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CRITICALITY STUDY OF NASA SOLUTION REACTORS WITH 25.4-CENTIMETER-DIAMETER CYLINDRICAL STAINLESS-STEEL TANKS

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16. Abstract

The NASA solution reactors ZPR-I and ZPR-II have been modified to use 25.4-cm-diameter cylindrical cores. The fuel solution consists of enriched (93.2 percent U^{235} uranyl fluoride salt (UO_2F_2) dissolved in water and is variable in concentration. Bare-core criticality data from tests with both reactors are presented and compared with multigroup transport calculations for a range of fuel concentrations having hydrogen to uranium atom ratios from 73.7 to 280.5. Parameters studied include critical mass, fundamental prompt-mode decay constant, temperature coefficient, and incremental reactivity worth at delayed critical.

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SUMMARY

The NASA Zero Power Reactors I and II (ZPR-I and ZPR-II) have been modified and presently have 25.4-centimeter-diameter cylindrical core tanks. They are fuel solution reactors using highly enriched (93.2 percent U^{235}) aqueous uranyl fluoride ($UO_2F_2 + H_2O$) that can be varied in concentration. ZPR-I is designed primarily for radial reflector criticality studies and leakage spectrum measurements for the purpose of checking calculational models of various reflector materials. For this purpose, ZPR-I is constructed with a minimum of surrounding structural or scattering material. ZPR-II is a reactor that is used primarily as a radiation source for the study of neutron and gamma attenuation in surrounding layered shielding media or streaming through penetrations in proposed shielding configurations.

This report briefly describes the two reactor systems and presents bare-core data obtained from tests using both reactors. These data include critical mass, fundamental prompt-mode decay constant, temperature coefficient, and incremental reactivity worth at delayed critical for a range of fuel concentrations having hydrogen to uranium-235 atom ratios from 73.7 (335 kg of U^{235} per cubic meter of fuel solution) to 280.5 (91.7 kg of U^{235} per cubic meter). Multigroup transport calculations (S_4P_1 approximation) were made for the same range of fuel concentration and the results are compared with experiment.

INTRODUCTION

The NASA Zero Power Reactor Facility (ZPRF) incorporates two solution-critical facilities designated as Zero Power Reactor I (ZPR-I) and Zero Power Reactor II (ZPR-II). The fuel used to operate both reactors is an aqueous solution of highly enriched (93.2 percent U^{235}) uranyl fluoride salt (UO_2F_2). These reactors were conceived, designed, and built to provide a high degree of operational flexibility in antici-

pation of a wide range of possible needs. The solution fuel provides the capability of achieving a wide range of fuel concentrations. The two reactor systems permit widely varying programs to be operated concurrently and make possible the modification of one reactor system without necessarily interrupting the operation of the second reactor.

A description of the ZPRF as it existed in 1965 is presented in reference 1. The ZPR-II was modified in 1969 and the ZPR-I in 1970 so that both reactors have new-geometry cores. Although both reactors now have 25.4-centimeter-inside-diameter cylinders for core tanks, the neutronics are not exactly identical and the kinds of experiments performed are entirely different.

Whenever a new reactor core is installed, it is essential that certain calibration experiments be conducted. Such experiments are important to the overall experimental program that is to follow. In addition, some of the parameters must be measured in order to meet various regulatory obligations required under the facility licenses.

In the present experimental program, the ZPR-II has been used essentially as a neutron and gamma ray source to surrounding media. There have been few changes in the reactor core and only a limited amount of calibration data has been required. On the other hand, the ZPR-I experimental program is such that measurement of basic parameters over as wide a range of fuel concentrations as possible is advantageous. To this end, tests have been made at five fuel concentrations covering most of the range physically possible with a 25.4-centimeter-diameter core. By using these data it is possible to infer the necessary parametric values at any intermediate fuel concentration that is required.

The post-neutron experiments with solution-critical systems include determination of critical mass, temperature coefficient, and reactivity worth of an incremental fuel addition at delayed critical. Such items as absolute power calibration and safety rod worths are also measured but are not described in this report. Pulsed-source measurements have been added in recent years for several reasons. The pulsed-source technique has been found to be a useful experimental tool (ref. 2) to measure a sensitive universal parameter, the fundamental prompt-mode decay constant. Since this parameter is readily calculable, an additional method of comparing experiment and analysis is now available.

Calculations have been made at each of the five fuel concentrations involved in this study. The analytical parameters compared to experiment are critical mass and fundamental prompt-mode decay constant. Good agreement between experimental and calculated results for these basic configurations is a prerequisite for the future radial reflector studies to test neutron cross sections and calculational models.

This report provides a brief general description of the ZPRF. The experiments and calculations are each described in enough detail that reference to other sources should not be required. The experimental data are presented so as to be useful directly in operating situations.

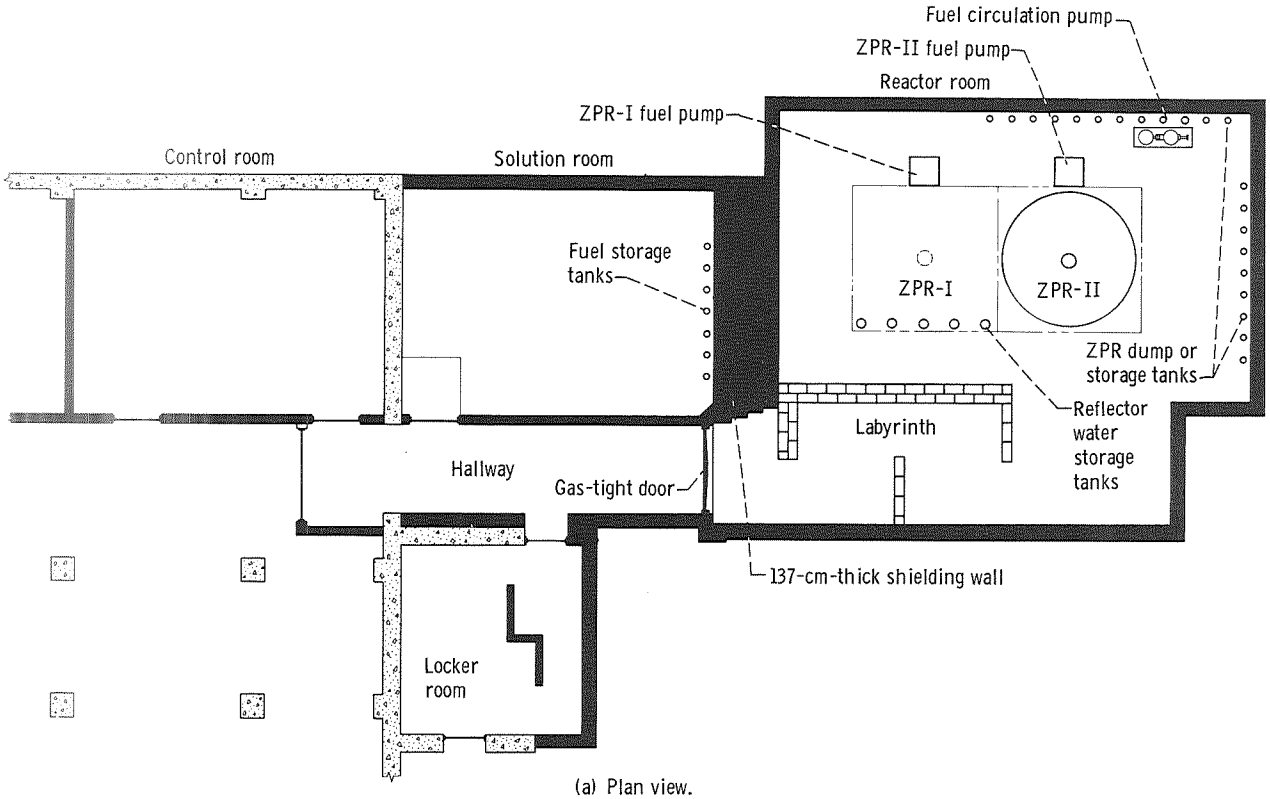
DESCRIPTION OF NASA ZERO POWER REACTOR FACILITY

The NASA Zero Power Reactor Facility includes two solution-critical facilities designated as ZPR-I and ZPR-II. The layout of the facility and the current reactor arrangement are shown in figure 1. The facility proper includes a control room, reactor room, solution room, locker room, and connecting hallway. The reactor room is constructed of reinforced poured concrete and is located under an earth mound to minimize radiation leakage from the facility. A thick (137 cm) wall is provided between the reactor room, and the solution room to permit personnel access to the solution room at all times.

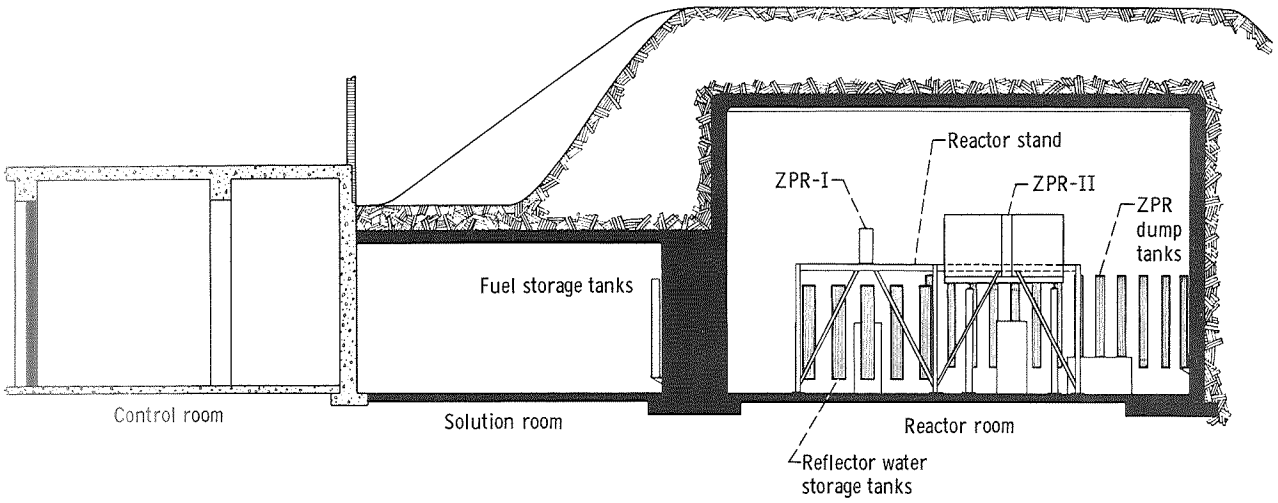
As a result of recent changes, both reactor systems presently have 25.4-centimeter-inside-diameter, cylindrical, stainless-steel reactor core tanks. The reactors are located approximately at epicenters of the reactor room and about midway between the floor and the ceiling. These locations are used to minimize wall effects within the limitations of the reactor room. The ZPR-I is mounted on the stand which makes up the working area for personnel and equipment at about the mid-level of the reactor room. Special effort has been made to keep structural materials to a minimum around ZPR-I. The ZPR-II, on the other hand, is mounted on a pedestal-like stand that is separated from the rest of the mid-level structure. This pedestal is designed to hold a load of 36 000 kilograms to accommodate the heavy materials used in connection with shielding experiments. The ZPR-II core tank is mounted on the axis of a 2.44-meter-diameter tank which can be used to surround the core radially with water, solid materials, or various combinations of materials in either wet or dry configurations.

This arrangement demonstrates again the flexibility possible with two reactor systems. In the past, two different-geometry-core tanks were used. This arrangement uses the same geometric core configuration but two very different modes of operation. The ZPR-I is designed to carry out studies associated with the reactor core itself; that is, critical mass studies, reflector effect studies, and the like. The ZPR-II is being used in a manner wherein a fixed-core configuration has been assembled and is being used as a neutron source to study neutron and gamma ray spectra in the surrounding media. The primary reflector region is rarely changed and the core is unperturbed by any changes occurring outside the primary reflector.

Except for the differences resulting from different experimental objectives, the ZPR-I and ZPR-II are nearly similar solution-critical systems. Each consists of a geometrically safe fuel storage system, a reactor tank where various critical configurations may be assembled, and a solution manipulating system that enables the experimenter to assemble the critical reactor remotely in a controlled manner. The assembly takes place with the reactor room closed to provide a sealed test cell. This pro-



(a) Plan view.



(b) Elevation view.

Figure 1. - Zero power reactor facility.

vides both adequate shielding from direct radiation and protection from the airborne activity associated with the operation of such systems.

The reactors share a single control console and instrumentation system to assure operation of only one reactor at a time. The fuel piping systems for the two reactors are kept separate and independent although they can be connected by opening the proper valves. In addition, both are connected to the solution room storage system and related equipment.

The reactor room fuel storage consists of a total of 21 polyethylene tanks with 12.7-centimeter inside diameters located on 45.72-centimeter centers along part of the east and north walls (see fig. 1(a)). The 12 tanks on the north wall are the ZPR-I dump tank system while the nine tanks on the east wall make up the ZPR-II dump tank system. Each reactor system has its own positive-displacement-type pump to transfer fuel between the dump tanks and the reactor vessel. The only reactor control is the fuel height manipulation possible with this reversible and variable-speed pump, although safety devices in the form of a cadmium safety blade and a dump valve are incorporated in each system.

The reactor vessels are constructed of 304 stainless steel and each is approximately 78.7 centimeters tall and 25.4 centimeters in diameter. The ZPR-I vessel walls and bottom are about 0.16 centimeter (1/16-in.) thick; and the ZPR-II walls and bottom are about 0.32-centimeter (1/8-in.) thick. Both tanks are coated with Glyptal to minimize the contamination of the fuel caused by fuel etching the stainless-steel containers. All components used in the fuel systems are constructed of materials that have been selected to minimize corrosion by the fuel solution. The fuel storage system uses polyethylene tanks, the piping and valves are of polyvinyl chloride, and Teflon bellows are used on the fuel pump. Any item that is likely to come in contact with the fuel is either constructed of corrosion-resistant material or protected by an appropriate type of coating.

The height of the fuel solution is measured by a direct-current probe arrangement. This probe is attached to a precision lead screw that is motor driven. Control of the probe and readout takes place remotely at the control console. Relative fuel heights can be read to less than 0.001-centimeter change in height. The tolerances on tank dimensions are such that the accuracy of absolute height measurements is probably of the order of 0.025 centimeter.

The fuel solution temperature is measured by an iron-constantan thermocouple swaged in a stainless-steel jacket for protection from corrosion. The reference for the thermocouple is an oven-type thermocouple junction. The output from the thermocouple is measured with a precision potentiometer and galvanometer to the nearest 0.005° C. These are relative temperature measurements since the thermocouples are not calibrated for absolute measurements.

EXPERIMENTAL PROGRAM AND PROCEDURES

The experimental program reported herein includes several measurements made with clean bare-core configurations at delayed critical. The study covers a wide range of fuel concentrations and includes data from both ZPR-I and ZPR-II although most of the work was carried out with ZPR-I. Parameters measured include critical mass, temperature coefficient, incremental reactivity worth, and the fundamental prompt-mode decay constant. The basic measurement involved was the determination of the critical mass of uranium-235 in a 25.4-centimeter-diameter cylindrical tank. Once this was achieved it was possible to compile the remainder of the data by making various controlled perturbations of the steady-state critical condition.

Critical Mass

An accurate experimental determination of critical mass requires the accurate determination of several variables. These include fuel concentration, fuel volume, fuel temperature, and finally the determination of the steady-state delayed-critical condition itself. The determination of the steady-state critical condition is the basic operation around which the other measurements are made. The reactor operation and the experimental procedures are outlined briefly in the paragraphs that follow. They are described in greater detail in reference 1.

The initial step in a typical critical experiment is the preparation of the desired fuel concentration. The desired fuel concentration can be achieved by starting with any given available mix and adding or removing water, or by mixing different fuel concentrations if that is more convenient. Water removal is the more difficult and time consuming and is accomplished by use of a vacuum evaporator-condenser apparatus. Regardless of how the desired concentration is achieved, care must always be used to assure adequate mixing of the fuel since a minimum of three storage bottles and a long transfer line are always a part of the system.

Determination of the final mixed concentration is usually made by using a fuel sample obtained immediately after the experiment of interest has been completed. This minimizes the chance of change due to water evaporation from the fuel. Two methods are used at the ZPRF for determination of fuel concentration. The first method involves a very careful measurement of the solution density. The method most often used is the precision hydrometer that gives a maximum error of about ±0.1 percent in solution density. Calibrated volumetric flasks can improve on this precision by about a factor of 3, but require considerably more time and effort to use. A second method of determining fuel concentration makes use of a type of chemical analysis known as gravimetric analy-

sis. Both methods of measuring fuel concentration are discussed in greater detail in reference 1.

The startup and operating procedures for these reactors are typical of those used for solution-critical systems. The fuel is assembled in a controlled manner in the presence of a neutron source. In this case a 1-curie plutonium-beryllium neutron source is located externally at the reactor tank bottom. With each increment of fuel addition there is an increase in the neutron flux due to multiplication of the original source neutrons. When the critical condition is approached closely (i.e., within a few centimeters of the critical height), the actual critical height can be predicted quite accurately from multiplication measurements. For most experimental arrangements, the multiplication must be of the order of 2 or more to have meaning, and a good prediction of critical height is generally obtainable with a multiplication of the order of 10 or more. When the reactor reaches criticality, the neutron source is withdrawn from the vicinity of the core and returned to its storage container to ascertain if the system is truly self-sustaining.

In order to determine the steady-state delayed-critical condition accurately, a constant self-sustaining neutron level is maintained for 15 minutes or longer in an effort to assure that all factors which contribute to nonequilibrium in the system have been allowed to stabilize. These factors include surface waves, structure vibrations, reactor system kinetics, and gaseous voids due to gas evolution and pump agitation. The magnitude of the neutron level used is such that the signal-to-noise ratio is high on the associated control instrumentation. The power level providing such a condition will vary depending on the particular configuration and the immediate electrical noise conditions, but is is generally no greater than 1 watt thermal.

The fuel volume is determined from the fuel height measured at delayed critical using the probe arrangement described in the previous section. Measurement of the fuel temperature is made using the thermocouple arrangement also described in the previous section. The temperature is important for two reasons as is discussed in the section "Temperature and Temperature Coefficient" that follows. It is sufficient to note here that determination of solution concentration, volume, and temperature at the delayed critical condition permits a determination of the critical mass of U-235 that is accurate enough for most purposes. Several smaller effects such as water evaporation from the fuel and the neutron interaction of the reactor with its environment must be considered but usually do not have a significant effect on critical mass.

Temperature and Temperature Coefficient

The need to measure temperature is associated with two interrelated properties

essential to the critical mass determination. These properties are fuel solution density and temperature coefficient of reactivity. The general practice at ZPRF is to correct all data to a single temperature of 20° C. To this end, all solution density measurements are made at 20° C, and all critical mass studies are corrected to this temperature. To correct the critical height to the base temperature, a study has been made of temperature coefficients at each of the fuel concentrations considered. The temperature span used in the measurements was of the order of 5° C.

Both reactors are operated at very low power and therefore undergo no temperature change as the result of heat generation within the core. The variation of fuel temperature is caused by the change in the reactor room air temperature. Since the reactor room is underground, this temperature change is primarily a matter of seasonal rather than short-term changes. The reactor room is provided with a heating system and an exhaust fan for ventilation but not with any air-contitioning equipment. The general practice has been to achieve conditions that will give short-time periods at nearly constant temperature and to let the long-term variations occur naturally. By measuring the temperature and knowing the temperature effects, it is possible to make the necessary corrections and eliminate the need for any closer control of the environment.

The actual experimental measurement of temperature coefficient makes use of the effect of room air temperature on fuel solution temperature to get the necessary fuel solution temperature changes. The general procedure is to raise the room air temperature enough to achieve a fuel temperature increase of about 5° C and then to determine the critical height at this fuel temperature. The fuel temperature is then reduced and the critical height determined at the lower temperature. It has been found that the temperature coefficient for a given fuel concentration is a constant over the temperature range involved within the accuracy of the measuring technique.

Incremental Reactivity Worth

The sensitivity of the reactor to small changes in fuel height near delayed critical is important to safe operation. Compliance with the licenses of both reactors limits the reactivity addition rate to 20 cents per second. The incremental reactivity worth can be related to a maximum fuel addition rate which in turn determines the maximum pump speed permissible for a given configuration.

The reactivity worths associated with small changes in the core height at criticality have been determined by making use of the inhour formula

$$\rho = \frac{l}{Tk_{eff}} + \sum_{i=1}^{6} \frac{\beta_i}{1 + \lambda_i T}$$

where

ρ reactivity

l mean prompt neutron lifetime, sec

T reactor period, sec

k_{eff} effective multiplication factor

β_i fraction of total number of fission neutrons belonging to i^{th} delay group

λ_i radioactive decay constant for that group, sec^{-1}

The technique involves perturbing the steady-state condition of the delayed-critical system by a small amount and measuring the associated period. Fuel additions resulting in positive periods are used at the ZPRF although both positive and negative periods can be used. From a series of small fuel additions, each of which has a positive period and related reactivity, it is possible to get the reactivity worth per incremental change in

TABLE I. - DELAYED-NEUTRON DATA

FOR URANIUM-235

[From ref. 3, p. 90; $\beta = 0.0065$.]

Delayed group	Decay constant, λ, sec^{-1}	Relative abundance, β_i/β	Delayed-neutron fraction, β_i
1	0.0124	0.033	0.000 214 5
2	.0305	.219	.001 423 5
3	.111	.196	.001 274 0
4	.301	.395	.002 567 5
5	1.14	.115	.000 747 5
6	3.01	.042	.000 273 0

fuel height. The values of β_i and λ_i used are shown in table I and are from reference 3 where the delayed-neutron fraction β equals 0.0065. The incremental worths reported herein are in units of cents, which eliminates the need of relating reactivity to the effective delayed-neutron fraction $\overline{\beta}_o$ ($\overline{\beta}_o$ = 100 cents regardless of its value). The system was perturbed such that periods longer than 200 seconds were involved. For this range of periods the first term in the inhour equation can be neglected and the reactivity as a function of fuel height increase is linear.

Fundamental Prompt-Mode Decay Constant

The use of the pulse-neutron-source technique with NASA solution reactors has been reported in references 2 and 4. In pulsed-source experiments a burst of neutrons of very short time duration is injected into the configuration being studied and the time behavior of the prompt neutron level is measured. In this particular study the critical configuration at each fuel concentration was pulsed by a burst of 14-MeV neutrons and the neutron level was measured with a 1/v detector and a multichannel analyzer used in a multiscalar mode. The prompt neutron level following the neutron burst decays in an exponential fashion until delayed neutron background is reached. The exponential decay value is known as the fundamental prompt-mode decay constant $(\alpha_o^p)_E$, where E denotes experimental. This is a very sensitive parameter which varies both as a function of fuel concentration (material buckling) and the reactivity of the system (geometric buckling).

Results of a study in which both fuel concentration and system reactivity were varied over wide ranges are presented in reference 4. In this previous study, the range in fuel concentration covered a range of hydrogen to uranium-235 atom ratios (H/X) from 510 to 1600 using a single (76.2-cm diam) reactor vessel. In the study reported herein, delayed-critical configurations have been pulsed over a range of concentrations with H/X values from about 74 to 280. No subcritical systems were pulsed during these tests.

MULTIGROUP TRANSPORT CALCULATIONS

Multigroup transport calculations were made of the critical cases to obtain calculated values of the critical size and of the fundamental prompt-mode decay constant α_o^p. The basic S_n transport calculation is discussed fully in appendix B of reference 2. Briefly, one-dimensional axial calculations to obtain both the fundamental prompt-mode flux and the static adjoint flux were made for these cylindrical reactor systems. The S_4 approximation to the angular flux and the P_1 approximation to the anisotropic elastic neutron scattering were used to treat the neutron leakage from the ends of the reactors. The actual radial dimension of the core is 12.70 centimeters but was adjusted to an effective value of 12.86 centimeters to include the 0.16-centimeter wall thickness of the bare core stainless-steel container. This constant effective value of bare core radius was found to achieve good agreement between the calculated and measured heights at delayed critical for the reactor system having an H/X of 237. This value was then used for the effective radial dimension for all reactors. The 25.4-centimeter-diameter core proved to be so sensitive to radial dimension changes that the bare-core radial and axial buckling interation technique employed in reference 4 was not successful. The

stainless-steel core containers were not explicitly represented in these calculations but do have an important criticality effect on these cores since they act as thin reflectors.

A total of nine energy groups were used, eight fast and one thermal group. The fast group cross sections were obtained using the GAM-II code (ref. 5) while the thermal cross sections which included an upscattering transfer component were obtained using the GATHER-II code (ref. 6). The energy boundaries for each neutron energy group are given in table II. The delayed-neutron data are the same as were used in the experimental inhour measurements and given in table I. The average energy of the delayed neutrons varies from about 0.25 MeV to about 1 MeV (ref. 3, p. 95) and therefore the spectra for each delayed group were taken to be in energy group 4. The atom densities used in these calculations are given in table III and are based on an experimental determination of the uranium content.

TABLE II. - NEUTRON ENERGY GROUP SPLIT

Group	Energy	Lethargy
1	14.92 MeV to 4.97 MeV	-0.4 to 0.7
2	4.97 MeV to 2.231 MeV	0.7 to 1.5
3	2.231 MeV to 0.821 MeV	1.5 to 2.5
4	0.821 MeV to 0.224 MeV	2.5 to 3.8
5	0.224 MeV to 9.12 keV	3.8 to 7.0
6	9.12 keV to 454 eV	7.0 to 10.0
7	454 eV to 8.32 eV	10.0 to 14.0
8	8.32 eV to 0.414 eV	14.0 to 17.0
9	0.414 eV to 0 eV	17.0 to ∞

TABLE III. - ATOM DENSITIES FOR UO_2F_2-H_2O FUEL SOLUTIONS

	Hydrogen to uranium-235 atom ratios, H/X				
	73.7	144.0	200.9	237	280.5
	Atom density, atoms/barn-cm				
N^H	0.063438	0.065095	0.065573	0.065758	0.065915
N^O	.033566	.033517	.033487	.033474	.033462
N^F	1.8469×10^{-3}	9.6977×10^{-4}	7.0045×10^{-4}	5.9523×10^{-4}	5.0425×10^{-4}
$N^{U^{235}}$	8.6075×10^{-4}	4.5196×10^{-4}	3.2644×10^{-4}	2.7741×10^{-4}	2.3500×10^{-4}
$N^{U^{238}}$	6.271×10^{-5}	3.292×10^{-5}	2.378×10^{-5}	2.021×10^{-5}	1.712×10^{-5}

RESULTS AND DISCUSSION

Using the methods discussed, experiments and calculations have been performed at the delayed-critical condition for a range of fuel concentrations in the 25.4-centimeter-diameter cores. Four parameters were determined experimentally: the critical mass (critical height), the incremental reactivity worth, the temperature coefficient, and the fundamental prompt-mode decay constant $(\alpha_o^p)_E$. The key parameters obtained from the comparable calculations were the fundamental prompt-mode decay constant α_o^p, the critical height, the neutron generation time Λ_o, and the effective delayed-neutron fraction $\overline{\beta}_o$. The ratio of $\overline{\beta}_o/\Lambda_o$ should give the same value at delayed critical as is obtained for $-\alpha_o^p$ in the calculations. These parameters are listed in table IV and are discussed individually in the paragraphs that follow.

TABLE IV. - PARAMETERS AT DELAYED CRITICAL AS FUNCTION OF FUEL SOLUTION CONCENTRATION FOR ZPR-I

Parameter	Hydrogen to uranium-235 atom ratio, H/X				
	73.7	144	200.9	237	280.5
Experimental critical height (0.16-cm wall), cm	32.74	37.16	44.08	50.92	66.63
Calculated critical height (r_{eff} = 12.86 cm), cm	33.25	37.00	43.96	51.20	66.45
Experimental critical height (0.32-cm wall)[a], cm	31.30	35.08	41.14	46.84	58.53
Experimental critical mass (0.16-cm wall), kg	5.563	3.318	2.841	2.788	3.091
Calculated critical mass, kg	5.649	3.303	2.833	2.804	3.083
Experimental critical mass (0.32-cm wall)[b], kg	5.318	3.132	2.651	2.565	2.715
Experimental prompt-mode decay constant $(\alpha_o^p)_E$, [c] sec^{-1}	-1090±12	-570±2	-422±2	-367±5	-317±3
Calculated prompt-mode decay constant α_o^p, sec^{-1}	-1192	-605	-448	-385	-334
Temperature coefficient (experimental), mm/°C	-0.512	-----	-0.997	-1.76	-2.96
Incremental reactivity worth (experimental), cents/mm	-7.48	-6.26	-3.62	-2.83	-1.34
Temperature coefficient (using fitted data), cents/°C	-3.87	-4.07	-4.31	-4.27	-3.87
Neutron generation time (calculated), Λ_o, μsec	7.09	13.40	18.15	21.00	24.34
Effective delayed-neutron fraction (calculated), $\overline{\beta}_o$	0.008292	0.008270	0.008213	0.008170	0.008116
$\overline{\beta}_o/\Lambda_o$, sec^{-1}	1169.5	617.2	452.5	389.0	333.4

[a] ZPR-II critical height data: 33.11 cm at H/X of 115; 50.62 cm at H/X of 252.
[b] ZPR-II critical mass data; 3.675 kg at H/X of 115; 2.609 kg at H/X of 252.
[c] Errors are statistical errors in data, not experimental error limits.

Critical Mass

Critical mass or core height determinations were made with the ZPR-I at fuel concentrations having H/X values of 73.7, 144, 200.9, 237, and 280.5. It was observed that the data measured with ZPR-I did not agree with previously measured critical height data taken with ZPR-II at H/X values of 115 and 252. The differences in critical core height proved to be primarily due to the difference between the stainless-steel reactor vessel wall thicknesses. By adding an additional 0.16-centimeter stainless-steel sleeve to the 0.16-centimeter-thick ZPR-I vessel wall, it was possible to duplicate the 0.32-centimeter wall thickness of ZPR-II. Very good critical height agreement was then obtained between the ZPR-I and the ZPR-II data, as shown by the results in figure 2. The 0.16-centimeter stainless-steel sleeve was employed with each of the five fuel concentrations. The critical heights and the critical masses are compared in figure 2, both as a function of fuel solution atom ratio (H/X).

The critical heights and critical masses measured in this study are in good agreement with previous measurements made at Oak Ridge, as presented in figure 2. According to the description of the experiments reported in reference 7 the earlier measurements should be compared to the present ZPR-I data because of similar tank wall thicknesses. Considering the many factors that could cause differences between the early and present experiments, the agreement is excellent for a core so sensitive to absolute radial dimension.

The results of the multigroup calculations are also presented in figure 2. As stated earlier, an effective radial dimension of 12.86 centimeters was selected to include the effect of the core container and to provide good agreement between the calculated and measured height at delayed critical for the ZPR-I case with a fuel solution having an H/X of 237. This dimension for the effective radius was subsequently used for all fuel concentrations and provides good agreement over the range of the fuel concentrations involved in these studies.

Comparison of the two experimental critical height curves demonstrates the great sensitivity of this small-diameter core to small changes in the radial dimension resulting from the change in radial reflection effect of the core containers. For example, at an H/X of 237, the 0.16-centimeter increase in tank wall thickness caused a decrease of 4.0 centimeters in a fuel height of 50.92 centimeters at delayed critical, or $\Delta H/\Delta r = -25.5$ (the negative sign indicating that an increase in one dimension causes a decrease in the other). A comparison with one-dimensional axial calculations at the same fuel concentration was made where the reduction in fuel height ΔH as a function of the change of effective radius Δr_{eff} has been calculated to be -26.2 over the same increment of radial change. Although the calculations do not specifically include the effect of the stainless-steel core container, the experimental and calculated coefficients are

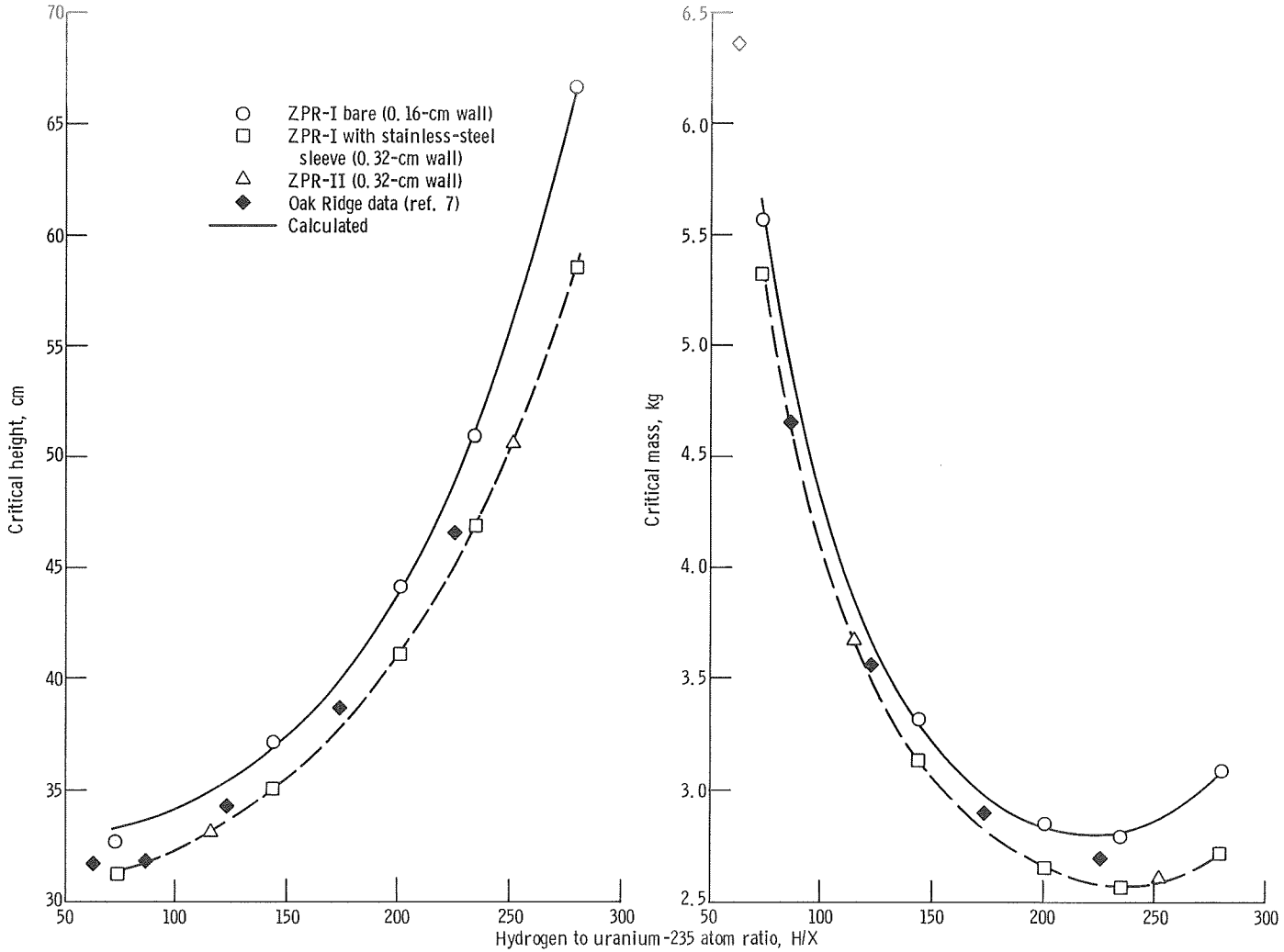


Figure 2. - Critical height and critical mass as function of fuel solution concentration.

numerically in good agreement, suggesting that the reflector effect of the container and the change in radius are linearly related. The calculated coefficient is not linear, however; and a further reduction in radius to the equivalent of no container wall results in an increase in fuel height of 4.84 centimeters, or a $\Delta H/\Delta r_{eff} = -30$ for the additional 0.16-centimeter reduction in radius.

Fundamental Prompt-Mode Decay Constant

A comparison of the experimental and calculated values of the fundamental prompt-mode decay constant α_o^p in seconds^{-1} as a function of the hydrogen to uranium 235 atom ratio is presented in figure 3 for the delayed-critical, 25.4-centimeter-diameter cores of ZPR-I. The calculated values are greater in absolute magnitude than the experimental values in all cases, with the percentage difference being greatest for the most concentrated solution (lowest H/X). The differences varied from about 4 to 5 percent at lower concentration to about 9 percent at the rich end. It was noted that the results from the pulse experiments were very sensitive to incremental deviations from the delayed-critical condition.

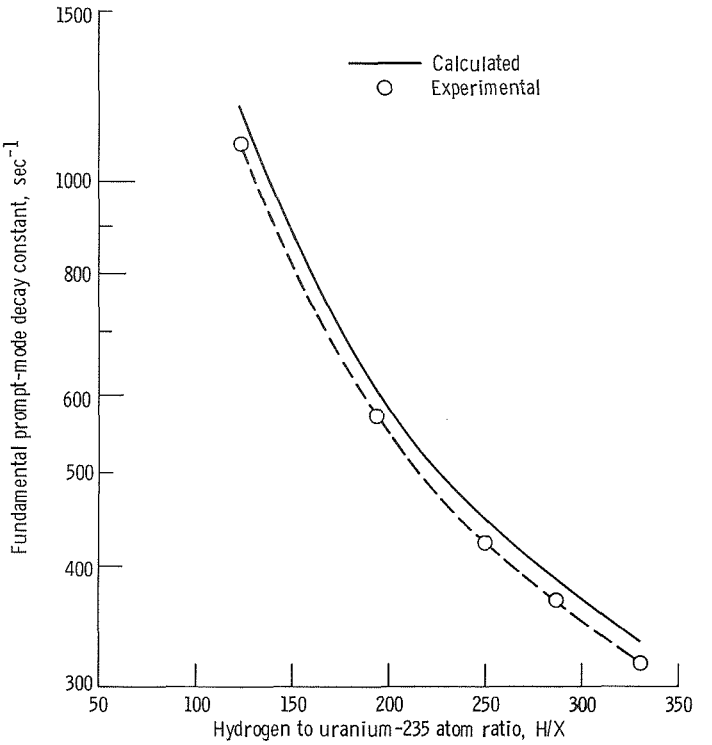


Figure 3. - Fundamental prompt-mode decay constant as function of fuel solution concentration - ZPR-I.

Whereas the addition of the 0.16-centimeter stainless-steel sleeve resulted in a large reduction in the critical height at all fuel concentrations, the observed values of $(\alpha_o^p)_E$ were not changed by the added radial thickness within the accuracy of the measurement. The values of $(\alpha_o^p)_E$ measured vary more as a function of the fuel concentration or material buckling than as a function of reactor geometry. Evidence of this can be observed when the data measured using a 76.2-centimeter-diameter core (ref. 4) are combined with the data from the 25.4-centimeter core, as shown in figure 4. The

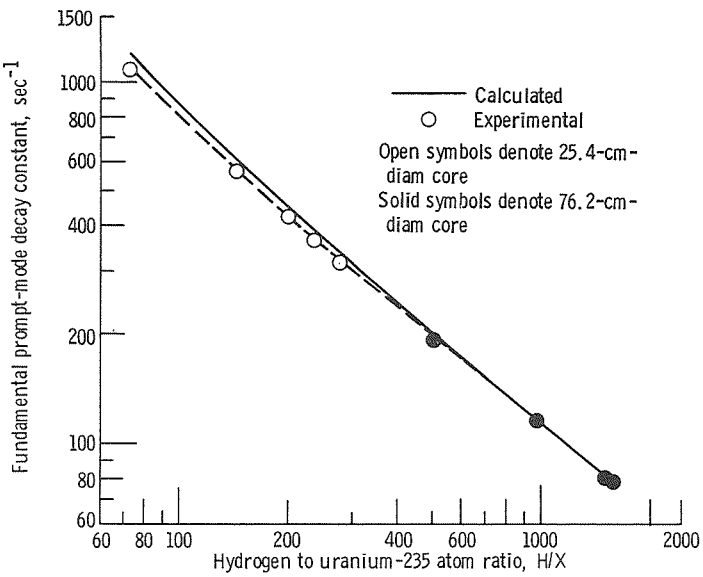


Figure 4. - Fundamental prompt-mode decay constant as function of fuel solution concentration.

log-log plot has been used to permit easy comparison of the widely ranging data. This plot indicates that the increasing differences between experimental and calculated values of α_o^p found at the richer concentrations are an extension of a trend that was just starting to become observable in the earlier work.

Incremental Reactivity Worth

For solution reactors of cylindrical geometry the most convenient method of measuring small changes in system reactivity as a function of geometry is by measuring the change in solution height. Each of the five fuel concentrations studied was calibrated in terms of reactivity change per unit incremental increase in fuel height. These results are plotted in figure 5 and tabulated in table IV. The units of reactivity are cents based on the use of $\beta = 0.0065$ in the inhour equation; however, for the long reactor periods

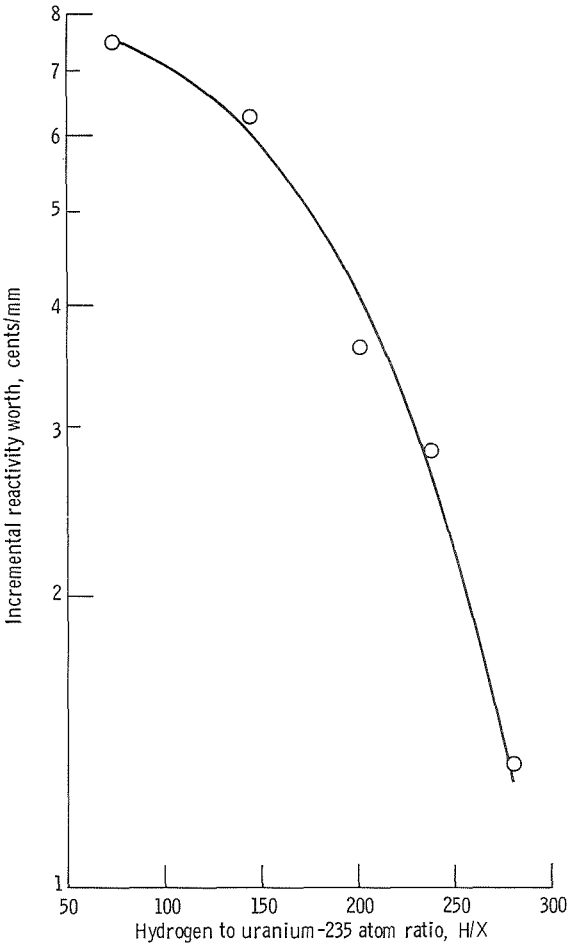


Figure 5. - Incremental reactivity worth as function of fuel solution concentration - ZPR-I.

measured, the reactivity in cents is independent of the value of β that is used.

The experimental error based on previous measurements is of the order of ±10 percent. Only one determination was made at each fuel concentration during this study. The curve drawn in figure 5 is an eye fit to the experimental points and provides reactivity data for all fuel concentrations within the range of measurement.

Temperature Coefficient

The experimental values for temperature coefficient, the change in core height per unit change in fuel solution temperature in millimeters per ^{o} C, are plotted in figure 6 as a function of the fuel solution atom ratio and are tabulated in table IV. The curve represents a reasonable fit to the experiment points, which have an accuracy of the order of ±10 percent.

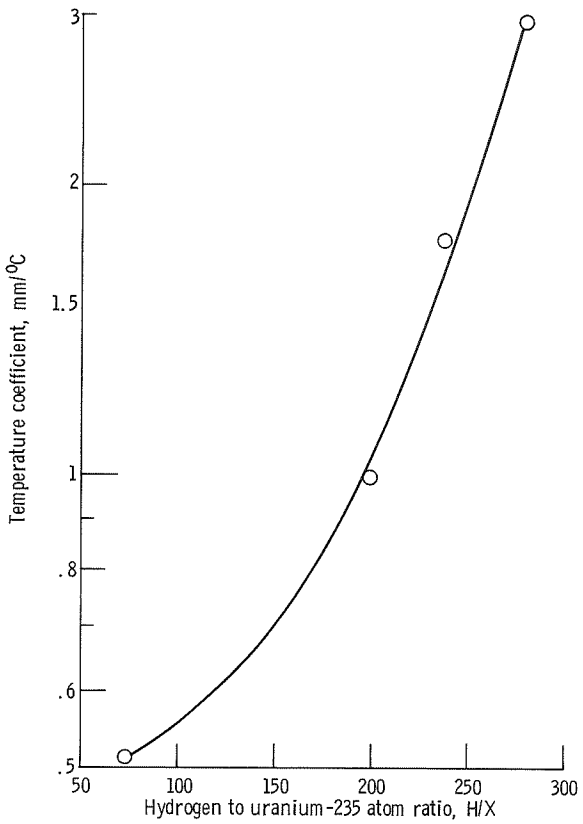


Figure 6. - Temperature coefficient as function of fuel solution concentration - ZPR-I.

Greater accuracy is possible for this measurement but it would require an expenditure of time and effort not commensurate with present needs. The experimental procedures are such that critical configurations are generally run within a few degrees of the desired 20° C so that the correction to the critical height associated with temperature is small. A negative sign implies that an increase in temperature results in a loss of reactivity.

In order to present the temperature coefficient in units of reactivity, the incremental reactivity worth must be factored in for each concentration. By using the data in figures 5 and 6, the temperature coefficient is found to be nearly constant at an average value of -4.1±0.2 cents per °C over the range of concentrations measured. When the calculated effective delayed-neutron fraction $\overline{\beta}_o$ is used, the temperature coefficient in terms of the overall neutron multiplication factor K is of the order of -3.4×10^{-4} ΔK per °C. This is in good agreement with prior measurements (e.g., ref. 1). The use of a single value is not meant to imply that the temperature coefficient is constant, but that it is relatively insensitive to fuel concentration changes over the range reported herein.

Calculated Parameters

The variation of other reactor calculated parameters as a function of fuel concentration is shown in figure 7. The neutron generation times Λ_o, the effective delayed-neutron fraction $\overline{\beta}_o$, and the ratio $\overline{\beta}_o/\Lambda_o$ are each plotted against fuel solution hydrogen to uranium-235 atom ratio. The curves shown are generated from calculated values at each of the five fuel concentrations that were used experimentally. These data are also tabulated in table III.

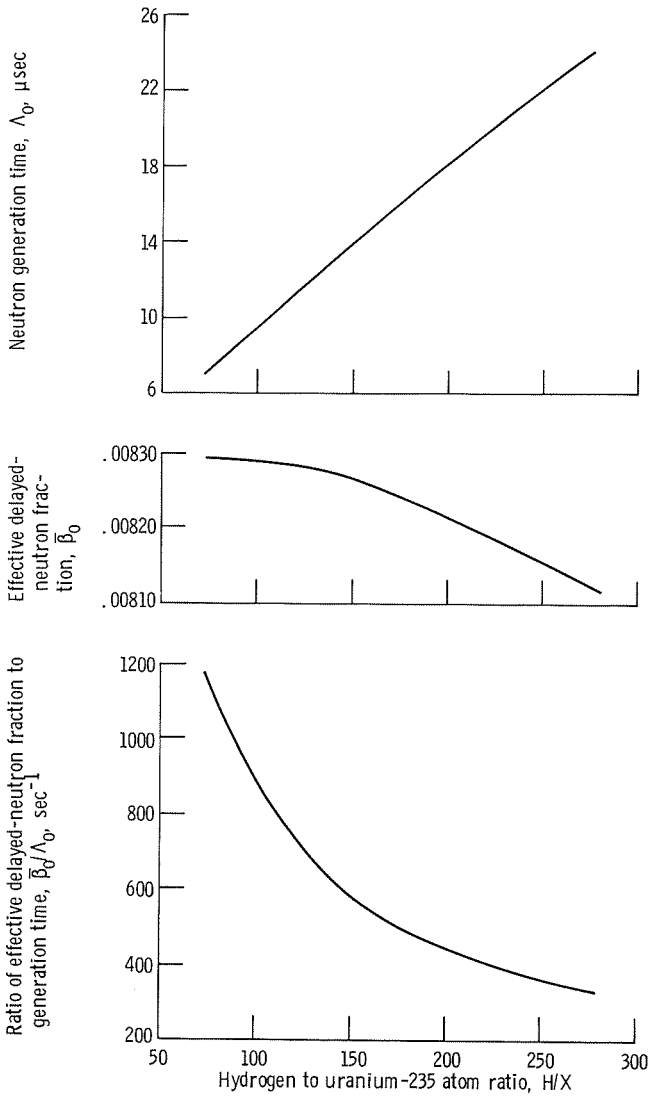


Figure 7. - Variation of calculated reactor parameters as function of fuel solution concentration for bare 25.4-centimeter-diameter reactors.

The neutron generation time is seen to vary from about 7 to 24 microseconds in a nearly linear fashion over the range of fuel concentration. This is nearly inversely proportional to the associated fuel density or the uranium-235 atom density. The effective delayed-neutron fraction varies only a few percent for the same range of fuel variation. The ratio $\overline{\beta}_o/\Lambda_o$ varies from about 1200 to 300 seconds^{-1} for the fuel concentration range studied.

CONCLUSIONS

Several conclusions can be made regarding the 25.4-centimeter-diameter cylindrical UO_2F_2-H_2O solution reactor cores as a result of this study:

1. The critical height of the 25.4-centimeter-diameter configuration is very sensitive to compositional changes in the radial direction and is a desirable core with which to conduct radial reflector experiments.

2. The fundamental prompt-mode decay constant at delayed critical varies consistently as a function of fuel concentration and is independent of geometric configuration.

3. The temperature coefficient of reactivity is strongly negative for all fuel concentrations of the 25.4-centimeter-diameter cores and is nearly a constant of the order of -3.4×10^{-4} ΔK per ^{o}C in reactivity.

4. The effective bare-core radius required by the one-dimensional axial calculations at one fuel concentration to achieve agreement with experiment gave good agreement between experiment and calculation at all concentrations.

Lewis Research Center,
National Aeronautics and Space Administration,
Cleveland, Ohio, July 6, 1971,
120-27.

REFERENCES

1. Fox, Thomas, A.; Mueller, Robert A.; Ford, C. Hubbard; and Alger, Donald L.: Critical Mass Studies with NASA Zero Power Reactor II. I - Clean Homogeneous Configurations. NASA TN D-3097, 1965.

2. Fieno, Daniel; Fox, Thomas A.; Mueller, Robert A.; and Ford, C. Hubbard: Analysis and Experiments with a Pulsed Neutron Source for an Unreflected Solution Reactor up to $50 Shutdown. NASA TN D-5487, 1969.

3. Keepin, G. Robert: Physics of Nuclear Kinetics. Addison-Wesley Publ. Co., 1965.

4. Fox, Thomas A.; Mueller, Robert A.; and Fieno, Daniel: Pulsed-Neutron-Source Studies with the NASA Zero Power Reactor II. NASA TN D-5956, 1970.

5. Joanou, G. D.; and Dudek, J. S.: GAM-II. A B_3 Code for the Calculation of Fast-Neutron Spectra and Associated Multigroup Constants. Rep. GA-4265, General Dynamics Corp., Sept. 16, 1963.

6. Joanou, G. D.; Smith, C. V.; and Vieweg, H. A.: GATHER-II. An IBM-7090 FORTRAN-II Program for the Computation of Thermal-Neutron Spectra and Associated Multigroup Cross Sections. Rep. GA-4132, General Dynamics Corp., July 8, 1963.

7. Beck, C. K.; Callihan, A. Dixon; Morfitt, J. W.; and Murray, Raymond L.: Critical Mass Studies. Pt. III. Rep. K-343, K-25 Plant, Carbide and Carbon Chemicals Corp., Apr. 19, 1949.

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