FUEL TEMPERATURES TO REACH STEADY-STATE - TRANSIENT DAMAGE LIMITS AND FUEL MELTDOWN AS RESULT OF VARIOUS REACTIVITY INSERTION RATES WITH REACTOR AT FULL POWER

Total reactivity inserted		Reactivity insertion rate, $/sec	Fuel drum angular velocity, deg/sec	Time, sec, to reach -		
percent ΔK/K	$			Steady-state fuel element damage limit	Transient fuel damage limit	Fuel element meltdown
1.60	2.42	1.21	22.5	[a]0.87 [b].85 [c].80	[a]1.25 [b]1.18 [c]1.00	[a]~10 to 15 [b]~10 [c]2.20
		0.48	9.0	[a]1.85 [b]1.85 [c]1.80	[a]2.80 [b]2.67 [c]2.35	[a]>15.0 [b]>13.0 [c]4.65
		0.24	4.5	[b]3.30	[b]4.65	[b]>13.0
		0.11	2.0	[b]5.50	[b]8.80	[b]>20.0
0.533	0.80	0.035	2	[b]11.90	[b]>20.0	(d)
0.267	0.40	0.071	8	[b]10.50	(d)	(d)
		0.035	4	[b]11.90	(d)	(d)
		0.018	2	[b]25.00	(d)	(d)

[a]Maximum coefficients.
[b]Nominal coefficients.
[c]Minimum coefficients.
[d]Limits were not reached during 30.0-sec time span of calculation.

Because of the preliminary nature of the reactor design, the reactivity feedback coefficients (Doppler, core expansion, and coolant density) were not well defined at this time. As a result, many of these calculations on reactivity insertion were carried out using three values for each of the coefficients, a maximum, a minimum, and a nominal value (see appendix A). Figure 4 shows the time to reach the steady-state and transient-fuel-element-temperature damage limits for various drum insertion velocities using the minimum and nominal values of feedback reactivity coefficients.

Cold critical drum malfunction. - With the reactor in the cold critical condition, 4.30 percent ΔK/K ($6.51), 1.43 percent ΔK/K ($2.17), and 0.717 percent ΔK/K ($1.08) were ramped into the reactor at various rates. These rates ranged from 60 to approximately 2.4 cents per second.

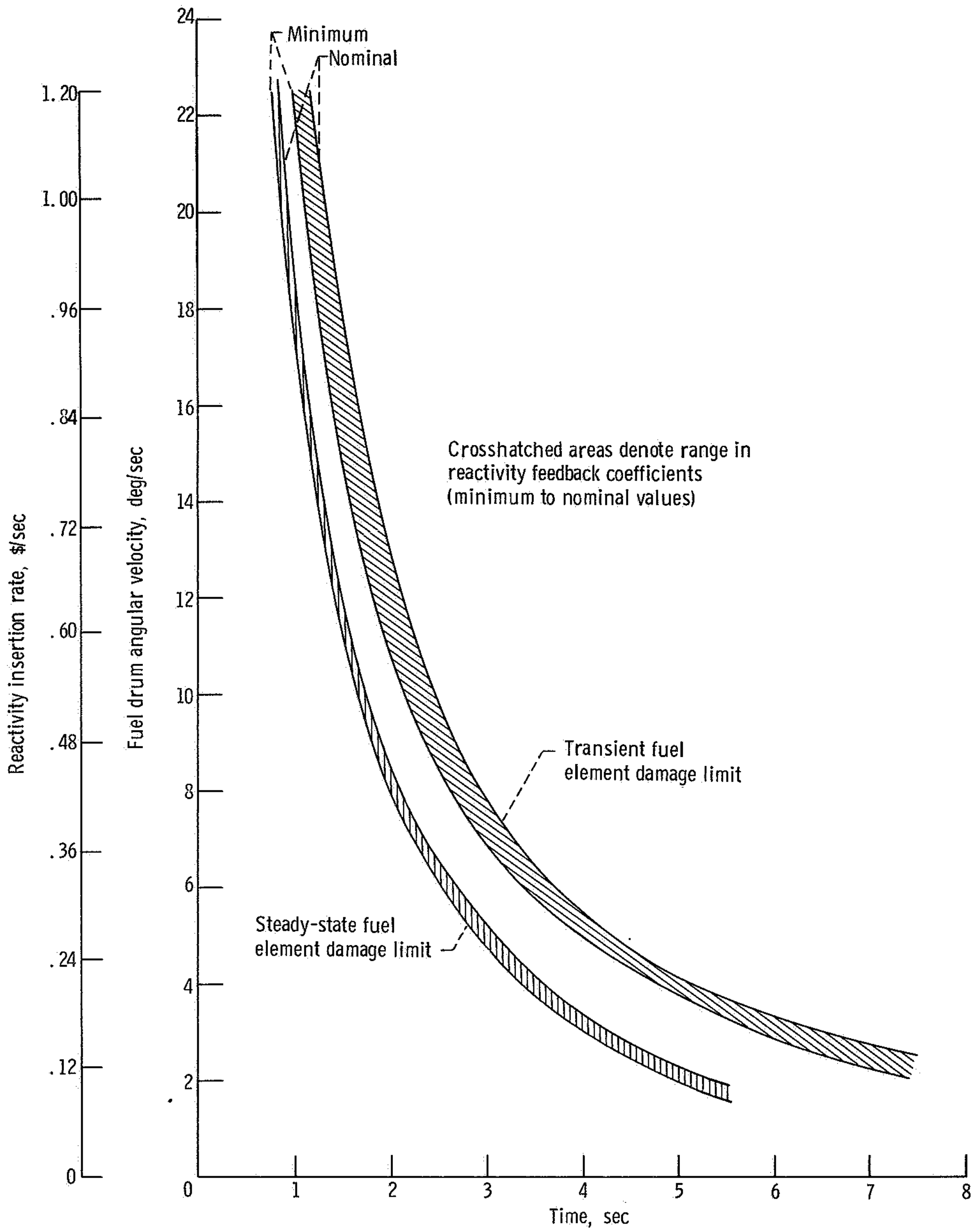

Figure 4. - Time required to reach steady-state or transient fuel element damage limits as function of run-in angular velocity of all six fueled control drums (hottest section of hottest fuel element). Reactor operating at design power (2.17 MW or 7.4×10^{6} Btu/hr) at time fuel drums began roll-in; total reactivity insertion, $\pm 0.016\,\Delta K/K$ ($2.42).

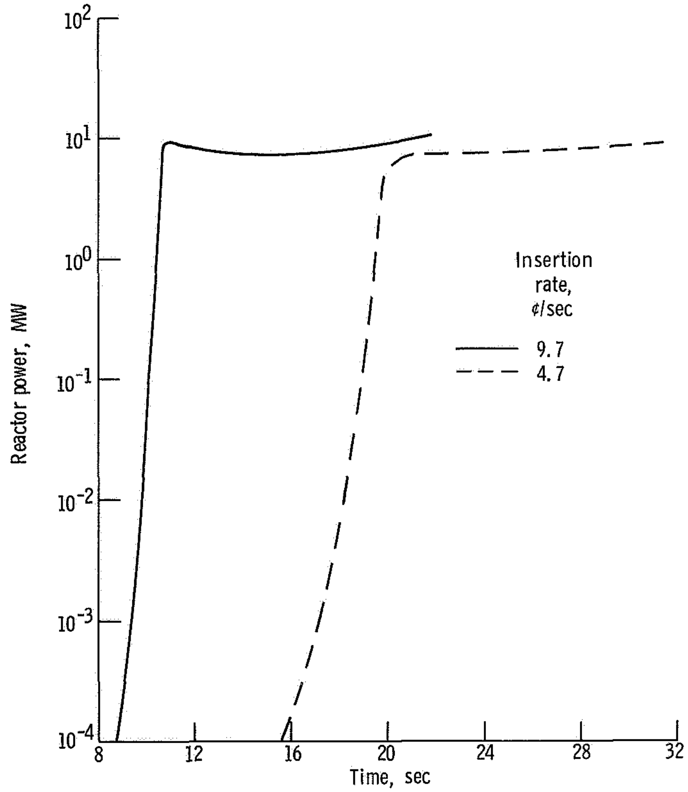

Figure 5. - Reactor power as function of time after start of single drum run-in with reactor initially at cold critical condition.

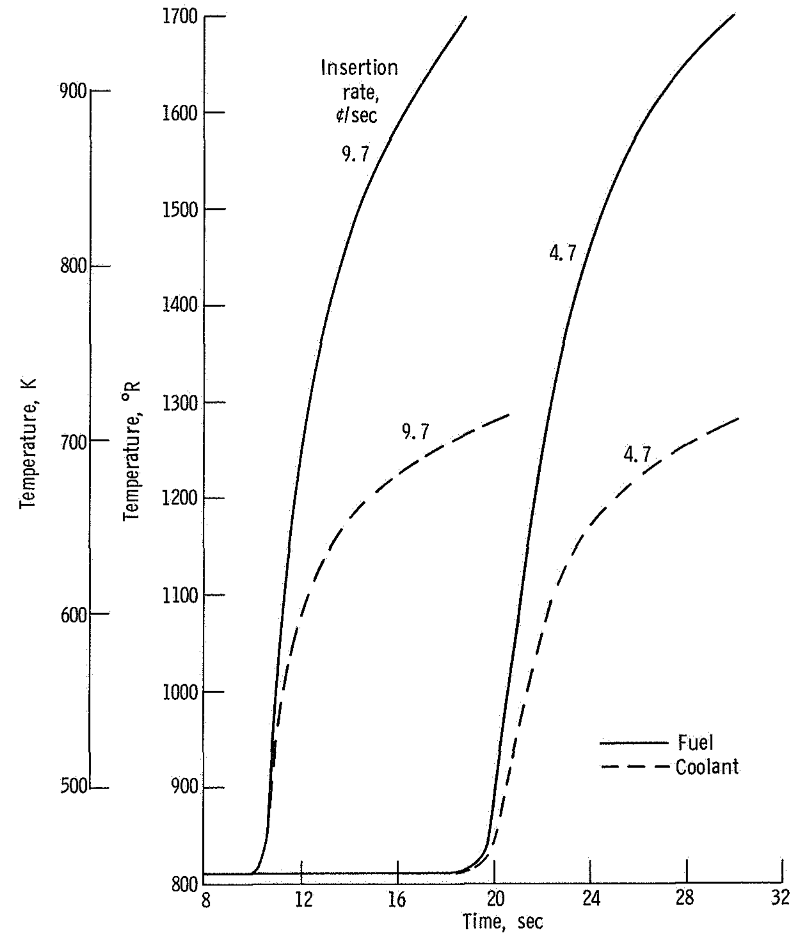

Figure 6. - Fuel element and coolant temperatures as function of time after start of single drum run-in with reactor at cold critical condition.

TABLE II. - RESULTS OF VARIOUS REACTIVITY INSERTION RATES

WITH REACTOR AT COLD CRITICAL CONDITION

Total reactivity inserted		Time, sec, to reach -				Data at end of computer run				
		Insertion rate		Prompt critical	Design power (2.17 MW)	Time, sec	Fuel temperature		Coolant temperature	
percent ΔK/K	$	¢/sec	deg/sec				°R	K	°R	K
4.30	6.51	60	9.5	1.50	1.55	2.80	1650	916	1310	728
		12.6	2.0	~6.90	7.00	13	2110	1172	1544	858
1.43	2.17	20.0	9.5	4.50	4.50	10	2690	1494	1920	1067
		8.4	4.0	(a)	~10.0	30.0	3210	1783	2170	1200
		4.2	2.0	(a)	~20.0	30.0	1970	1094	1440	800
0.717	1.08	9.7	8.0	~10	~10.0	30.0	1850	1028	1350	750
		4.8	4.0	(a)	19.6	30.0	1708	949	1280	711
		2.4	20	(a)	>30.0	30.0	820	455	820	455

[a]Was not reached.

The results of these calculations are shown in table II. These data indicate that, if the reactor is to avoid a prompt critical situation, precautions must be taken to limit the fueled drum run-in at cold critical to a reactivity insertion rate no greater than 8 to 9 cents per second.

Figures 5 and 6 show the power and the fuel and coolant temperatures as a function of time after the start of a drum roll-in. The reactor is initially at the cold critical condition for ramp rates of 9.7 and 4.7 cents per second. A detailed examination of the case in which 9.7 cents per second was inserted showed that the reactor approaches extremely close to the prompt critical condition. Therefore, in order to assure safe operation during startup, the maximum permissible reactivity insertion rate should be limited to a safe margin below the 9.7 cents per second rate.

Decrease in Coolant Flow Rate

The decrease-in-lithium-coolant-flow-rate accident is the type of malfunction that might occur if all or part of the pumping power in the primary loop is lost or if there is an increase in resistance to flow. For this particular analysis, the lithium flow velocity was ramped down from 100 percent of design (3.95 ft/sec or 120.4 cm/sec) to 50 percent (1.975 ft/sec or 60.2 cm/sec) and 10 percent (0.395 ft/sec or 12.0 cm/sec) and zero flow in 1 second. Once the flow was reduced to these values it was held constant at that

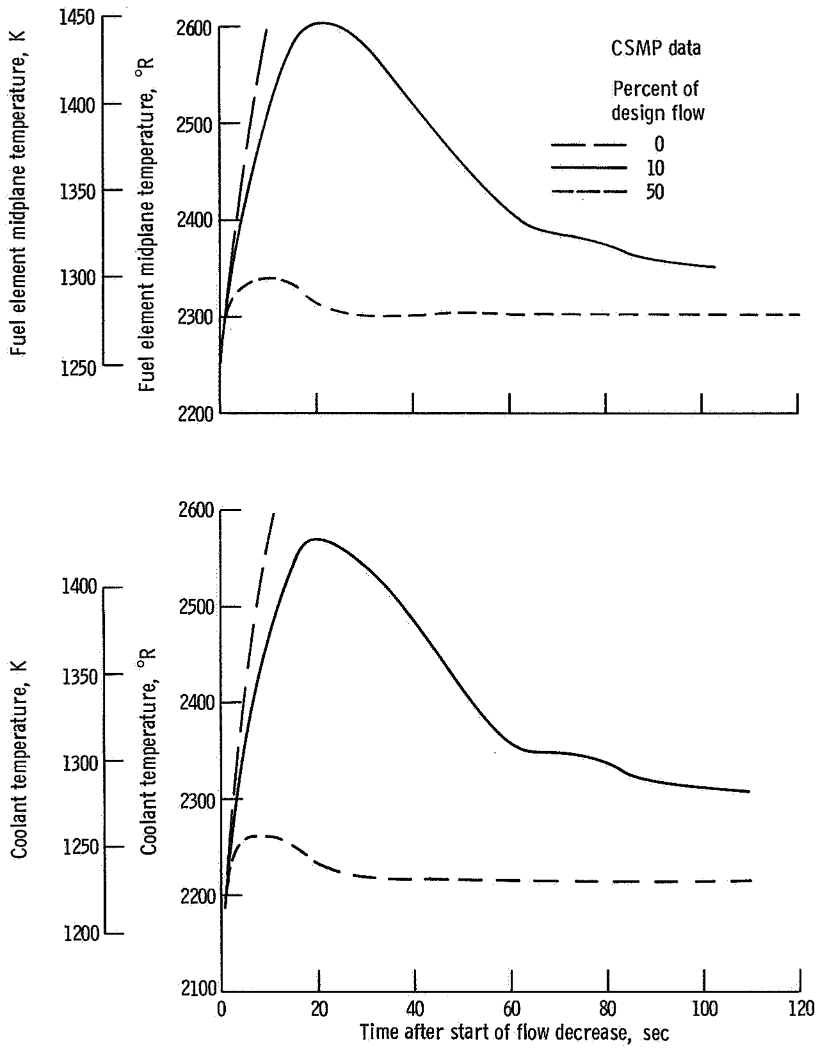

Figure 7. - Fuel element and coolant temperatures at various times after decrease in coolant flow from 100 to 50 and 10 percent of design.

value for the remainder of the calculation.

The results of these calculations are presented in figures 7 and 8. Figure 7 shows fuel element and coolant temperatures as a function of time after the initiation of coolant flow decrease. From this figure we see that the reactor fuel elements are relatively insensitive to rather large reductions in flow. A decrease in coolant flow to 50 percent of design appears to present no severe problems in the way of forcing fuel element temperatures to their damage limits. The fuel temperature, in this case, reaches a maximum of only 2335° R (1297 K) in about 10 seconds and then settles back to a steady-state value of 2300° F (1278 K). It is not until the coolant flow is reduced to 10 percent or below of design that fuel temperatures rise above the damage limits. At 10 percent of design flow, the fuel temperatures reach 2500° R (1389 K) in about 10.0 seconds. Even with the

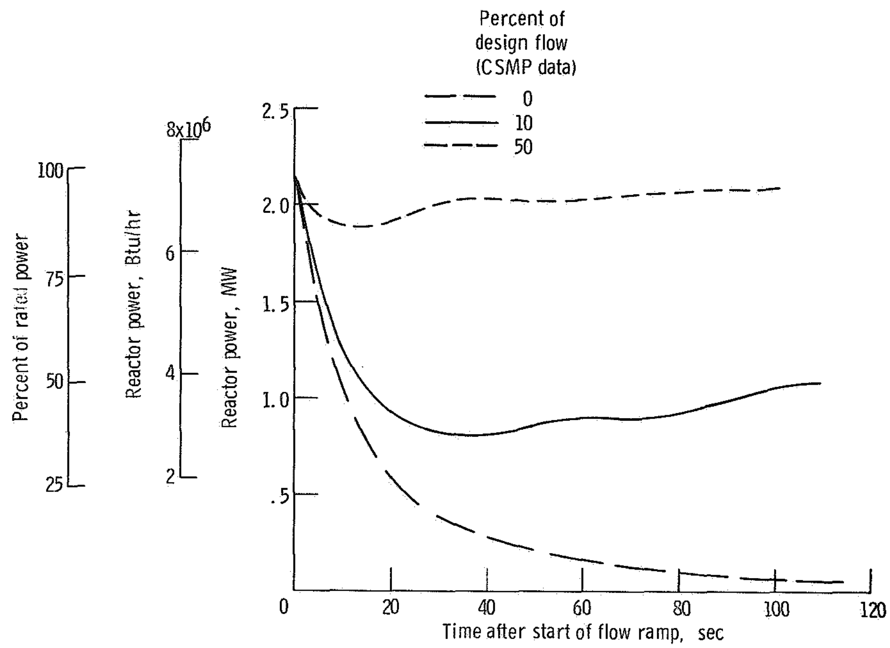

Figure 8. - Reactor power as function of time after start of ramp decrease in lithium coolant flow.

flow ramped to zero in 1 second, there still remains 8.0 seconds before fuel temperatures approach 2500° R (1389 K) and about 30.0 seconds before they reach 2900° R (1611 K).

Figure 8 is a plot of reactor power as a function of time after the various decreases in flow. The significance of these data is that for the 50-percent decrease in flow the power remains essentially constant, while the decrease to 10 percent of flow reduces the power to one-half of design and zero flow drops the power to about 2 percent of design.

During each of the loss-of-pumping-power computations, the fueled control drums were held in a stationary position. No effort was made to compensate for the power perturbations by changing or scramming the fueled control drum positions.

Decrease in Coolant Inlet Temperature

For this analytic investigation, the reactor was at full-power and steady-state conditions, and then the coolant inlet temperature was suddenly decreased. This type of malfunction corresponds to the cold slug accident. Cold lithium, possibly from a makeup system, is assumed to be introduced into the primary loop between the heat exchanger and the core inlet, thus reducing the overall temperature of the coolant entering the reactor.

At full-power steady-state design conditions, the lithium inlet temperature is

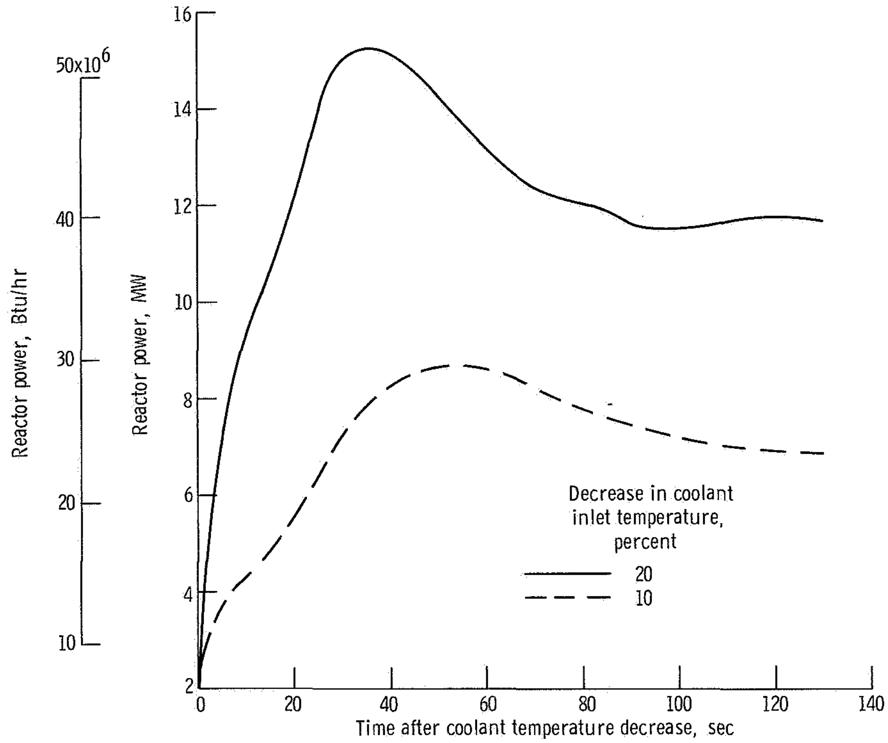

Figure 9. - Reactor power for various times after decrease in lithium coolant inlet temperature.

2100° R (1667 K). With the use of a step function in the calculations, the coolant inlet temperature was dropped 10 percent to 1890° R (1050 K) and then by 20 percent to 1680° R (933 K). The results are presented in figures 9 and 10. Figure 9 shows the reactor power as a function of time after the step decrease in coolant temperature. The 20-percent decrease in coolant temperature to 1680° R (933 K) (solid line) results in a power increase to 15 megawatts in about 30 seconds. The 10-percent decrease in coolant inlet temperature is also shown in figure 9 (dashed line). The peak reactor power of 8.6 megawatts (29.36 Btu/hr) is not reached until almost 50 seconds after the cold lithium is introduced into the reactor.

Figure 10 is a plot of the fuel element and coolant temperatures at various times after the start of the malfunction. The severity of the 20-percent (420° R or 233 K) drop in inlet temperature is seen by the rapid increase in fuel element temperature. The steady-state damage limit (2500° R or 1389 K) was reached in 14 seconds and the transient limit (2900° R or 1611 K) in about 38 seconds.

The 10-percent drop in inlet temperature malfunction is not nearly as drastic, in that 40 seconds were required for the fuel element temperature to reach the steady-state damage limit. It further appears from the data that the fuel element temperature will not exceed the transient damage limit of 2900° R (1611 K).

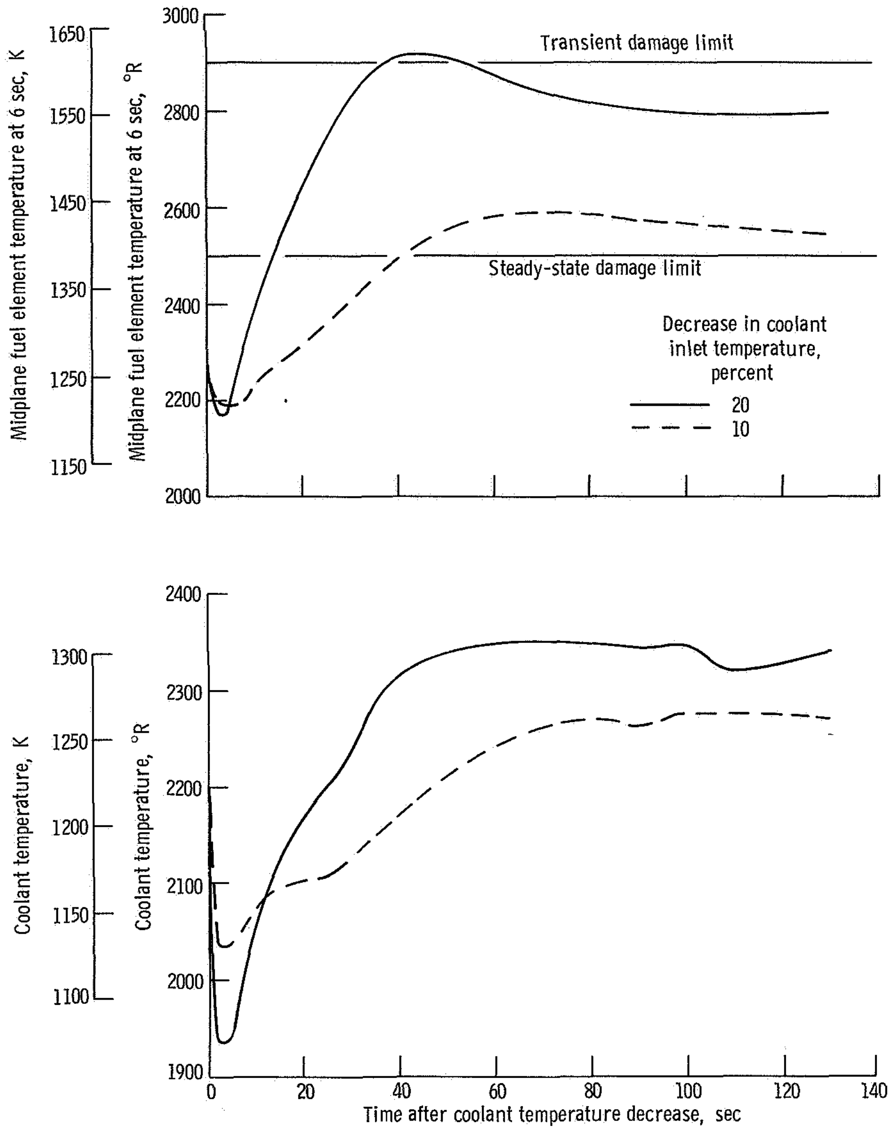

Figure 10. - Fuel element and coolant temperatures as function of time after 10- and 20-percent decrease in coolant inlet temperature.

Loss of Coolant

One of the more hazardous malfunctions that can occur is the complete loss of lithium coolant. If the core suffers a loss of coolant, the reactor will inherently shut itself down; however, the afterheat power generated within the fuel pins will be of sufficient magnitude that fuel element meltdown will occur. For the fuel pin and core design investigated, there is a possibility that the molten uranium could reassemble in a critical mass. Naturally, this type of situation must be avoided.

This section of the report examines the loss-of-coolant accident and determines fuel element and clad temperatures as a function of time after the start of the malfunction. Two cases were studied: (1) the case where no auxiliary lithium coolant was introduced into the core following the incident and (2) the case where auxiliary lithium coolant was supplied in either a continuous or pulsed mode.

The analysis of the loss-of-coolant malfunction required the writing of a subprogram for use with the main CSMP program so that the afterheat power generated by the reactor could be determined. A description of the equations used and the final results are given in appendix B.

The problem of the loss-of-coolant malfunction can be divided into two phases. The first phase, the actual loss of coolant and the dynamic response of the reactor system to this loss, was calculated using the basic reactor kinetics equations. When the core is voided, the loss of reactivity due to the loss of lithium is sufficient to shut the reactor down. As a result, the reactor power continuously decreases to a point where it becomes secondary to the afterheat power generated by fission products (see fig. 11). At this point, the problem passes into the second phase with the decay heat providing the driving energy to force fuel element temperatures to their melting point.

The basic assumptions of the problem are that, prior to the incident, the reactor had been operating for either 1 day or 1 year at steady-state design operating power. Furthermore, the loss of coolant resulted from a rupture or shearing of the coolant line in such a manner that the reactor core was completely voided of lithium within about 2 seconds.

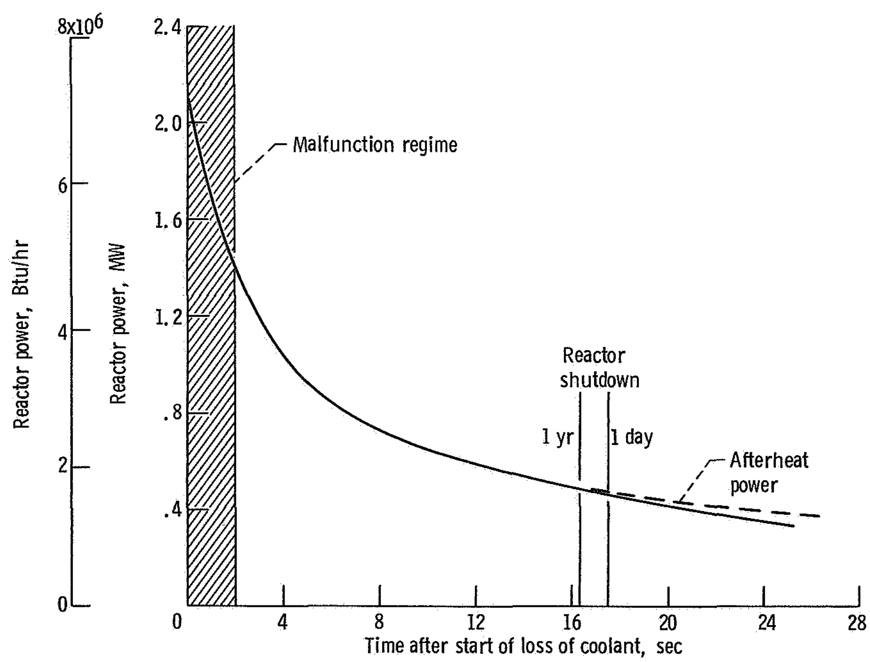

Figure 11. - Reactor power as function of time after start of malfunction.

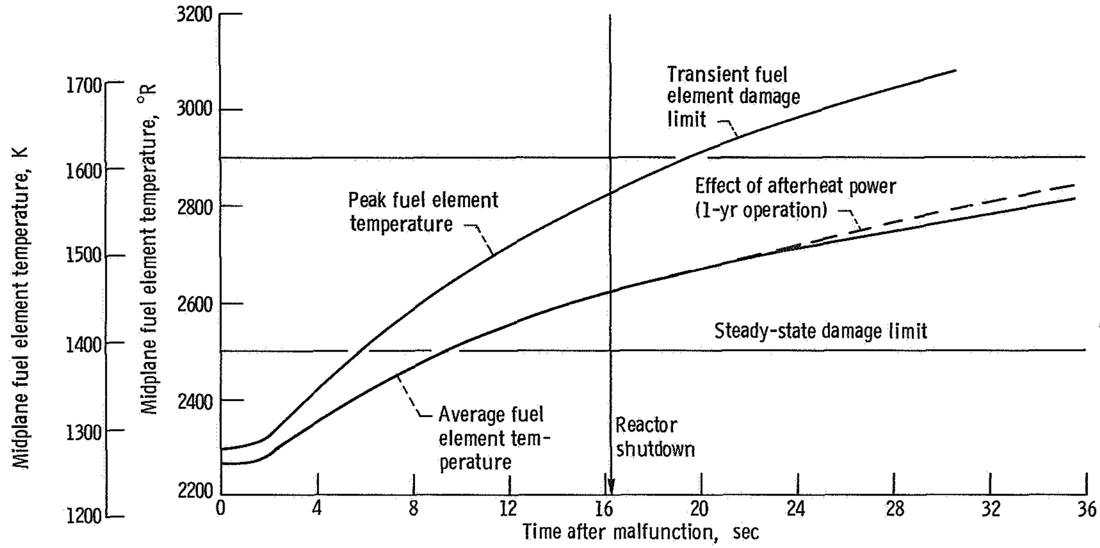

Figure 12. - Peak and average midplane fuel element temperatures as function of time after loss-of-coolant malfunction. Reactor operating for 1 year.

Figure 12 shows core midplane fuel element temperatures as a function of time after the start of the incident. The curve of the peak fuel element temperatures refers to an element located near the center of the core. These elements have slightly different dimensional characteristics from those located farther out near the periphery of the core. These elements are also subject to a power peaking factor as a consequence of their location in the core. The average fuel element curve noted in figure 12 is essentially the plot of a fictitious element that incorporates the average dimensions of all the fuel elements in the core and uses a power peaking factor of 1.00.

Figure 12 shows that the important peak fuel element temperatures rise quite rapidly and approach the steady-state damage limit within 6 seconds after the start of the incident. By the time the afterheat power becomes the driving source of energy (approximately t = 17.0 sec), the peak fuel element temperatures are already approaching the transient damage limit of 2900° R (1611 K).

Figure 13 is an extension of these data showing peak and average fuel element temperatures (at core midplane) over a period of 600 seconds following the loss of coolant. Approximately 50 seconds after the start of the accident, the reactor run history begins to influence the rate of fuel element temperature rise. This figure shows that the voided fuel elements will rather quickly approach their slumping or melting points. A peak fuel element that has been operating at design conditions for 1 year reaches its melting point within about 550 seconds after the initial loss of coolant.

These data show that, unless the decay heat is dissipated in some manner, the fuel elements will eventually melt. Considering the high uranium 235 inventory carried by this reactor (approximately 180 kg - the equivalent in an unreflected configuration of

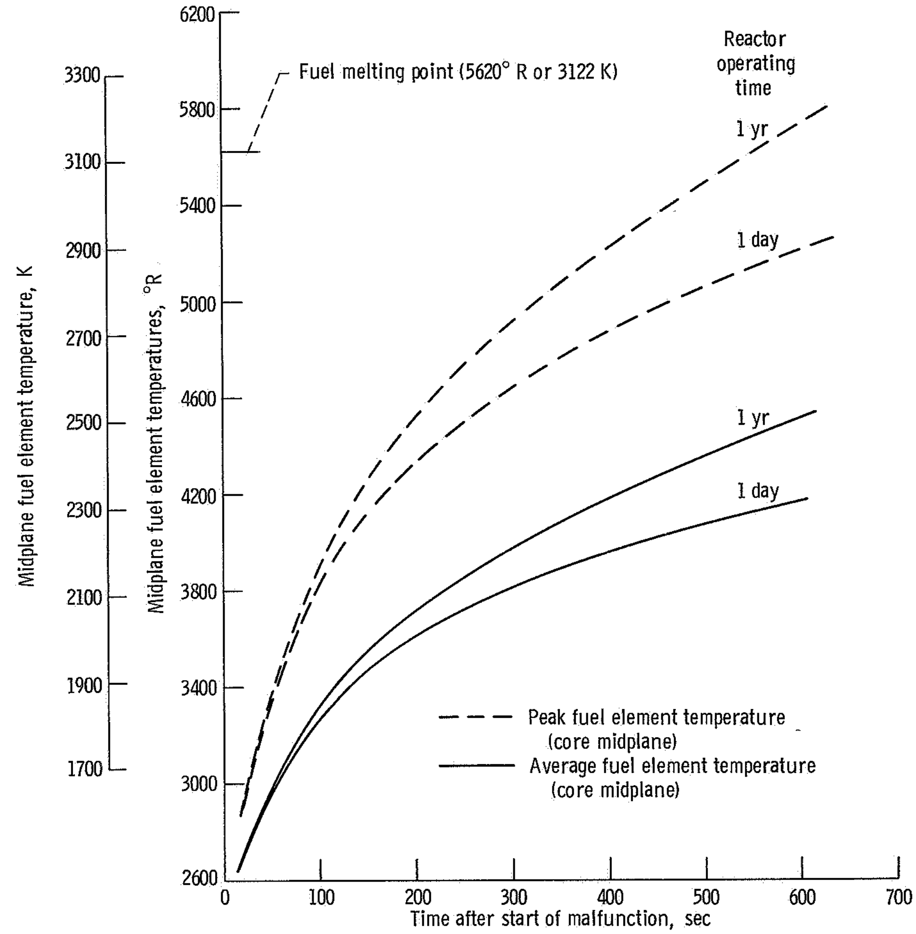

Figure 13. - Fuel element temperatures as function of time after start of loss-of-coolant malfunction.

about three critical masses, ref. 4) a fuel meltdown and possible reassembly could present a serious problem. In order to avoid this type of situation, some method or methods should be devised to safely carry away the afterheat power and thus prevent a meltdown of the core.

One method of dissipating the afterheat power is by re-establishing lithium coolant flow to the reactor core. Two modes of aftercooling the reactor were examined and will be discussed in the following sections.

Continuous flow mode. - This method of aftercooling assumes that lithium coolant flow is introduced into the core at full-design flow (20.7 lb/sec or 9.4 kg/sec) when the peak fuel element temperature (at core midplane) reaches 3000° R (1667 K). The flow then remains at that rated condition indefinitely.

Pulsed flow mode. - This method of aftercooling again assumes that lithium coolant flow is introduced into the core at full-design flow (20.7 lb/sec or 9.4 kg/sec) when the peak fuel element temperature reaches 3000° R (1667 K). However, in this mode, the flow remains on for only 15 seconds and is then turned off. Subsequently, each time the fuel element temperature reaches 3000° R (1667 K), the coolant is again turned on for 15 seconds.

In both instances, it was assumed that the aftercooling lithium was supplied from a reserve tank at constant inlet temperature in an open-loop fashion. A 3000° R (1667 K) peak fuel element temperature was arbitrarily chosen as the trigger temperature to turn on lithium coolant. Boiling of the lithium coolant was not considered for these calculations.

The effect of the continuous flow method on fuel element temperature as a function of time after the start of the malfunction is shown in figure 14. A constant inlet coolant temperature of 2100° R (1167 K) was assumed for these calculations. The reactor operating history, of 1 year or 1 day, prior to the time of the malfunction appeared to have no effect on the results. The data for both operating times coincided.

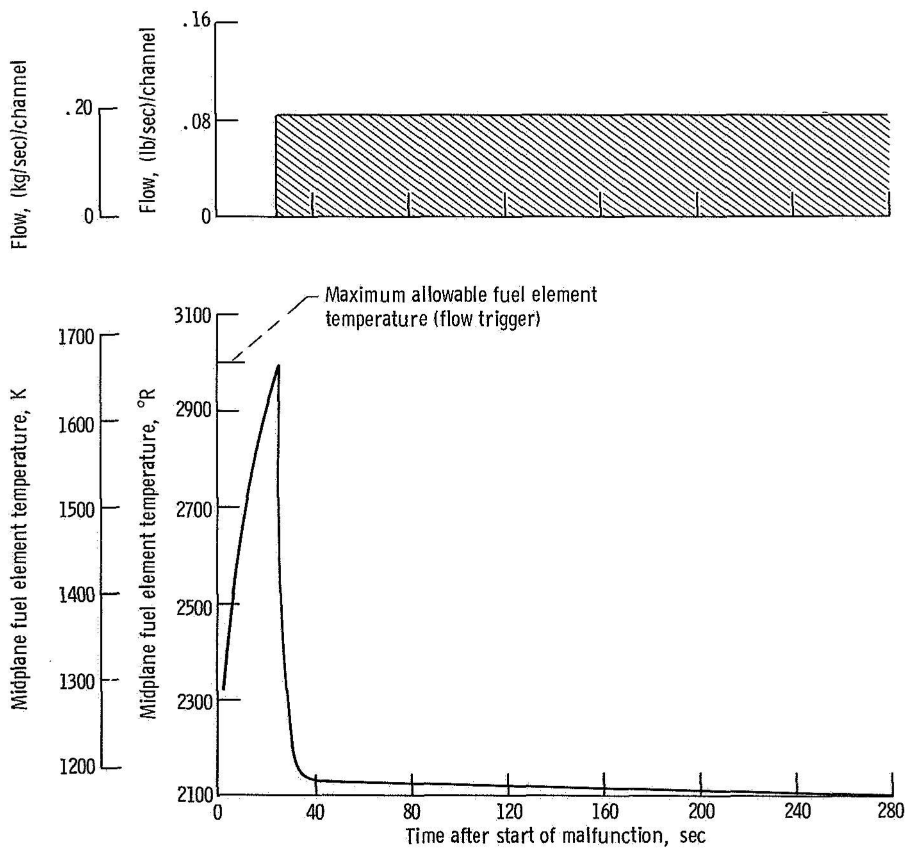

Figure 14. - Afterheat temperatures for continuous cooling following loss-of-coolant malfunction. Reactor operating for 1 year prior to malfunction; coolant inlet temperature assumed constant at 2100° R (1167 K).

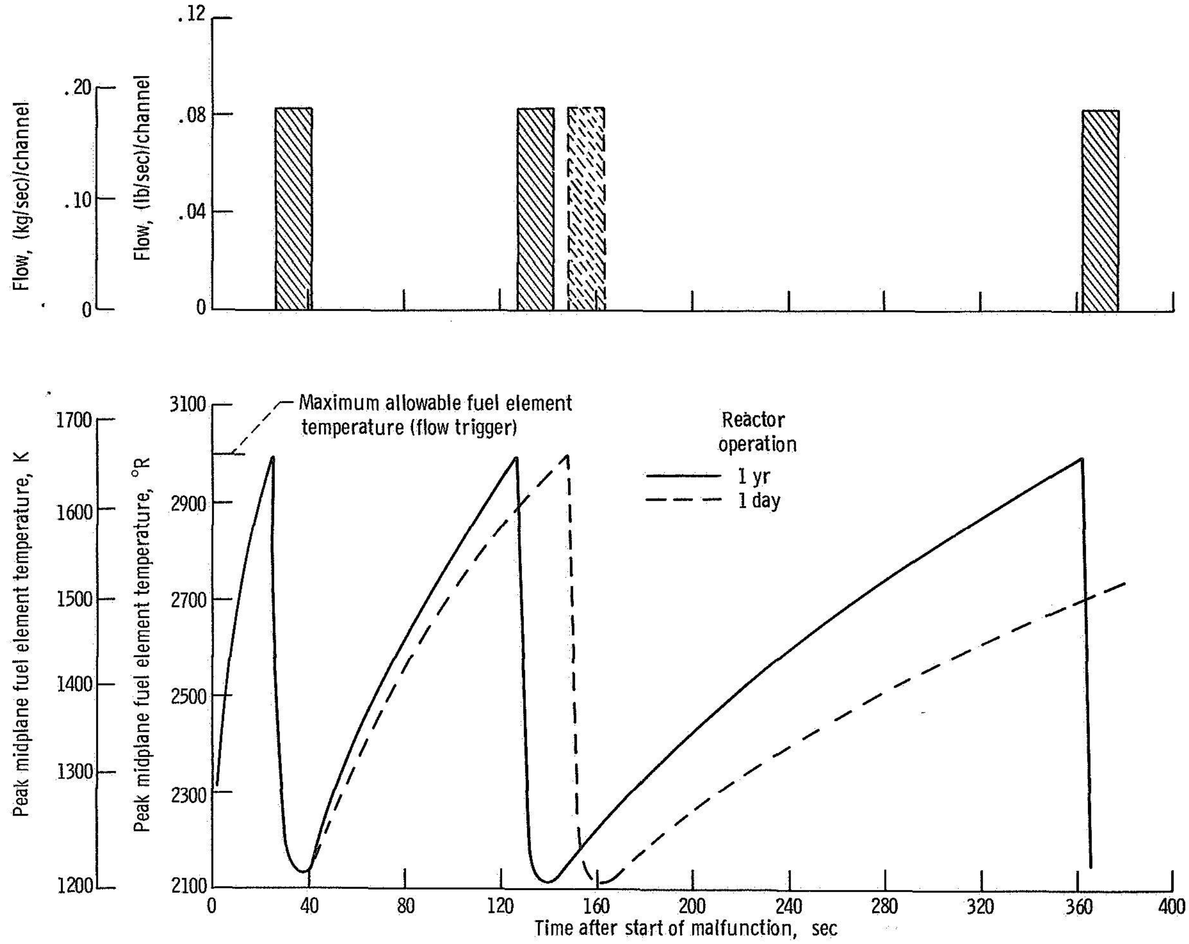

Figure 15. - Afterheat fuel temperatures for pulsed cooling following loss-of-coolant malfunction. Reactor operating for 1 year prior to malfunction; coolant inlet temperature assumed constant at 2100° R (1167 K); pulse width, 15 seconds.

The results of pulse cooling the core following this type of malfunction are shown in figure 15. With the use of a lithium pulse of 15 seconds duration and a flow rate of 20.7 pounds per second (9.4 kg/sec), the peak fuel element temperature was reduced by aftercooling each time the fuel temperature rose to the maximum allowable temperature of 3000° R (1667 K). Figure 15 shows that the data for 1 year and 1 day of reactor operation coincide through the first 40 seconds of the incident. At times greater than 40 seconds, the reactor run history prior to the accident begins to influence the rate of fuel element temperature rise.

The data shown in figure 15 were extended to 7200 seconds after the start of the malfunction. The results showed that some 16 pulses of coolant were required in order to keep the fuel element temperature below 3000° R (1667 K) during this period. This would amount to approximately 5000 pounds (2272 kg) of lithium. At the 7200-second point, the pulses were approximately 1500 seconds apart, thus indicating that a considerably larger number of pulses would still be required to maintain fuel temperatures below 3000° R (1667 K).

CONCLUSIONS

Four potential malfunctions of a fast spectrum lithium-cooled space nuclear powerplant were studied. Based on the results of this analysis, the following conclusions were made:

1. The reactor exhibits an inherent safety factor due to its relatively slow response to each of these malfunctions.

2. The ramp reactivity insertion accident due to the run-in of the fueled control drums does not present a serious problem provided the insertion rates are limited to a safe margin below 9.7 cents per second.

3. Primary coolant flow can be reduced to almost 10 percent of design before fuel element temperatures approach 2500° R. At a zero flow rate 8.0 seconds elapse before fuel temperatures reach 2500° R (1389 K) and about 30.0 seconds to reach 2900° R (1611 K).

4. Coolant inlet temperatures can be reduced as a step function by as much as 210° R (117 K) and still maintain a relatively safe time margin (40 sec) to respond to the malfunction.

5. The complete loss of primary coolant from the reactor will require that the decay heat be dissipated in some manner in order to avoid fuel element meltdown. If the reactor has operated at design power for 1 year prior to the malfunction, fuel temperatures will approach their melting point in approximately 550 seconds after the core is voided.

One method of avoiding fuel meltdown would be to re-establish lithium coolant flow from an emergency coolant supply. This analysis shows that an open loop mode of supplying lithium is not feasible because of the large lithium inventory (>5000 lb or 2272 kg) that would be required.

Lewis Research Center,
National Aeronautics and Space Administration,
Cleveland, Ohio, June 22, 1970,
120-27.

APPENDIX A

REACTOR PARAMETERS

Core power, MW (Btu/hr)	2.17 (7.4×10^{6})
Fuel volume, ft^3 (m^3)	0.515 (0.01457)
Fraction of energy due to gamma and neutron heating	0.050
Radial power peaking factor	1.20
Neutrons/fission, ν	2.52
Neutron lifetime, l, sec	4.00×10^{-8}
Decay constants, sec^{-1}	
Group 1, λ_1	3.88
Group 2, λ_2	1.40
Group 3, λ_3	0.311
Group 4, λ_4	0.116
Group 5, λ_5	0.317
Group 6, λ_6	0.0127
Decay fraction	
Group 1, β_1	0.0001719
Group 2, β_2	0.0008431
Group 3, β_3	0.002691
Group 4, β_4	0.001238
Group 5, β_5	0.001402
Group 6, β_6	0.0002517
Core height, ft (m)	1.2333 (0.3758)
Outer radius of fuel rod, ft (m)	0.0260 (0.00792)
Outer radius of clad, ft (m)	0.03125 (0.00953)
Outer radius of coolant, ft (m)	0.03458 (0.01054)
Thickness of clad, ft (m)	0.00525 (0.00160)
Density of fuel, lb/ft^3 (kg/m^3)	0.892×10^{2} (1.429×10^{3})
Density of clad, lb/ft^3 (kg/m^3)	0.1045×10^{4} (1.67×10^{4})
Density of coolant, lb/ft^3 (kg/m^3)	0.2760×10^{2} (4.42×10^{2})
Density of structure, lb/ft^3 (kg/m^3)	0.1045×10^{4} (1.67×10^{4})
Melting temperature of fuel, ^{o}R (K)	5620 (3122)
Inlet coolant temperature, ^{o}R (K)	2100 (1167)
Velocity of coolant in average channel, ft/sec (m/sec)	3.95 (1.204)
Product of Doppler coefficient and absolute temperature	
Maximum	-0.0040
Minimum	0.00

Nominal . . . . -0.0040

Core height coefficient

Maximum . . . . -0.6537

Minimum . . . . -0.4576

Nominal . . . . -0.5884

Core radial coefficient

Maximum . . . . -1.4514

Minimum . . . . -1.0159

Nominal . . . . -1.3062

Coolant density coefficient

Maximum 0.02512

Minimum 0.00837

Nominal 0.01675

APPENDIX B

AFTERHEAT CALCULATIONS FOR SPACE POWER FAST-SPECTRUM REACTOR

Afterheat power produced by the lithium-cooled nuclear reactor space powerplant is generated by the following processes:

(1) Delayed neutron fissions

(2) Fission product decay

(3) Absorption induced radioactivity

The contribution of each of these sources of power to the total afterheat power is calculated using the following equations:

Delayed neutron fission power (ref. 5):

$$P = P_0(0.146)e^{-\tau/80}$$

where

P total power

P_0 reactor steady-state power, 2.17 MW (7.4×10^6 Btu/hr)

τ time after shutdown, sec

Fission product decay power (ref. 6):

$$P = 0.005\ P_0\left[A\tau^{-a} - A(t + \tau)^{-a}\right]$$

or

$$P = 1.085\times10^{-2}\ A\left[\tau^{-a} - (\tau + t)^{-a}\right]$$

where

t reactor operating time, sec

and if $1.0 \le \tau \le 1.5\times10^2$ seconds, then $A = 15.31$ and $a = 0.1807$; and if $1.5\times10^2 \le \tau \le 4.0\times10^6$ seconds, then $A = 26.02$ and $a = 0.2834$.

Absorption induced power (ref. 7):

$$A_i = N_p \sigma_j \varphi_j \left(1 - e^{-\lambda_i t}\right) e^{-\lambda_i \tau}$$

where

A_i activity produced by i^{th} element

N_p mass of isotope P which produces i

σ_j activation cross section per unit mass of P for j^{th} energy group

φ_j neutron flux for j^{th} energy group

λ_i decay constant for i^{th} element

Figures 16 and 17 show the total afterheat power generated by the reactor for times after shutdown ranging from 10 to 4×10^6 seconds.

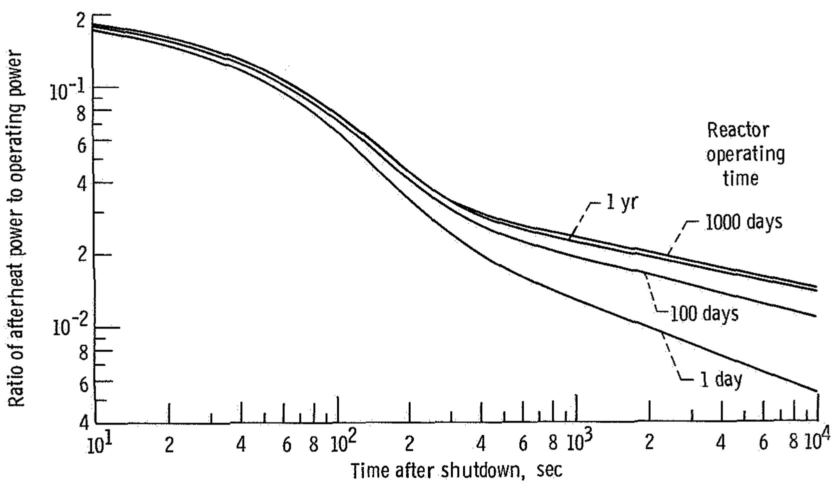

Figure 16. - Ratio of reactor afterheat power to operating power (2.17 MW or 7.4×10^6 Btu/hr) for short times after shutdown.

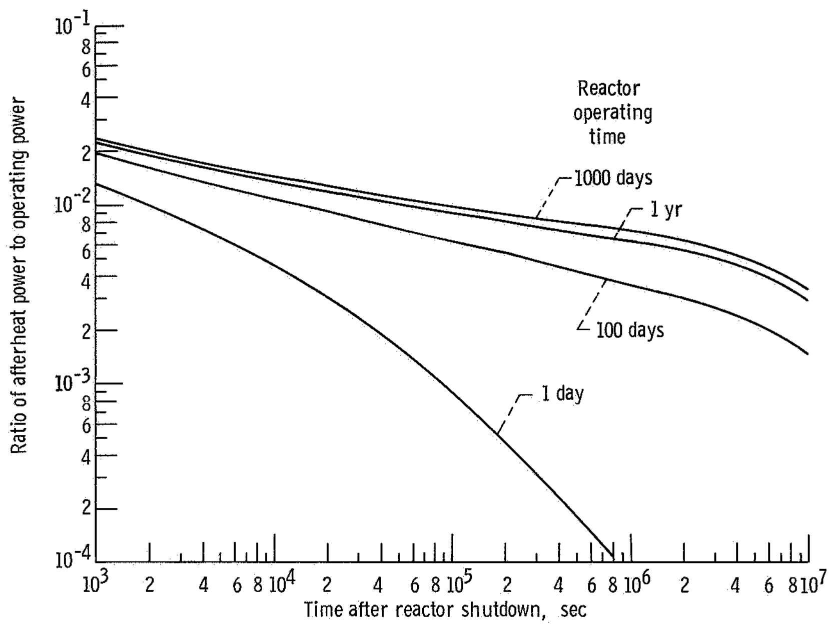

Figure 17. - Ratio of reactor afterheat power to operating power (2.17 MW or 7.4×10^4 Btu/hr) for long times after shutdown.

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