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CONTROL OF A NUCLEAR REACTOR WITH A GASEOUS CONTROL SYSTEM

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Lewis Research Center
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SUMMARY

A control system concept utilizing a neutron-absorbing gas, helium 3, was investigated for use with a water-moderated reactor. A static gas system is finely distributed throughout the reactor core to minimize flux depression effects by the control medium. Reactivity control is performed by controlled pressurization of the helium gas.

Investigations have been performed in the areas of helium 3 containment and thermal and neutronic stability during operation. Although there is some probability that leaks could occur in the control system, the rate of loss of helium can be limited such that the reactor can be shutdown before it is damaged. Potential accident situations are limited by using flow restricting orifices in the system and by providing a positive water pressure seal on the control system.

The possibility of positive power coefficients of reactivity with a gaseous poison system was examined. Power coefficients of reactivity can be made small, and in some cases negative, by careful design. Configurations of a control element with small temperature and density gradients and a distribution system that provides a negative power coefficient are presented.

The concept of reactor control with helium 3 in metal control elements located in the moderator region appears feasible for the system investigated.

INTRODUCTION

Thermal nuclear reactors are generally controlled by inserting a neutron absorber directly into the reactor core. Historically, the neutron absorber is in the form of a

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*Some of the material herein was presented at the Winter Meeting of the American Nuclear Society, Pittsburgh, Pa., Oct. 30-Nov. 3, 1966, in a paper entitled "The Control of a Nuclear Reactor Using Helium-3 Gas Control Elements."

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cylindrical- or cruciform-shaped rod which is positioned within the reactor. Its position depends upon the amount of reactivity control required. Control rods are heavy, expensive to fabricate, require elaborate positioning mechanisms, and perturb the neutron flux distribution when they are moved. These undesirable characteristics of control rods have not prevented their use in present systems. However, with the advent of space power systems and reactor engines for rocket vehicle application, a need has arisen for the reappraisal of control systems.

A control system concept utilizing a neutron absorbing gas, helium 3, was investigated for use with a water-moderated reactor. Helium 3 is chemically inert and has a thermal neutron absorption cross section of about 5000 barns (5000×10^{-24} sq cm) (ref. 1). It can be obtained at a cost of about 100 dollars per liter at standard temperature and pressure (ref. 2).

The absorbing gas, or poison, could be either dissolved within the moderator or separated from the reactor materials by placing the gas in metal containers which are uniformly distributed throughout the reactor core. The homogeneous solution technique introduces problems in (1) moderator pH and corrosion, (2) uniform distribution of the solute, and (3) delay time in introducing the control gas. The contained gas system avoids these areas and will be the only one considered in this report.

A major advantage is gained if the absorbing gas can be distributed fairly evenly throughout the core region. Such distribution alleviates the problems associated with local flux distortions produced by control rods. Limiting consideration to static gas systems without continuous circulation gives a concept which would require a minimum number of moving parts.

On the other hand, unique properties of a gaseous control system present new features that could become problem areas. The mass of the required amount of gas for reactor control is an order of magnitude less than the equivalent mass of a solid control system. Consequently, the response of the control system is much faster than usual and extra care must be taken to avoid catastrophic loss of control gas. This problem can be solved by installing flow-restricting orifices in the control system.

Another possible problem that might arise is the inherent stability or instability of a reactor that utilizes a gaseous control element. The neutron capture process in the absorbing gas is exothermic. If the energy released by the capture event is deposited within the gas, gas density changes with temperature may result in a lower concentration of control material near the center of a core where the neutron fluxes are highest. It is possible that movement of the control material could produce positive coefficients of reactivity. Furthermore, axial temperature gradients in a poison gas system may lead to natural convection loops and a possible instability mechanism.

Previous studies of gaseous flux control systems have been performed with boron trifluoride (refs. 3 and 4). One study included an experimental test within a reactor operated at a low power level (6 W) (ref. 3). The other study employed the gas control as a means to establish a constant flux level within a single fuel element in a reactor for irradiation purposes.

The French Comissariat of Atomic Energy is presently constructing a 240 megawatt (thermal) power reactor in Brittany commonly referred to as "EL4" (ref. 5). This reactor is being built as a prototype for a heavy water-moderated gas-cooled concept and contains several different types of control system. One of these types is a system of helium 3 gas banks that are to be used as compensation elements for fuel element burnup, power coefficients, and xenon buildup.

A detailed analysis of such a system has been carried out (ref. 6) with the result that these banks are to be installed. However, the concept as developed in France has many different requirements from those considered in this report. Specifically, power density requirements are lower, long term operating effects must be accounted for, and the ambient conditions are considerably different. The last factor is, of course, the vast difference between the climate of Brittany and the vacuum of outer space.

In the remainder of the report, the proposed gaseous control concept for use in a water-moderated reactor is examined in terms of the possible disadvantages mentioned. These possible drawbacks, containment and stability, exist for any gaseous control element and can, therefore, be discussed in general.

More detailed study includes the definition of a specific control element design and the specification of a reference reactor for calculational purposes. The actual modes of system operation are discussed in detail. Control element performance involves local flux perturbations and reactivity effects. The mechanism of possible control element instability includes both thermal and neutronic effects which are coupled to each other.

As with any control system, analysis must be performed for optimal design configurations and the effect of off-design conditions. Finally, preliminary feasibility conclusions may be drawn concerning a gaseous control system in a water-moderated reactor.

DISCUSSION OF CONCEPT

Theoretically, the reactivity of a nuclear reactor can be controlled by varying the pressure of a gaseous neutron absorber such as helium 3 within the reactor core. It is postulated that a helium 3 control system would consist of (1) reactivity control elements uniformly distributed throughout the core, (2) a helium 3 distribution system which interconnects the control elements and transports the neutron absorber to and from the control elements, (3) a helium 3 supply and vent system located outside the reactor core, and (4) associated controls, valves, and instrumentation required to operate the control system.

The design of such a system must be based upon certain safety, reactivity, and environmental requirements. The reactivity and environmental requirements for a given control system design depend upon the particular reactor and are discussed in the section entitled "Performance requirements." The primary safety requirements are (1) containment, or the prevention of a rapid loss of helium 3 which could result in a reactor power excursion, and (2) thermal and neutronic stability.

Although the thermal and neutronic stability analysis is applicable to several types of reactor, the primary emphasis of the containment discussion is placed upon liquid-moderated nuclear reactors in which the gaseous poison is at least partly surrounded by the moderator.

Containment

The control system can be mechanically designed with the use of conservative safety factors; however, this alone would not reduce the severity of a nuclear excursion if a leak developed in the control system either inside or outside the reactor core. If it is assumed that a break could occur in the metal walls of the control system, the resulting reactivity increase can be reduced to tolerable limits by two methods:

(1) Placing flow-restricting orifices in the distribution line such that all the gas in the control elements must pass through an orifice before leaving the core

(2) Maintaining the moderator pressure higher than the gas pressure in the control system; this water seal will prevent significant loss of helium into the moderator region

The flow restricting orifices can be sized to limit the loss rate of gas from the core when a leak develops outside the core (outboard side of the orifice). The orifice must be surrounded by the water moderator to prevent rapid loss of gas should a leak occur on the inboard side of the orifice. If the gas pressure is greater than the water pressure, some of the poison would be forced into the moderator system and would possibly cause an increase in reactivity. Although this type of failure may not be so severe as the loss of gas when a leak develops outside the core, the rate of poison loss can also be reduced by maintaining the water pressure higher than the pressure of the gas in the control system. Since the water is much more dense than the gas, the flooding rate of the control system with higher water pressure will be much slower than the venting rate would be if the gas pressure was higher. The flooding rate can be further reduced by constricting the passages connecting the control elements to the distribution system.

Either type of situation described previously may render the control system inoperable; therefore, it may be desirable to provide an independent supplementary safety system which would allow the reactor to be shut down. Many types of supplementary safety system such as moderator dump, chemical poison injection, or poison rod insertion are currently in use and would be applicable with the gaseous control concept.

If a restart capability is desired without repairing the aforementioned damage, it is possible to provide multiple and separable gaseous control systems which would permit restart. Operation at reduced reactor power might then be required because of local power distortion.

Thermal-Neutronic Stability

In some reactor designs there may also be a problem with the inherent stability of the gaseous control system. Heat is generated within the gas as a result of the deceleration of the recoil protons and tritons produced in the ${}_2He^3({}_0n^1, {}_1p^1){}_1T^3$ reaction. Since helium 3 is a poor conductor of heat, large temperature gradients may exist in the control system. A possible instability mechanism may then exist in that a local increase of internal heat generation will produce a locally elevated temperature in the gas, with a commensurate increase in gas pressure. This local pressure perturbation will give rise to outward moving pressure waves as the gas seeks a new pressure equilibrium. The pressure fronts propagate with the local speed of sound,[1] influencing gas pressures, temperatures, and densities throughout the control system in a complex manner. The pressure wave may force gas out of the initially perturbed region, lowering the gas density there and raising the local neutron flux. Two competing influences are set into action. The lowered density will tend to reduce the internal heat generation q and the increased neutron flux will tend to increase q. If the neutron flux is the predominant influence, a further increase in internal heat generation results and provides the positive feedback for an instability.

The stability problem is discussed quantitatively later in the section entitled "Steady-state stability," in connection with a specific reference reactor and control system design concept. No general statement can be made about the stability of gaseous control systems without further calculations using other reference reactor designs.

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[1]The wave fronts produced propagate through the tube at sonic speed (about 900 m/sec). The characteristic perturbation time is, therefore, of the order of milliseconds, which is long compared with neutron generation times.

Control-Element Design Considerations

The control-element design represents a compromise between heat transfer and neutronic considerations. On one hand, the volume must be large enough to hold sufficient helium 3 to satisfy reactivity requirements, and the gas pressure should be maintained lower than the water pressure to avoid accidental loss of helium. Therefore, the gas must be adequately cooled to keep the temperature and the pressure low. On the other hand, since the thermal conductivity of helium is low, it becomes desirable to keep the gas passages small with high pressures to reduce the average temperature within the gas.

An annular control element in which the gas is held in the space between two concentric tubes will provide a large surface area for heat removal, and the annulus width can be made small to minimize the average temperature within the gas. Other design considerations which would reduce the gas temperature are

(1) Tube material thermal conductivity: A material with high thermal conductivity should be selected to minimize the temperature rise across the tube wall. This material should have a low neutron absorption cross section.

(2) "Proton traps": The heat generated in the helium 3 can be reduced by adding internal fins with high thermal conductivity and stopping power into the gas space. The energy of the recoil projectiles, from the ${}_2He^3\left({}_0n^1, {}_1p^1\right){}_1T^3$ reaction, would be dissipated in the high-conductivity "trap" rather than in the low-conductivity gas.

ANALYSIS OF REFERENCE CONTROL SYSTEM

In this section a reference control system is analyzed. A heterogeneous water-moderated reactor is selected as a reference reactor design, and the performance requirements for the control system are specified. The control system component configuration or size is established to satisfy as many of these requirements as possible, and operating methods are discussed. Determining whether the other requirements are satisfied requires an investigation of the reference control system performance in the reference reactor. The performance study includes a determination of the neutronic, stability, and hydraulics and heat-transfer characteristics of the system.

Reference Design for Analysis

Description of reactor. - The concept of a gaseous control system is applied to the control of a heterogeneous water-moderated rocket propulsion reactor. A schematic

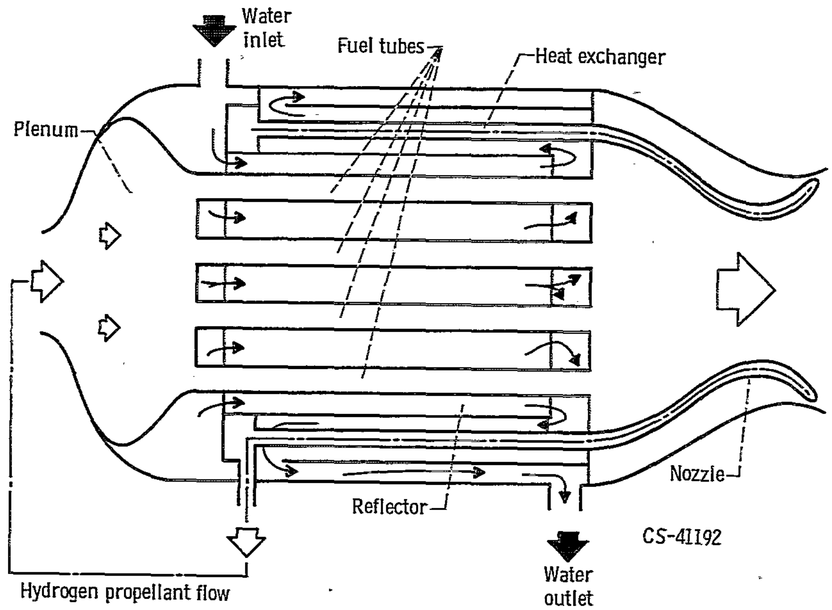


Figure 1. - Water-moderated rocket reactor concept.

diagram of such a reactor is illustrated in figure 1 with emphasis on two fluid circuits in the system. The open hydrogen circuit is indicated by dashed lines and open arrows. The liquid hydrogen enters the bottom of the nozzle and then cools the water as it passes through a heat exchanger. After passing through turbomachinery (not shown), it reaches a plenum above the reactor core, passes through the fuel tubes in the core, and finally passes out through the nozzle to provide propulsive thrust. The circuit shown solid with solid arrows is a closed water circuit providing sufficient circulation through the core, reflector, and heat exchanger to remove the heat generated in the water, which stems primarily from neutron moderation and gamma heating. For the gaseous control system, the water serves as a large, almost-constant-temperature heat sink.

Performance requirements. - The reactivity and environmental requirements for the gaseous control system (table I) depend upon the design parameters of the parent reactor.

In the reference reactor, the control system must accommodate sufficient helium 3 to control both 0.5 dollar reactivity during xenon override and 16.1 dollar reactivity while the reactor is shut down and the fuel and moderator are cold. The reactor has a negative temperature coefficient of reactivity for both the fuel and the moderator such that less helium 3 is required when the fuel and moderator have been heated. During normal operation (hot critical) the control system controls 10.7 dollar reactivity. The maximum rate of removal of helium from the control system is limited to 6 cents per second. The minimum helium 3 addition rate required for a reactor scram is 10 dollars per second. Fine control reactivity increments as small as ±1/2 cent are required to

TABLE I. - CONTROL-SYSTEM DESIGN REQUIREMENTS

Reactivity	
Worth of helium 3 held in reactor, $\Delta k/k$, $	
At cold shutdown	16.1
At hot clean critical	10.7
At xenon override	0.5
Addition and removal	
Maximum removal rate, ¢/sec	6
Minimum scram rate, $/sec	10
Increments of fine control, ¢	±1/2
Maximum allowable variation of poison mass between elements, percent	±5
Environmental	
Pressure outside control system, psia (N/sq cm)	
In core (moderator region)	
Normal operation	600 (414)
Shutdown	100 (68.9)
Outside reactor core	0
Water coolant	
Inlet temperature, ^{o}R (^{o}K)	656 (365)
Flow per lattice cell, gal/min (cu dm/sec)	30 (1.9)
Aluminum temperature limit, ^{o}R (^{o}K)	760 (422)
Average heating rates at 100 percent power, W/cu cm	
In water	150
In aluminum	124
Reactor operating time, hr	1
Number of reactor startups	Five

compensate for gradual fuel and poison burnup and fission product poisoning. The maximum radial element-to-element variation of helium density allowed is ±5 percent.

The in-core portion of the control system is surrounded by a water moderator at a pressure of 600 pounds per square inch absolute (414 N/sq cm) during normal operation and 100 pounds per square inch absolute (68.9 N/sq cm) during shutdown. The out-of-core portion of the system is exposed to space environment (0 psia).

The in-core portion of the control system is cooled by the water moderator flowing adjacent to the walls of the control element annulus. During normal operation the average water flow to a cell[2] is 30 gallons per minute (1.9 cu dm/sec) and the water inlet temperature is 656^{o} R (365^{o} K).

[2]A cell contains an equivalent of one nuclear fuel assembly and two control elements. The 30 gal/min (1.9 cu dm/sec) must be allocated for cooling these components.

Control-System Configuration

The reference configuration consists of a series of annuli placed in the interstitial regions between fuel assemblies (fig. 2). Several fuel lattice geometries might be possible in the reactor, but for the purposes of this study a triangular array of fuel tubes was chosen.

The distribution system is indicated in figures 2 and 3. This system is composed of channels of rectangular cross section to fit into the desired location in the core. These channels connect the annuli in the reactor to a main distribution header. This

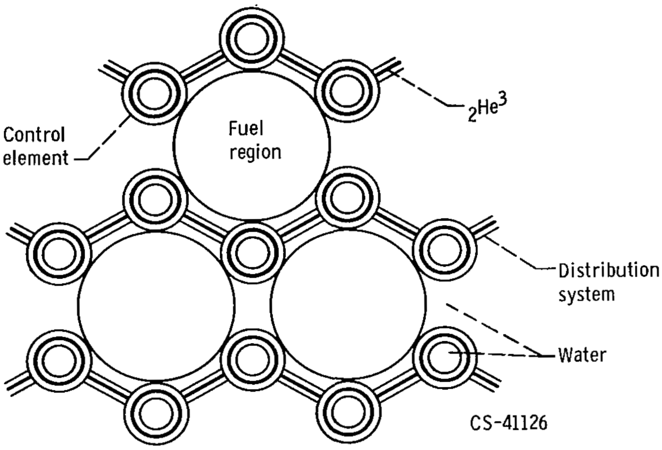


Figure 2. - Annular control element.

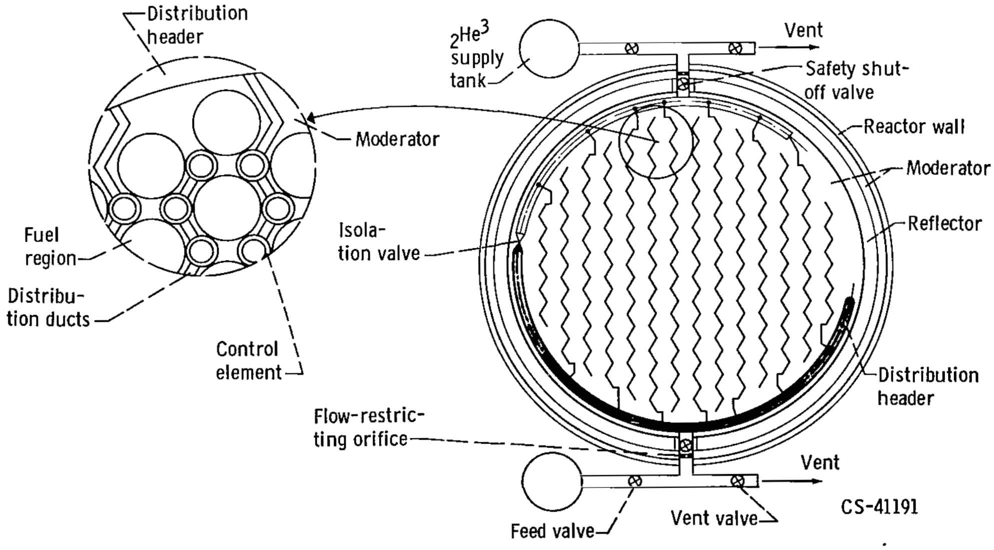


Figure 3. - Reference control system.

distribution system would be located in a region of low nuclear importance at the outlet end of the core. This reference design does not necessarily reflect the best or final design for a gaseous control system. For instance, an alternative system is discussed in appendix A. However, the stability and containment characteristics of an optimized system would be very similar to those of the system discussed herein. The total control system consists of two identical but independent gas circuits (fig. 3). The two systems can be either connected or separated by opening or closing the isolation valve in the distribution header.

The primary components of each of the reference control circuits are fabricated from aluminum and consist of (1) annular-type control elements, (2) distribution lines with rectangular-cross-section gas passages, (3) flow-restricting orifices, and (4) a helium 3 supply tank. Control instrumentation is not discussed in this analysis. The only moving parts in the system are the control valves. Pumps are not required in the system, because the heat generated in the gas can be adequately removed by conduction to the moderator. Mechanical mixing of the gas is not required, because helium 3 dilution as a result of burnup $\left({}_2\mathrm{He}^3\left({}_0\mathrm{n}^1, {}_1\mathrm{p}^1\right){}_1\mathrm{T}^3\right)$ is insignificant for the short operating life of the reference reactor.

Reactivity is controlled by adding helium 3 from the supply tanks or by venting the gas from the core. No attempt is made to reprocess or collect the gas vented from the system because only about 35 grams of helium 3 are required to satisfy the five startups required in table I.

If it were desirable to save helium 3, the vented gas could be collected in a low-pressure tank for later repressurizing. A purification system for the helium 3 could also be included in the system design. LaGrange, Dolle, Satre, and Dirian (ref. 6) have investigated several helium 3 purification systems for use with the EL4 reactor.

The reference system component dimensions are summarized in table II. The corresponding volumes and masses of helium 3 and aluminum in the control system are summarized in table III. Masses and volumes for controls and instrumentation are not tabulated. The total mass of the control system excluding valves, actuators, and instrumentation is about 141 kilograms. The total mass of helium 3 in the control system is about 0.5 kilogram. Since the cumulative amount of helium 3 required to maintain reactor power at 100 percent, shut down, and restart the reactor is unknown, a conservative estimate of the total gas required was assumed.

The control-system components which are surrounded by the moderator were sized to operate with at least 10 psi (7 N/sq cm) lower pressure than the moderator to prevent escape of the gas into the moderator. Therefore, the maximum allowable pressure in the system during reactor operation is 590 psia (407 N/sq cm) compared with 90 psia (62 N/sq cm) when the reactor is shut down.

The system pressures required to maintain the reactor at 100 percent power and to

TABLE II. - DIMENSIONS OF GASEOUS CONTROL SYSTEM

Annular control element		
Annulus outside diameter, in. (cm)	0.764	(1.94)
Annulus inside diameter, in. (cm)	0.690	(1.75)
Wall thickness, in. (cm)	0.060	(0.15)
Element length, in. (cm)	39.5	(100)
Distribution lines (rectangular cross section)		
Helium passage width, in. (cm)	0.20	(0.51)
Helium passage height, in. (cm)	1.27	(3.23)
Line wall thickness, in. (cm)	0.060	(0.15)
Total length of lines, in. (cm)	370	(940)
Distribution header (rectangular cross section)		
Helium passage width, in. (cm)	0.18	(0.46)
Helium passage height, in. (cm)	4.13	(10.5)
Header wall thickness, in. (cm)	0.060	(0.15)
Total length of header, in. (cm)	200	(508)
Poison supply tanks (spherical)		
Tank inside diameter, in. (cm)	20.6	(52.3)
Tank outside diameter, in. (cm)	21.0	(53.3)
Number of tanks		Two
Poison supply lines (from supply tanks to distribution header)		
Line inside diameter, in. (cm)	0.25	(0.64)
Line outside diameter, in. (cm)	0.31	(0.79)
Line length, in. (cm)	200	(508)

TABLE III. - MASSES OF COMPONENTS OF CONTROL SYSTEM

[Total weight of nuclear rocket engine system, about 6.5×10^6 g.]

Component	Volume, cu cm		Mass, g	
	Helium 3	Aluminum	Helium 3	Aluminum
Control elements	11 900	38 700	4.45	104 500
Distribution ducts	1 540	1 160	.44	3 130
Two distribution headers	2 440	1 750	.70	4 720
Two 1/4-in. supply lines	322	350	1.20	944
Two spherical supply tanks	150 900	10 300	563.00	27 800
Two flow-restricting orifices				[a]4
Total	167 100	52 260	569.8	141 100

[a]Each flow-restricting orifice is fabricated from stainless steel and weighs about 2 g.

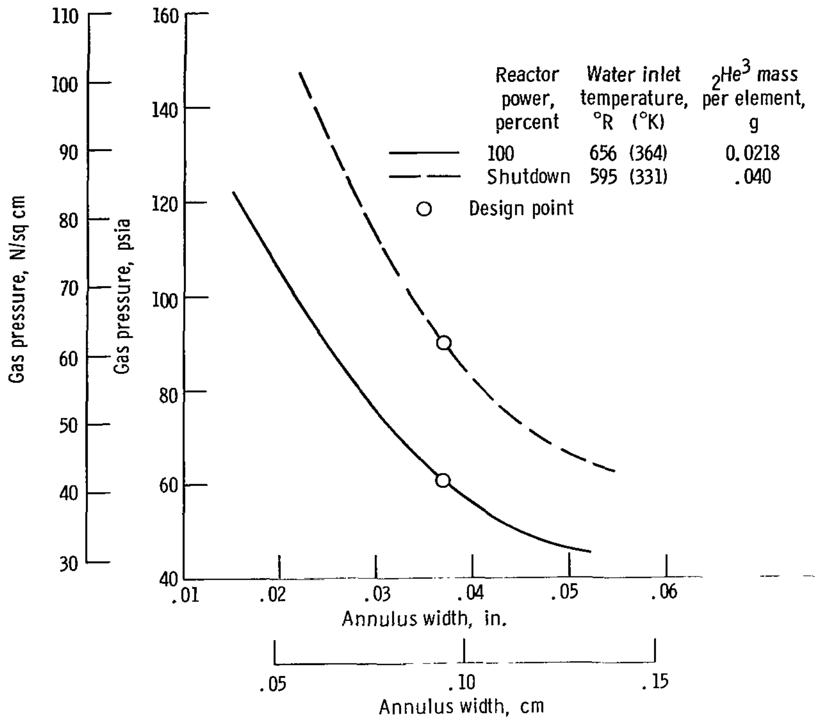


Figure 4. - Gas pressure in control system. Annulus outside diameter, 0.764 inch (1.94 cm).

keep the reactor shut down are shown in figure 4 for various control annulus widths. For criticality the mass of helium 3 required in each control element is 21.8 milligrams, while the mass of helium 3 required to keep the reactor shut down is 40 milligrams. The size of the control-element annulus was based on shutdown conditions rather than on 100 percent power because

(1) The maximum allowable pressure is least during shutdown (90 psia; 62 N/sq cm).

(2) The mass of helium 3 required is greatest during shutdown. A 0.037-inch (0.094-cm) annulus width was selected such that the reactor could be maintained shutdown with a gas pressure of 90 psia (62 N/sq cm). During 100-percent power the gas pressure required with this control element is only 61 psia (42 N/sq cm), only about 10 percent of the maximum allowable pressure during operation at 100-percent power.

If the water and gas temperatures should rise above 595° R (331° K) during shutdown conditions, it would be necessary to raise the pressure above 90 psia (62 N/sq cm) to maintain the same mass of helium 3 in the control system. However, because of the negative temperature coefficient of reactivity, an increase in water temperature would be accompanied by a decrease in core reactivity, and less helium 3 would be required to maintain the reactor subcritical.

The distribution system consists of (1) hollow (rectangular cross section) distribution ducts which feed the control elements and (2) a distribution header which connects the ducts with the supply system as shown in figure 3 (p. 9). The distribution system

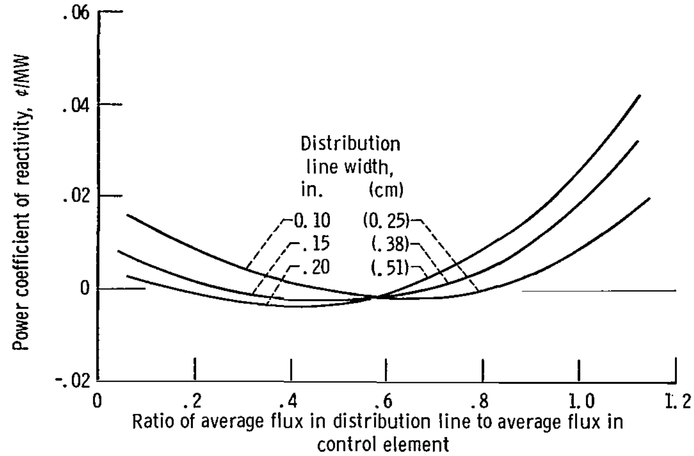


Figure 5. - Power coefficient of reactivity for single cell. Reactor power, 100 percent; constant mass of helium 3 in cell.

was placed in a region of low nuclear importance near the nozzle end of the core to minimize the perturbation in the axial power distribution. The distribution-system flow area was sized such that the flow disparity among control elements attached to a given distribution duct is less than 5 percent.

If the initial average gas temperature in the control element is higher than the average gas temperature in the distribution system, an increase in reactor power causes a greater percentage increase in control gas temperature than in distribution-system gas temperature. Therefore, gas is forced out of the control element and into the distribution system to maintain pressure equilibrium. When poison is transferred from a region of high importance (the control element) into a region of lower importance (the distribution system), reactivity is increased. The overall effect is a positive power coefficient of reactivity. It is desirable, on the basis of reactor safety, to have a negative power coefficient of reactivity associated with the control system. The distribution duct width (0.20 in. or 0.51 cm) was enlarged sufficiently to yield a small negative power coefficient of reactivity (about 0.005 ¢/MW) for the control system, as shown in figure 5. The value of the flux in the distribution system relative to the control element φ_d/φ_c is 0.35 (fig. 15, p. 22).

Each of the curves in figure 5 has a minimum power coefficient. The region to the right of the minimum where power coefficients become positive again represents relative neutron fluxes in the distribution system which cause the distribution system to have a greater nuclear importance than the control element. Because the distribution system in the reference design is located at the outlet end of the reactor, the relative flux is sufficiently low to avoid positive power coefficients due to this effect. The calculation of power coefficients is shown in appendix B.

The flow-restricting orifice in each control circuit shown in figure 3 (p. 9) is required to limit the helium 3 venting rate to 6 cents per second, as required in

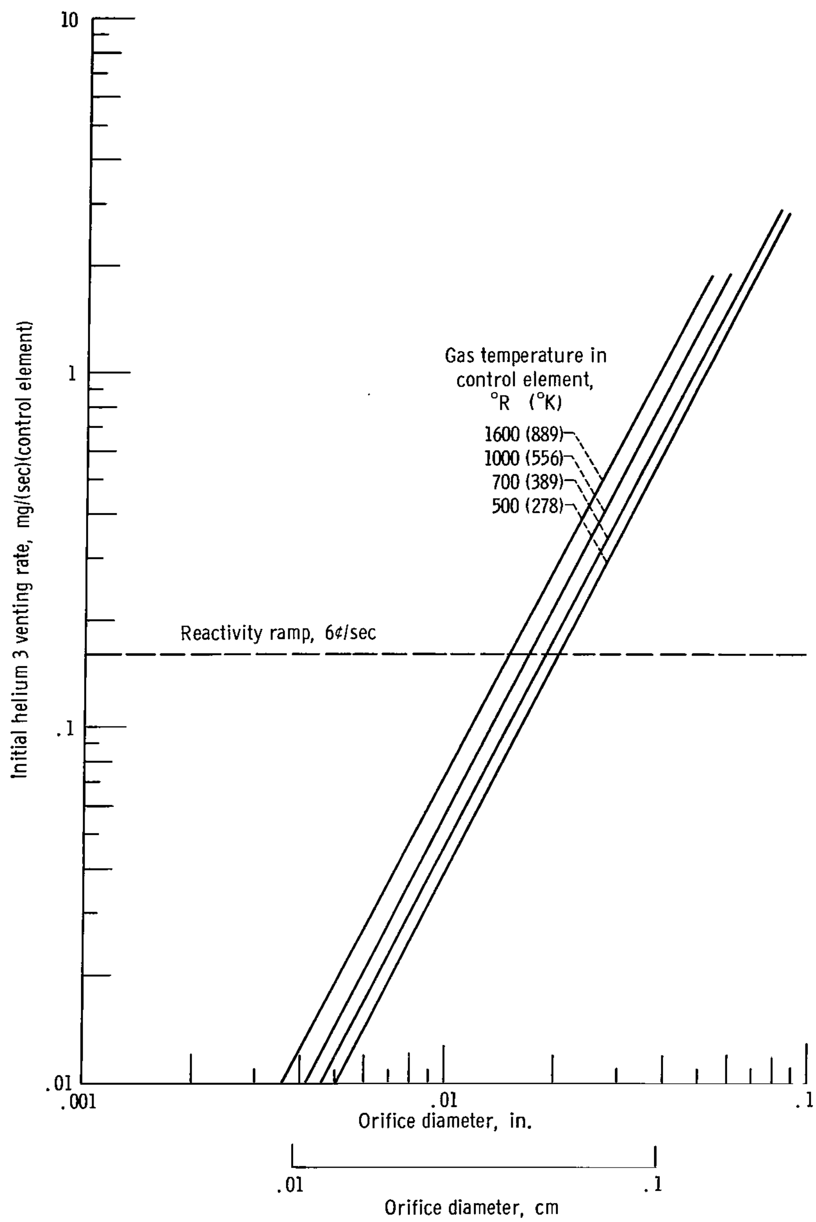


Figure 6. - Control-element venting rate. Initial poison content, 21.8 milligrams per element; gas vented to space; control-system volume, 0.56 cubic foot (0.016 cu m).

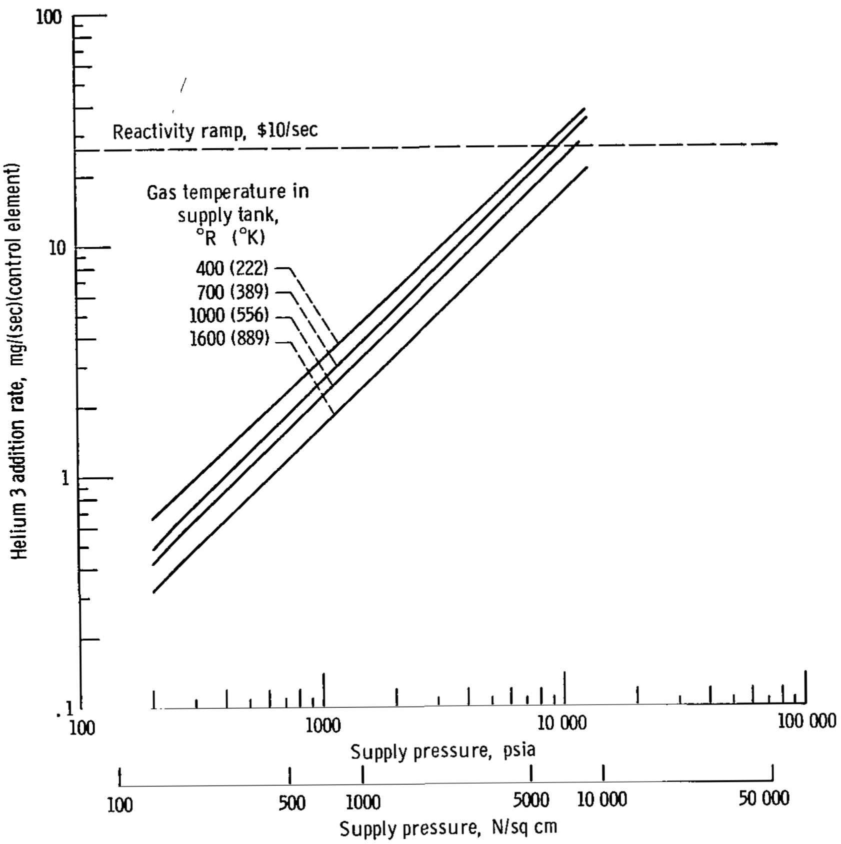


Figure 7. - Helium 3 addition rates. Orifice coefficient, 0.90; orifice diameter, 0.018 inch (0.046 cm).

table I (p. 8). With this maximum venting rate, in the case of outside leaks, the reactor could be shut down with the use of either the alternative control circuit or the supplementary safety system before the reactor is damaged. The venting rates through various size orifices are shown in figure 6 for various helium 3 temperatures. Under normal operating conditions, a 0.018-inch (0.046-cm) orifice would limit the venting rate to 6 cents per second. The helium 3 supply rate for this orifice is shown in figure 7 as a function of helium 3 supply pressure and temperature. A supply rate of 26.7 milligrams per second of helium 3 is required for fast shutdown. From figure 7, it is obvious that a supply pressure of about 12 000 psia (8274 N/sq cm) would be required to supply 26.7 milligrams per second through a 0.018-inch (0.046-cm) orifice. Therefore, a scram system is incorporated with a separate supply tank within the pressure vessels. This system will be discussed in the section entitled "Methods of Operation."

The fine control (±1/2¢) required in table I would be provided by restricting the length of time that the supply or vent valve is held open. For example, a reactivity addition of 1/2 cent would be provided by holding the vent valve open only 0.1 second at a venting rate of 5 cents per second. If it is impractical to hold a valve open only 0.1 second, the venting or supply time might be increased to a more practical value by reducing the size of the flow-restricting orifice. If the orifice size is reduced, however, the gas venting rate is reduced and the time required for starting the reactor would be longer.

The poison supply system consists of duplicate spherical aluminum tanks in which the helium 3 is stored and aluminum supply lines which transport the helium 3 to the control system. The supply system shown in figure 3 (p. 9) is located outside the reactor core to minimize the neutron-gamma heating in the system. The supply system was designed to operate at 500 psia (345 N/sq cm).

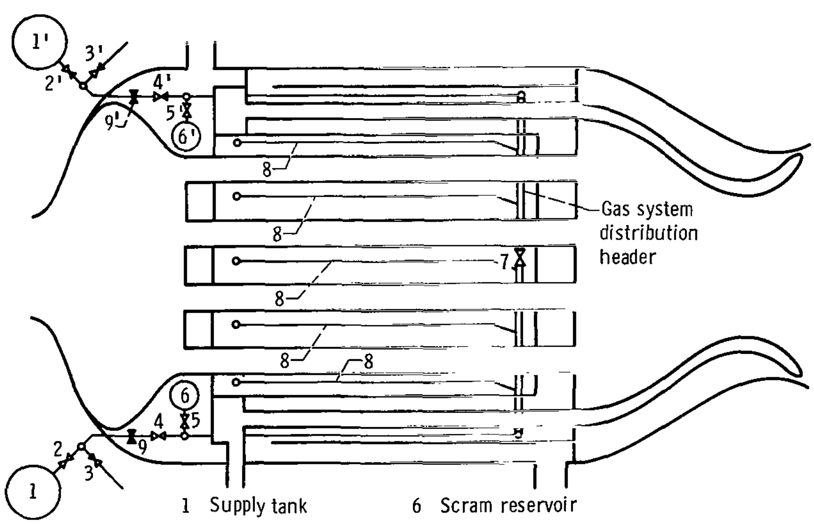
Methods of Operation

The underlying purpose of the investigation was to conceive a system as reliable and safe as a conventional control-rod system.

Overall circuit design is governed by three design decisions:

(1) There are at least two completely self-sufficient circuits, each with its own supply and controls. However, the possibility of connecting the systems either to equalize the system pressures or to compensate for possible system malfunctions is incorporated in the design.

(2) The helium supply tanks are placed outside the reactor pressure vessel to prevent excessive heating. Two spherical tanks 20.6 inches (52.3 cm) in diameter



1 Supply tank
2 Gas feed valve
3 Gas vent valve
4 Safety shutoff valve
5 Scram valve
6 Scram reservoir
7 Isolation valve
8 Control element
9 Flow-restricting orifice

Primes denote components of second system.

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Figure 8. - Reactor with gaseous control system.

jointly hold 563 grams of helium gas at 500 psia (345 N/sq cm), which is sufficient for about 70 reactor startups.

(3) The supply header and manifold are located near the outlet end of the reactor.

The location and the size of the header received considerable attention because the header can be designed to produce a negative temperature coefficient. If the header passage is made thicker than the effective control gas region, the gas temperature in the header can be made higher than that in the core region. There is thus a tendency to displace gas into the core during reactor power increases. The complete control circuit in the reactor is shown in figure 8. It consists of two identical systems which can be connected or separated by means of valve 7. External to the reactor vessel are only the supply tanks 1 and 1', the gas feed valves 2 and 2', and the gas vent valves 3 and 3'.

Normally, valves 2, 2', 3, and 3' are the only moving parts in the system. The remaining circuit components are in the reactor, submerged in water and subject to the

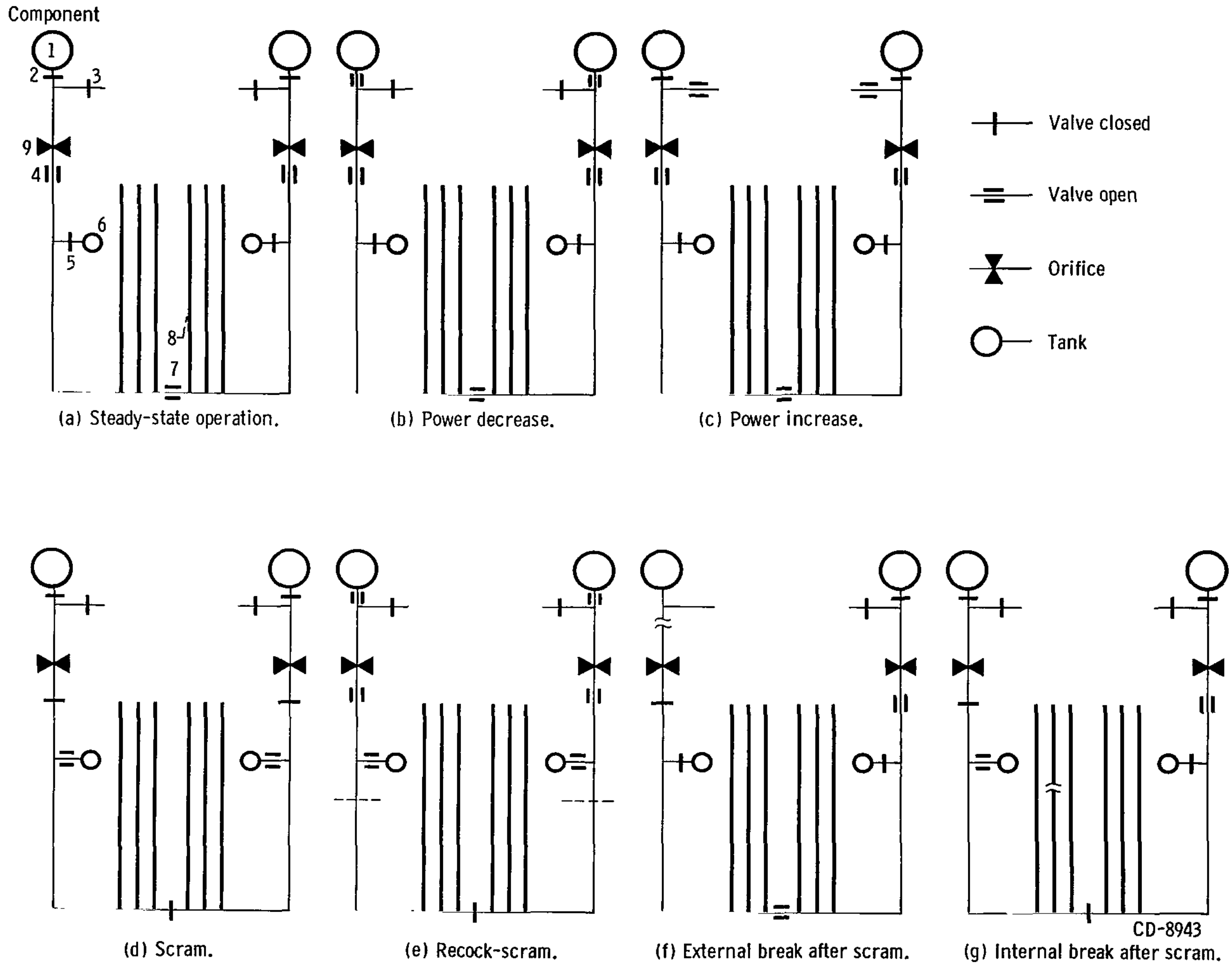


Figure 9. - Methods of operation (single distribution system).

water overpressure in case of internal leaks. The components are in series: a flow-restricting orifice 9, a safety shutoff valve 4, a scram valve 5, a scram reservoir 6, and control elements 8.

This basic circuit is repeated in schematic form in figure 9. The control circuit shown in figure 9 represents a distribution header arrangement identical to the reference system shown in figure 8. If necessary, a dual header arrangement which offers the advantage of allowing the system to be purged could be designed. In reactor systems experiencing low burnup of poison, the dilution of the poison is small and a purge system is not required.

The valve modes for various operating conditions are briefly described with reference to figure 9. During normal operation (figs. 9(a) to (c)) all valves are closed except valves 4 and 7. Valve 4 closes only during emergencies, and valve 7 is open to equalize the system pressures.

To decrease reactivity, as shown in figure 9(b), it is simply necessary to admit gas to the reactor by opening valve 2, which is an on-off type. Admission rate is controlled by the orifice designed for choked flow. The amount of control gas inserted depends solely on the length of time valve 2 is held open. (This operation is similar to the operation of a control-rod drive motor.) For automatic operation, valve 2 presumably would respond to a neutron flux signal. Valves 2 and 3 should be so interlocked that only one valve is open at any one time. Gas is vented to space or to a collection tank through valve 3 to add reactivity.

Emergency conditions are considered in figures 9(d) to (g). A scram is achieved by isolating the systems from the outside by closing valve 4 and from each other by closing valve 7. Subsequently, valve 5 opens and discharges the reservoir supply of gas from tank 6 into the system. Supply tank 6 should be sized to admit sufficient gas to raise the system pressure to 100 psia (68.9 N/sq cm). In addition, it will probably be necessary to shield these tanks to prevent excessive internal neutron heating.

Tank 6 can be recharged as shown in figure 9(e). The system must either be designed to take a momentary overpressure (also desirable for leak testing), or an additional valve (shown dashed) must be installed in the vertical header.

Two emergency conditions are considered. Figure 9(f) illustrates that in case of an external break, valve 4 closes in response to a low pressure signal and thus isolates the system from the outside. In case of an internal break, water would enter the control system. A leak detection device would sense the presence of the moderator, cause valve 7 to be closed, and thus separate the dual control circuits. Simultaneously a reactor scram would be caused by adding gas from the scram reservoir to the undamaged control circuit. With one of the two circuits not controllable, the neutron flux distribution will be distorted to the point that reactor power will have to be reduced. Control will be effected by the remaining circuit.

Control-System Performance

The preceding section indicates the physical configuration used to investigate the gaseous control-system concept. The operating performance that might be expected from this system is now considered.

Neutronic analysis. - The investigation of the gaseous control concept included a number of nuclear calculations. Local flux depression effects of both the control element and a distribution system were studied and found to be much smaller than would occur with control rods. The reactivity effects of a distribution system were checked, and a total control worth curve for a gas control system was calculated. A method was also developed to study possible spatial instabilities that might arise in a gaseous control element. The details of neutron cross section preparation and nuclear calculations in this work are described in appendix C.

In order to evaluate the extent of the local neutron flux depression in the region of the helium 3 annulus, a "reverse-cell" calculation was set up, as indicated in figure 10. The cell has as its center the centerline of the helium 3 annulus. The mass of all materials enclosed by the dashed triangle is preserved in the new cylindrical cell. For instance, the outer annulus of the new cell contains the same amount of fuel as one-half of the actual fuel assembly of the reactor.

The curves in figure 11 represent the radial thermal flux profiles with and without poison in the annulus. The thermal flux (below 0.414 eV (6.62×10^{-18} J)) relative to the

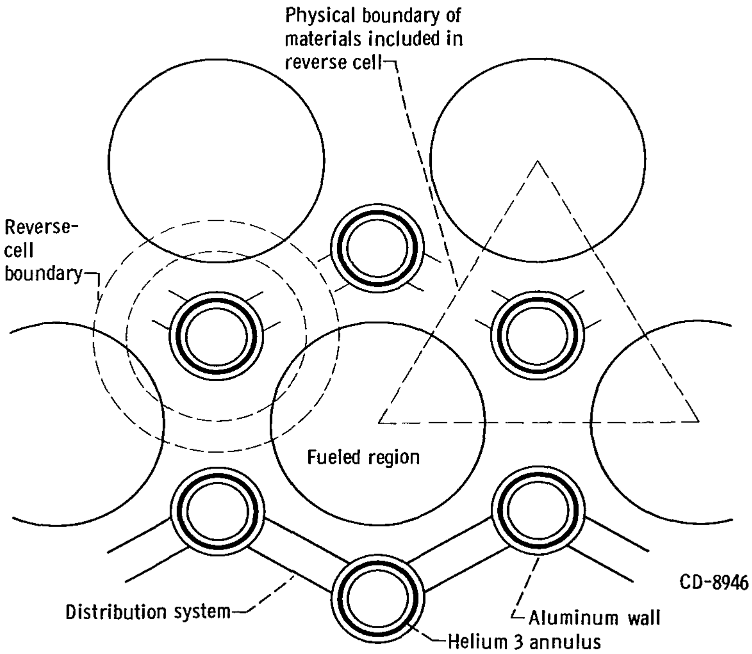


Figure 10. - Geometry of "reverse cell".

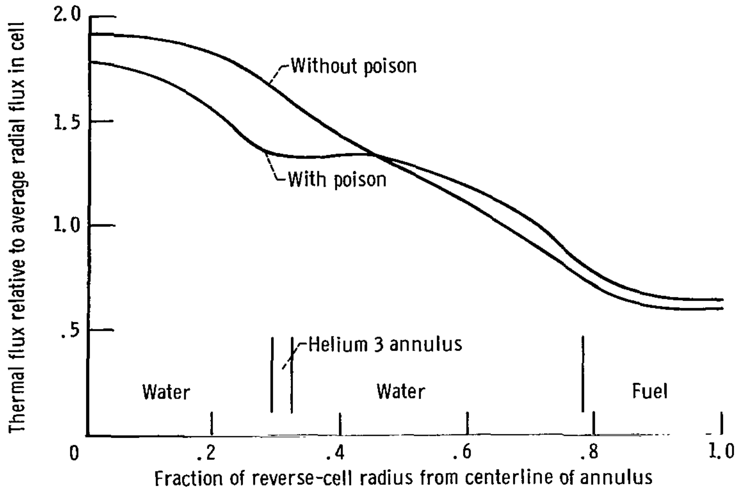


Figure 11. - Thermal flux profiles in reverse-cell calculations.

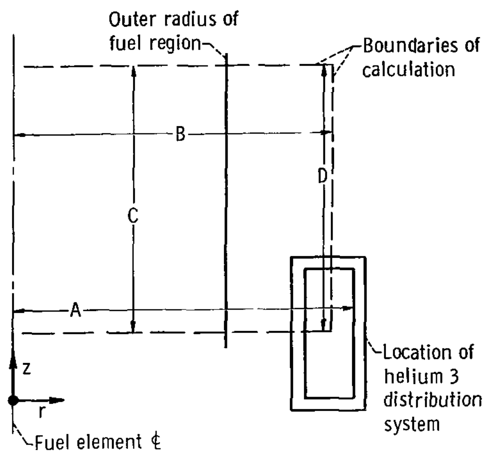


Figure 12. - Two-dimensional r-z calculational geometry of distribution system.

cell average thermal flux is plotted in this figure. Local flux depression within the control element is approximately 15 percent when the annulus contains poison. The value of the perturbation in the fuel will be much less, as indicated by the flux profiles of the outer region of the reverse cell. This "with-poison" case represents a typical gas concentration that might be required for hot-critical operation.

An investigation of the local flux perturbations due to the distribution system at the outlet end of the core was made with the helium 3 gas at a typical operating density (0.371×10^{-3} g/cu cm). The calculation consisted of a two-dimensional S_4 calculation in the r, z geometry of figure 12. The boundaries of the calculation indicated cut a one-quarter section of the rectangular distribution system cross section. The lines A, B, C, and D represent the locations of the flux traverses shown in figures 13 and 14.

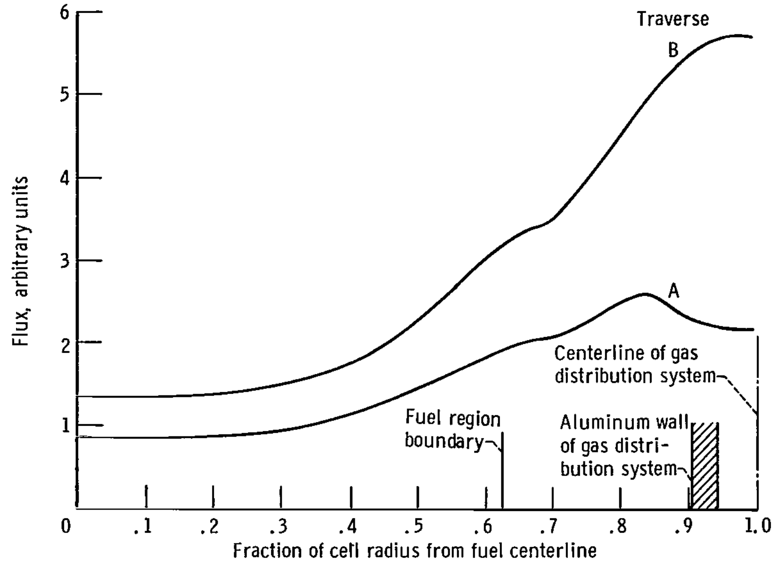


Figure 13. - Radial thermal flux profile near distribution system.

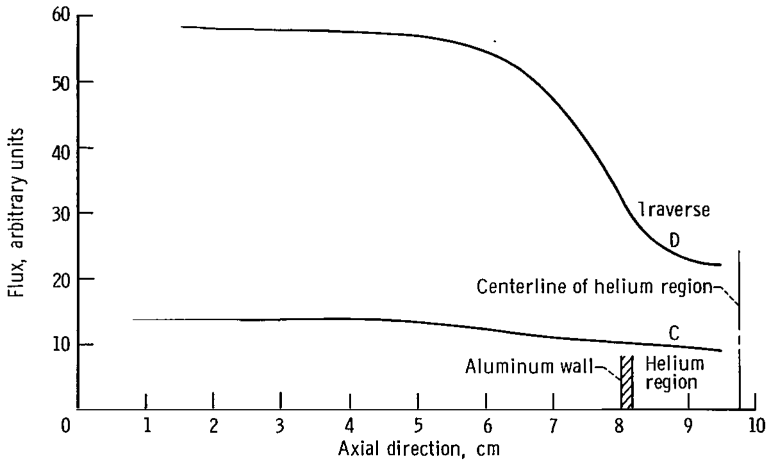


Figure 14. - Axial thermal flux profile near distribution system.

In figure 13, the shape of the flux within the fuel region remains relatively unperturbed despite a large flux depression at the distribution system. Figure 14 indicates that perturbation effects in the axial direction diminish within 3 centimeters of the distribution plenum.

As described in appendix C, cross sections were developed from this r, z calculation to be applied to a one-dimensional axial calculation. This axial calculation establishes the reactivity effect of the distribution system. While variable gas density within the control-element annulus was accounted for in this analysis, no gaseous redistribution between annuli and feed system was included.

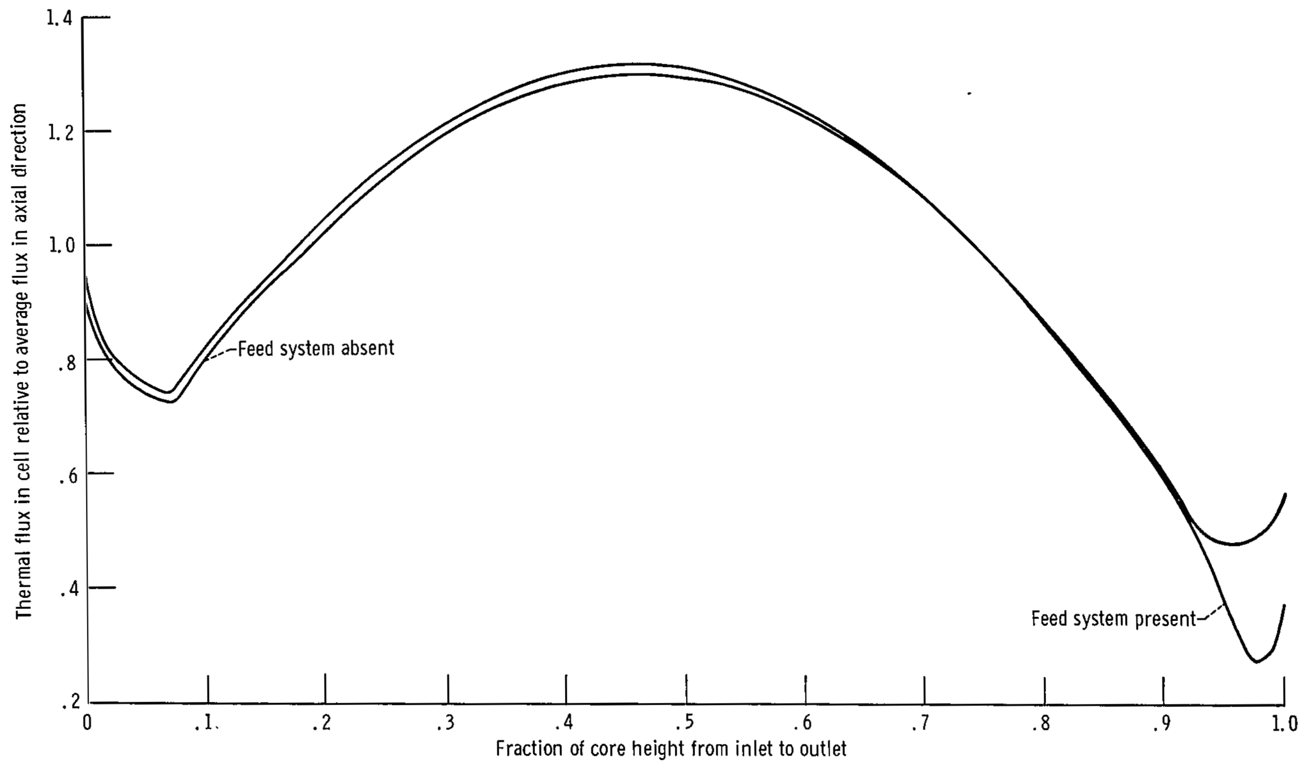


Figure 15. - Effect of distribution system on axial profile of thermal neutron flux.

The reactivity effect of such a helium distribution system amounts to 12 cents negative when this system replaced the lower $3\frac{1}{2}$ centimeters of the control annuli in the reference calculation. Figure 15 represents the axial distribution of the radial average thermal flux in the cell. The perturbation due to the feed system results from the flux depression within the moderator region seen in traverses A of figure 13 and D of figure 14. The gross axial power distribution will only be slightly affected by the distribution system as indicated by traverse C of figure 14.

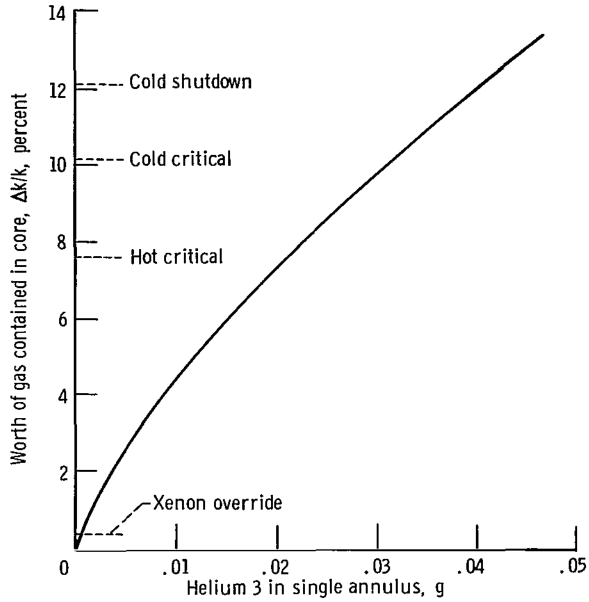


Figure 16. - Typical reactivity curve for helium 3 system.

Figure 16 illustrates helium worth values as a function of gas content for this annlus design. The data points used to generate this curve were obtained from axial core calculations with a constant gas density in the annuli. The deviation from this curve due to the additional worth of a gas distribution system and the variable axial gas density is not sufficiently large to show on this scale.

Steady-state stability. - Control stability represents a possible problem area in the use of a gaseous control system. That portion of the system that is located in the center of the reactor will have a tendency to run hottest, and the resulting gaseous expansion will remove poison from the most important flux region.

Although the mechanical design of the system has been focused upon restricting the temperature rise (the average gas temperature) through the control gas, it is still necessary to investigate the causes of possible instabilities that might arise. Therefore, the following technique was devised:

An initial transport theory calculation in the axial direction was performed with the assumption that the poison gas density was axially uniform in the core. The thermal

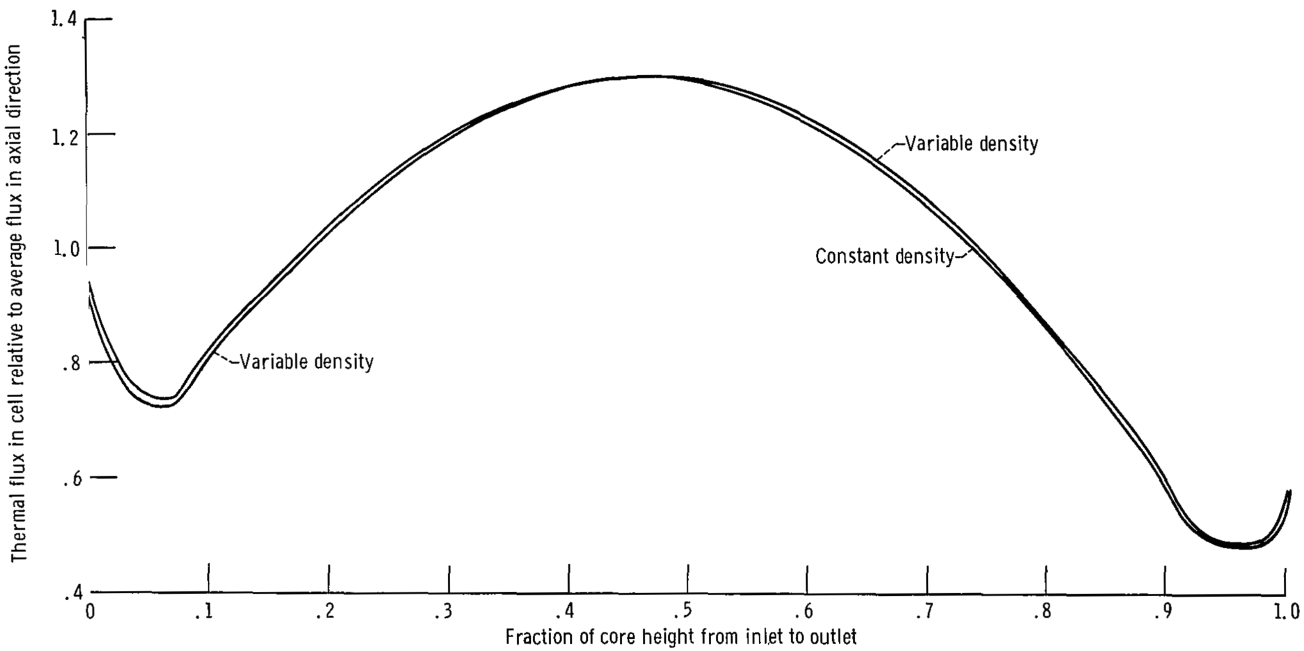


Figure 17. - Effect of variable gas density on axial profile of thermal neutron flux.

group flux profile for this axial cell calculation was then used to represent the internal heat generation profile in the helium 3 system.

A heat-transfer analysis was carried out to produce an axial density distribution in the gas. The axial distributions of temperature and gas density for constant total mass of helium in a control element were determined by calculating the radial temperature distribution at successive axial positions with the steady-state heat-transfer model described in appendix D. In this model the heat transfer in the axial direction is neglected. The error introduced by neglecting axial heat transfer is less than 1/2 percent, primarily because the gas annulus and the aluminum wall of the reference control element are so thin relative to their length.

Cross-section values for a series of zones in the axial direction were then obtained from the cross-section curves that had been generated as a function of poison concentration. The axial transport calculation was then repeated. This iteration technique was continued until a successive iteration made no significant change in the calculated axial gas density profile. (No significant change means that the calculated difference in density at any given location did not change the macroscopic absorption cross section by 5 parts in 10 000.) The condition of no density change in successive iterations was assumed to represent convergence of the solution.

This iterative technique converged rapidly. The difference in eigenvalue for an axial calculation with a constant poison density as opposed to a varying gas density for typical operating conditions represents about 1 cent in reactivity worth. The effect of variable gas density upon a typical axial flux profile is represented in figure 17. The maximum change in local flux values is 1.78 percent at the inlet end of the core.

The assertion that the axial thermal flux profile in the cell represents the profile of the internal heat generation within the poison annuli involves assumptions in both energy and space.

The nuclear analysis of this concept was performed with the use of six energy groups as indicated in appendix C. The thermal, or sixth neutron energy group, flux in the transport calculations accounts for approximately 86.6 percent of all absorptions in the helium 3 gas. The fifth and fourth group fluxes account for 7.5 and 5.2 percent, respectively. Furthermore, the axial profiles of the fourth and fifth group fluxes are similar in shape to that of the sixth group. The axial distribution of the helium 3 absorptions is thus well represented by the thermal flux profile.

Employment of the sixth group axial profile to represent the heat generation distribution within the poison system requires a knowledge of local radial flux distribution as a function of local poison concentrations. Specifically, the ratio between the radially averaged flux and the thermal flux within the helium has been assumed to be constant over the range of poison densities existing in the core. Once again, the so-called reverse-cell geometry of figure 10 (p. 19) was used to check this effect. The range of

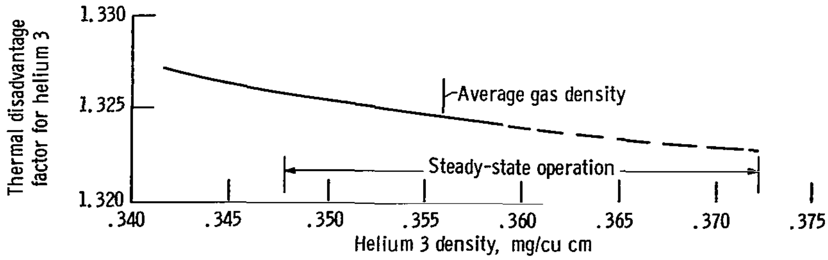


Figure 18. - Thermal disadvantage factor for helium 3 as function of helium 3 density.

gas densities is indicated in figure 18. The vertical line represents the average gas density in the poison annulus, while the arrow shows the range of gas densities in the annulus at steady-state operation. The ratio of the flux in the poison annulus to the radial average flux in the cell varies from 1.3256 to 1.3228 or ±0.08 to -0.13 percent.

Since the iterative calculation converged so rapidly, an additional check was applied which confirmed the validity of the steady-state solution. The density change encountered during the iteration procedure was so small that the possibility of a metastable solution existed; that is, the solution to the numerical technique could have been oscillating or moving slowly to a different final value.

Rather than starting the calculation with a constant density gas, a greatly perturbed gas distribution was used as an initial guess. This initial guess contained 80 percent of average density in a central region and 110 percent of average density over a portion of the system at either end. This case converged to the same solution as the original case after three iterations. This convergence would indicate that the solution by the numerical technique is correct.

The iterative technique for steady-state conditions just described gives no information about the time-dependent behavior in the system. It was therefore postulated that the flux shift obtained from the iteration procedure could occur in one prompt neutron lifetime, approximately 3×10^{-5} second. Based on this minimum time assumption and for the axial position where the flux change was greatest, the change in the rate of internal heat generation in the gas $d\dot{q}/dt$ could be calculated. The resulting $d\dot{q}/dt$ ramp was calculated to be 1000 watts per cubic centimeter per second taking place in 3×10^{-5} second.

A transient heat-transfer calculation was performed with a $d\dot{q}/dt$ ramp of 1000 watts per second per cubic centimeter introduced for a 3×10^{-5}-second interval. The resulting temperature response (time required for the temperature to reach 90 percent of its steady-state value) lagged the heat ramp by a factor of 100; that is, the density perturbation produced by a sudden shift in $\dot{q}$ is controlled not by the prompt neutron lifetime but by the thermal diffusivity of the control gas with a hundredfold longer response time. Thus, any oscillations that might occur will be governed by the response of the gas system, and the neutron distribution can be considered as being essentially steady state between changes in density in the gas.

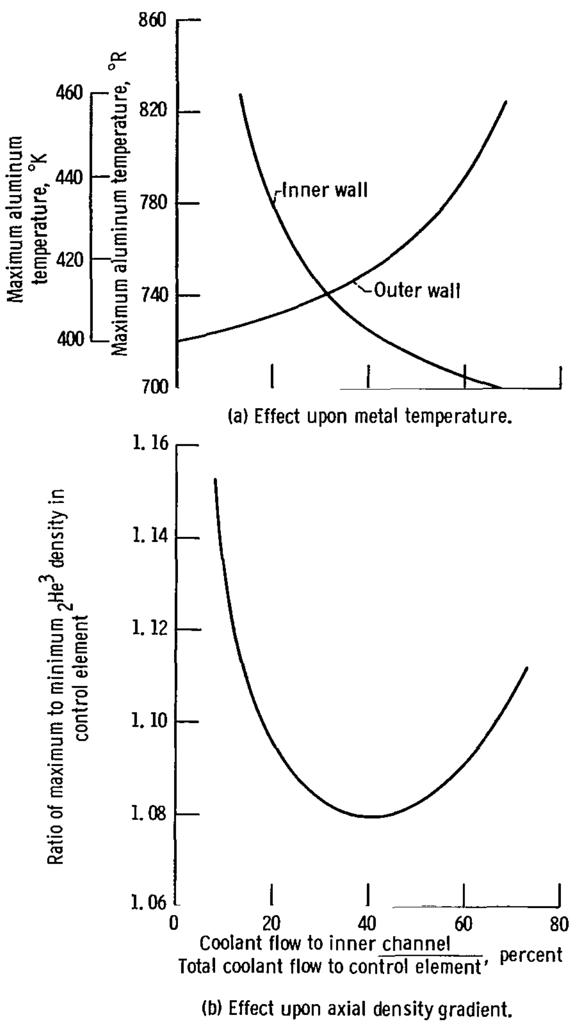


Figure 19. - Effect of coolant flow split on operating conditions of control element. Mass of $_2He^3$, 0.0218 gram per element; annulus inside diameter, 0.690 inch (1.75 cm); annulus outside diameter, 0.764 inch (1.94 cm).

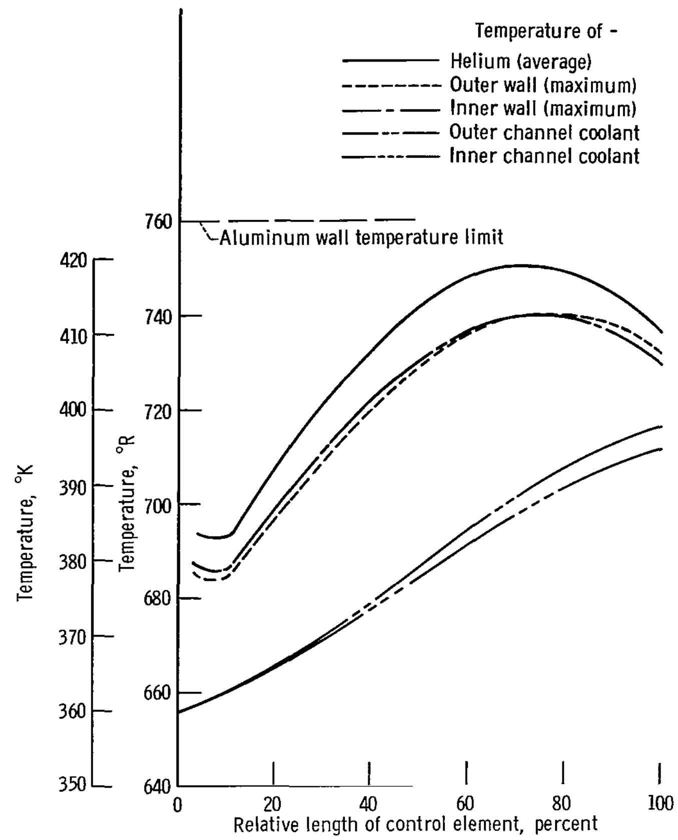


Figure 20. - Axial temperature distribution in control element. Reactor power, 100 percent.

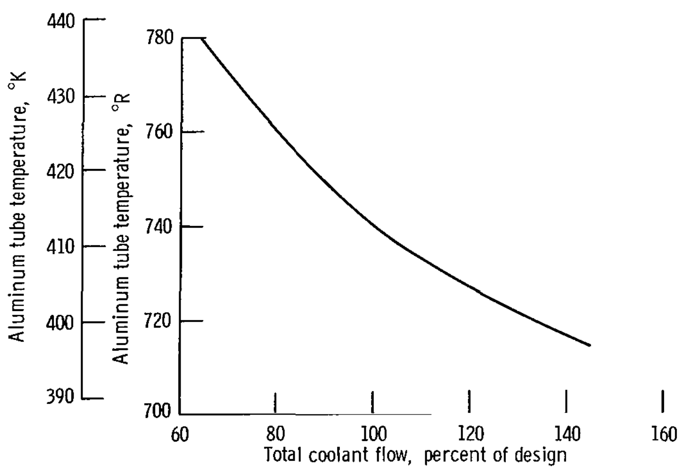


Figure 21. - Effect of coolant flow on tube temperature. Mass of $_2He^3$ in control element, 0.0218 gram; design flow per element, 15.0 gallons per minute (0.95 cu dm/sec); control annulus inside diameter, 0.690 inch (1.75 cm); control annulus outside diameter, 0.764 inch (1.94 cm); coolant inlet temperature, 656° R (365° K).

The net conclusion of these instability studies is that no sudden neutron flux perturbation due to gas redistribution seems feasible. During nonequilibrium states, while the density distribution is changing, positive reactivity of a few cents may be introduced. This would be well in the range of correction for a servomechanism.

Thermal and hydraulic analysis. - Thermal and hydraulic characteristics of the reference control element (fig. 2, p. 9) were analyzed primarily to illustrate potential problem areas encountered by the designer. Specific problem areas investigated are as follows:

(1) Effect of coolant flow split inside and outside the annulus on gas density gradients in the annulus and aluminum tube temperature: This analysis is required to establish the design flow split.

(2) Effect of off-design conditions on mass of helium 3 in a control element and helium 3 mass variations among control elements: These off-design conditions include

(a) Effect of fabrication tolerances for gas annulus

(b) Effect of radial neutron flux variations

(c) Effect of total flow and channel flow variations

Each of these problem areas is discussed subsequently.

Coolant flow split: The temperature of the metal walls of the control element and the helium 3 density gradients in the control element depend upon the flow split between the inner and the outer coolant channels. The wall temperatures of the inner and the outer tube for a constant (0.0218 g) mass of ${}_2He^3$ in a control element are shown in figure 19(a), and the axial density gradients are shown in figure 19(b). The flow split was designed to provide the maximum margin between the operating aluminum temperature and the temperature limit of 300° F (420° K).

The design flow distribution to a control element (4.8 gal/min (0.30 cu dm/sec) to the inner channel and 10.2 gal/min (0.64 cu dm/sec) to the outer channel) produces a gas density gradient which is only 1/2 percent higher than the minimum indicated in figure 19(b). According to the previous discussion of neutronic stability of the control system, 1/2 percent decrease in density gradient would cause an insignificant improvement in the axial flux distribution. The axial distribution of wall temperature, water temperature, and gas temperature for the design conditions are shown in figure 20.

Aluminum temperatures could be further reduced by increasing total coolant flow while maintaining the design flow split, as shown in figure 21. A 10-percent increase in total coolant flow would cause a 7° R (4° K) reduction in aluminum temperature.

Off-design conditions: The effect of fabrication tolerances, radial flux gradients, total flow, and channel flow variations on mass of helium 3 held in a control annulus at constant pressure are illustrated in figures 22, 23, 24, and 25, respectively. In all cases mass changes are due to gas temperature changes. The effect of channel flow reduction on wall temperature is illustrated in figure 26.

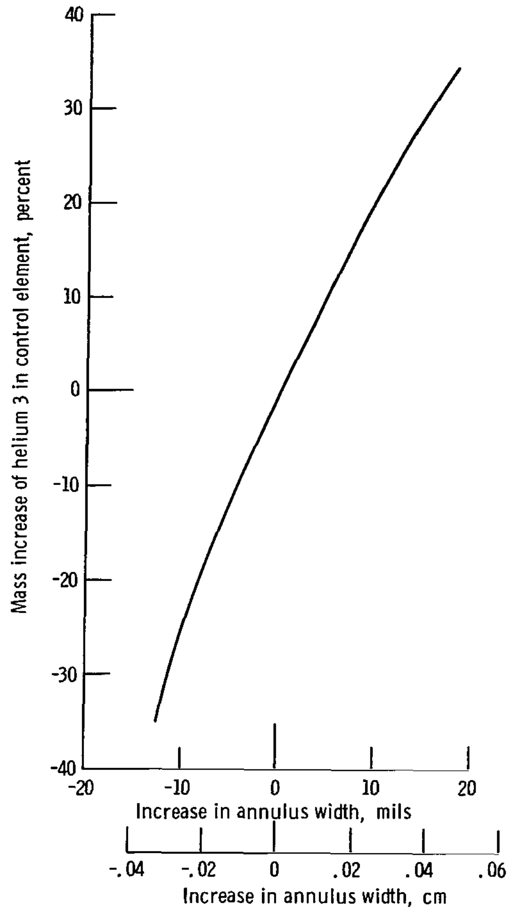


Figure 22. - Sensitivity of mass of helium 3 in control element to fabrication tolerances.

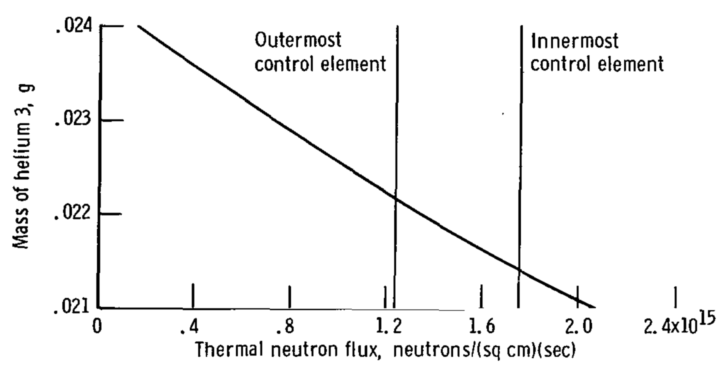


Figure 23. - Mass of helium 3 in each control element as function of neutron flux. Control annulus inside diameter, 0.690 inch (1.75 cm); control annulus outside diameter, 0.764 inch (1.94 cm); helium 3 pressure, 61 psia (42 N/sq cm); water inlet temperature, 656° R (365° K).

The variation of control annulus width from control element to control element is unavoidable because of tolerances on fabrication and assembly of control elements. The deviation from nominal of the mass of helium 3 in a control element is shown in figure 22 as a function of the deviation of annulus width. The variation in helium 3 content for the reference design at 100 percent power is approximately ±2 percent per 0.001 inch (0.00254 cm) increase or decrease in annulus width. The calculations include the effect of change in gas volume and the change in heating rate in the gas. The effect of eccentricity between inner and outer tubes was not investigated.

The heating rate and the temperature of the helium 3 in the control element depend upon the neutron flux. Since all control elements operate at the same pressure, the control elements in the regions of higher neutron flux (temperatures) have lower average helium 3 densities. Therefore, the radial variation of neutron flux causes a difference in helium 3 content among control elements. The mass of helium 3 held in each control element is shown in figure 23 as a function of neutron flux. The two vertical lines shown represent a typical flux variation that might occur across the radius of a rocket reactor. This maximum variation in neutron flux corresponds to a mass difference between control elements of 0.72 milligrams of helium 3. This is a variation of helium 3 density of ±1.7 percent.

Two types of coolant flow variation were investigated: The first is variation of total coolant flow to an element while a constant flow split is maintained between the inner and the outer channels of the control element. The second is the loss of flow to only one of the coolant channels while the flow to the other remains constant. The loss of flow to both channels is, of course, more severe than the loss of flow to either channel. The loss of flow in the outer channel is more severe than the same flow reduction in the inner channel. For example, data shown in figure 24 indicate that at constant pressure

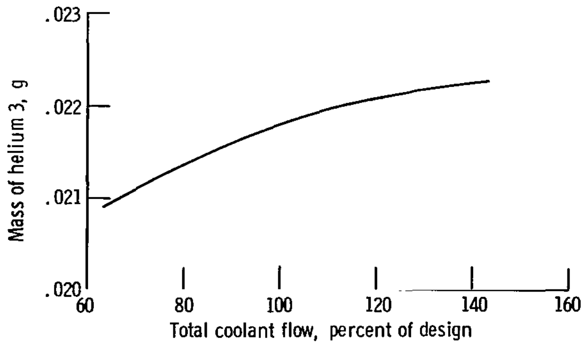


Figure 24. - Effect of coolant flow on mass of $_2He^3$ in control element. Gas pressure, 61 psia (42 N/sq cm); design flow per element, 15.0 gallons per minute (0.95 cu dm/sec); control annulus inside diameter, 0.690 inch (1.75 cm); control annulus outside diameter, 0.764 inch (1.94 cm); coolant inlet temperature, 656° R (365° K).

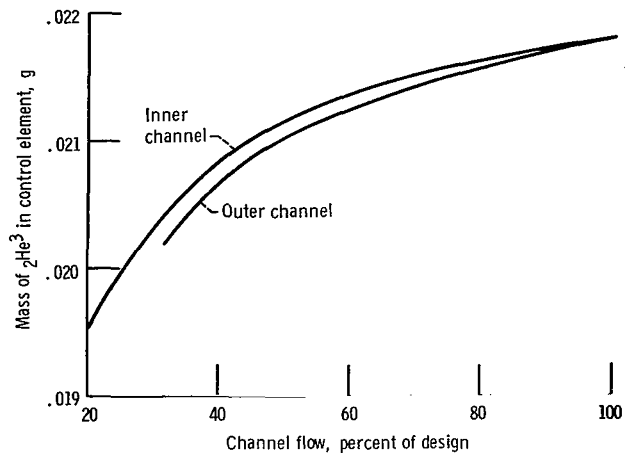


Figure 25. - Effect of channel flow reduction on helium content of control element. Reactor power, 100 percent; pressure, 61 psia (42 N/sq cm); nominal outer channel flow, 10.2 gallons per minute (0.64 cu dm/sec); nominal inner channel flow, 4.8 gallons per minute (0.30 cu dm/sec).

TABLE IV. - HELIUM 3 MASS VARIATIONS

AMONG CONTROL ELEMENTS

Condition causing variation	Variation in *condition* (a)	Variation in *mass percent* (a)
Fabrication tolerances on annulus width	±0.001 in.	±2 Percent
Radial neutron flux variations	±18 Percent	±1.7 Percent
Coolant flow variation		
Total flow to element	±5 Percent	±0.5 Percent
Channel	±5 Percent	±0.3 Percent
Total		±4.5 Percent

[a]Variations are from nominal or design conditions.

(61 psia; 42 N/sq cm) a 10-percent reduction in total flow would cause a 1-percent loss of helium 3 from an element. A 10-percent reduction in inner channel flow would cause a 0.3-percent loss of helium 3, while the same reduction in outer channel flow would cause a 0.5-percent loss of gas (fig. 25).

Typical tolerable design variations and resulting helium 3 mass variations are summarized in table IV. The total variation shown (±4.5 percent) is less than the ±5 percent variation allowed in table I (p. 8).

The effect of channel flow reduction on aluminum wall temperature is illustrated in figure 26. During operation at a constant system pressure, a 10-percent loss of flow to the inner or the outer channel would cause a 6° R (3° K) or an 8° R (4° K) increase in wall temperature, respectively. This illustrates the possible effect on wall temperature of a blockage or flow restriction in a coolant passage.

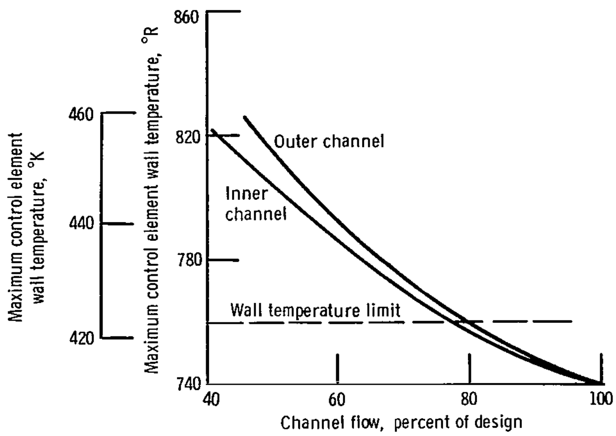


Figure 26. - Effect of channel flow reduction on wall temperature. Reactor power, 100 percent; pressure, 61 psia (42 N/sq cm); nominal outer channel flow, 10.2 gallons per minute (0.64 cu dm/sec); nominal inner channel flow, 4.8 gallons per minute (0.30 cu dm/sec).

None of the problems discussed previously negate the feasibility of the gaseous control system concept. All the problems may, however, affect the design of the system and should be investigated thoroughly before the designer establishes reference control systems. In many instances it may be necessary to compromise one or more of the design objectives in order to satisfy others.

CONCLUDING REMARKS

The concept of reactor control using helium 3 in metal control elements located in the moderator region appears feasible. This method of reactor control offers several desirable features:

(1) Low system mass: The mass of the reference control system excluding isolation valves and control instrumentation is about 142 kilograms or about 2.2 percent of the total reactor engine weight.

(2) Neutron flux perturbation: The helium 3 is uniformly distributed throughout the control system during the reactor exposure cycle, and the resulting perturbation of the neutron flux distribution is small.

(3) No chemical processing: The helium 3 is chemically inert. If the exposure is short and helium 3 burnup is small in the reference system, no chemical processing or purification is required.

The consequences of a failure within the control system can be limited to mild excursions by using a flow-restricting orifice in the system and by providing a positive water pressure seal on the control system. These excursions may readily be sensed and controlled by the scram system. A reactor restart capability after such situations could be provided by using duplicate but independent control systems.

Power coefficients of reactivity associated with a gaseous poison system can be made small and in some cases can be made negative by taking the following design precautions:

(1) Selecting a thin gas space for heat transfer in the control element to minimize gas temperature and density gradients

(2) Installing projectile "traps" in the control system for capturing protons and tritons in the metal walls of the system; this procedure reduces the heating and therefore the temperature and density gradients in the system

(3) Placing the gas distribution system in a region of low nuclear importance to minimize neutron flux perturbations

(4) Selecting the distribution system gas space such that it operates at a slightly higher temperature than the control element; this will cause poison to be displaced into a region of higher nuclear importance (negative power coefficient) when power is increased

Lewis Research Center,
National Aeronautics and Space Administration,
Cleveland, Ohio, March 23, 1967,
122-28-03-05-22.

APPENDIX A

ALTERNATIVE CONTROL-ELEMENT CONFIGURATION

The control system discussed previously may be difficult to fabricate because a large number of annuli made of thin-wall aluminum tubing would have to be joined to a distribution system, as shown in figure 2 (p. 9). Alinement would be crucial, since ports in the distribution system require accurate positioning to connect with the thin annuli. Manufacturing difficulties may thus be a major drawback to an annular system.

To ease the fabrication problem, it is desirable to incorporate the distribution system as an integral part of the gas system, a type of printed gas circuit. A device which satisfies these requirements is shown in figure 27. This control element is hexagonal and completely surrounds the fuel element. This element contrasts with the annular control element, which is located in the interstices between fuel elements. The two concepts (interstitial and circumferential) are compared in figure 28. Figure 28(a) represents a section taken through the distribution system and control element of the

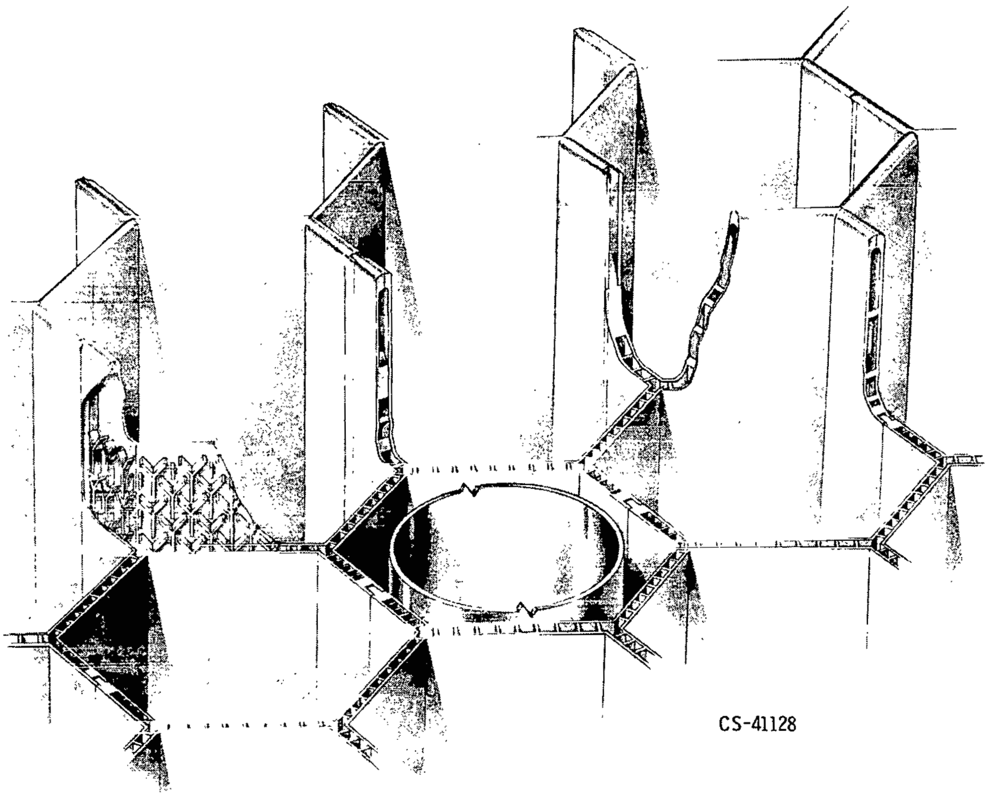
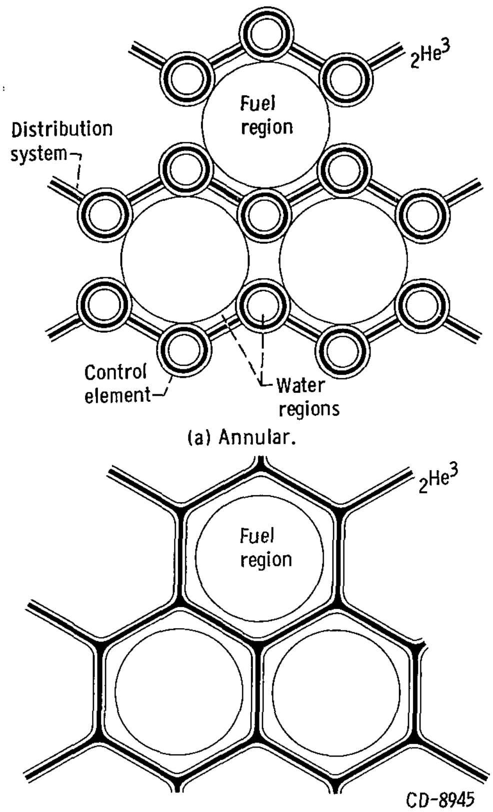


Figure 27. - Alternative control system configuration.



(b) Hexagonal.
Figure 28. - Control elements.

interstitial-type system. Figure 28(b) represents a section at any elevation in the circumferential system. The circumferential-control-element concept could also incorporate the "proton-traps" or "baffles" shown in figure 27.

The circumferential-control-element concept has the additional advantage that the assembled structure could be used as a structural support member for the top tube sheet holding the fuel support tubes. Since all elements are interconnected, the feed header can be incorporated integrally. Two manufacturing methods seem feasible (fig. 29). One (fig. 29(a)) envisions an extruded module, the surfaces of which are etched or machined to contain the proton traps according to patterns illustrated in figure 30. (The etching technique has been demonstrated with a 2- by 3-in. (5- by 7.5-cm) aluminum plate. The plate was originally 0.060 in. (0.15 cm) thick and the "Y" pattern shown in fig. 30 was chemically etched in it. The resulting pattern contained 0.040-in.- (0.10-cm) high ribs and 0.020-in.- (0.050-cm) thick plate.)

The identical modules are then assembled into a honeycomb array and the entire assembly sealed by suitable techniques. For a single gas system, the only seal required is at the top and bottom faces of the assembled honeycomb, places which are easily accessible for welding and testing. The dual-circuit system requires additional longitu-

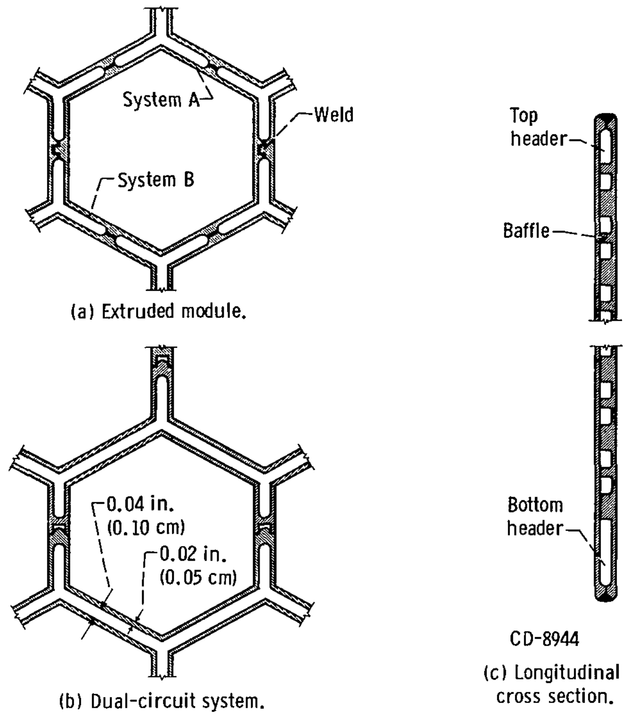


Figure 29. - Possible configurations for circumferential elements.

dinal welds (to separate the two circuits), which are believed feasible from the manufacturing point of view. The entire honeycomb can be prefabricated and inserted to press mating faces of adjacent modules together, which will add to the integrity of the system.

Figure 29(b) illustrates an alternative manufacturing concept of a dual-circuit system, which obviates the need for leak-tight longitudinal welds during assembly, because each gas circuit can be tested for leak tightness prior to assembly. The entire honeycomb can be preassembled as in the first design.

Figure 29(c) shows a longitudinal cross section. For optimum design, the bottom header gas space should be wider than the active gas channel (not illustrated). Other designs and manufacturing schemes are conceivable, such as a system similar to corrugated cardboard; even a honeycomb with three independent and uniformly distributed gas systems is conceivable.

The patterns of feeding and connecting the gas passages in the honeycomb assembly is illustrated by two schemes. Figure 31(a) illustrates a simple scheme which is similar to the header arrangement envisioned for the annulus or interstitial concept. Also shown is the circumferential manifold feeding the two circuits. In studying figure 31(a), it must be remembered that a fuel element is centered inside every hexagon. Each fuel element is surrounded by control gas, with equal contributions from both circuits.

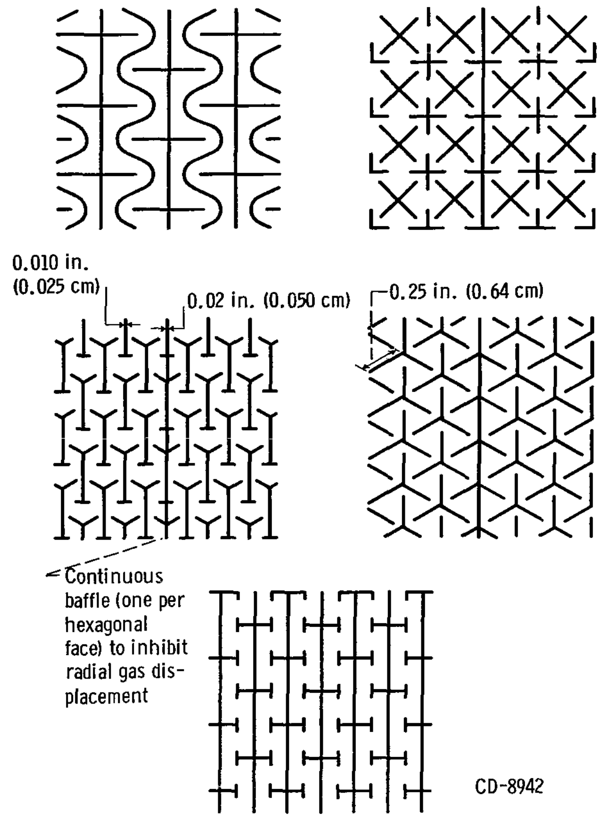
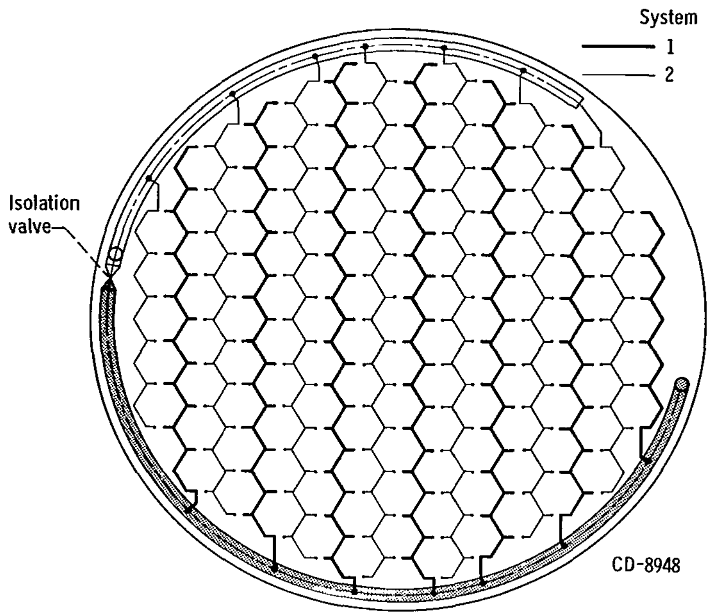
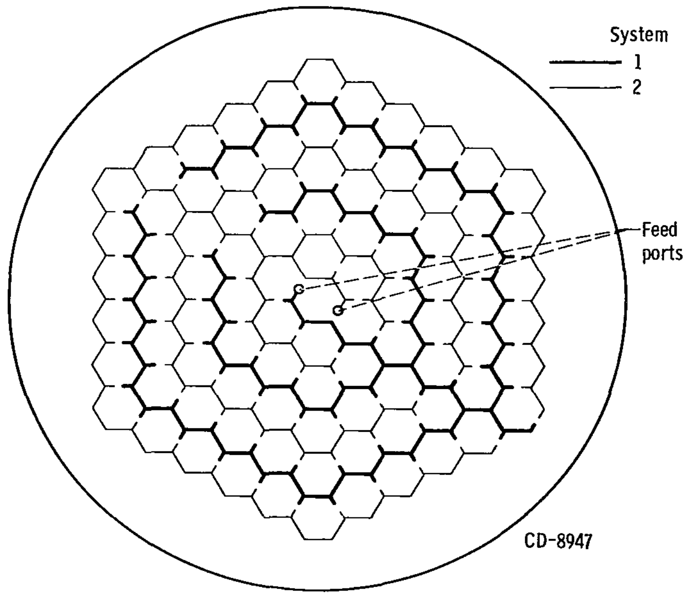


Figure 30. - Possible "proton-trap" patterns.

Figure 31(b) illustrates a radially symmetric layout designed to prevent radial gas displacement from the high-flux central region to the low-flux peripheral region. The continuous vertical lines in the patterns shown in figure 30 were incorporated into the designs to prevent radial displacement of the gas. The layout shown in figure 31(b) envisions a central feed point for each circuit in order to get the quickest response in the high-importance region.



(a) With circumferential feed.



(b) With central feed.

Figure 31. - Two-system header arrangement.

APPENDIX B

POWER COEFFICIENT ASSOCIATED WITH CONTROL ELEMENT

The power coefficient of reactivity[3] for a control element and its associated portion of a distribution line were calculated with a one-velocity (thermal) unreflected neutronic model and a one-dimensional heat-transfer model. The calculated power coefficient of reactivity is shown in figure 5 (p. 13). The general procedure for calculating the power coefficient follows:

(1) The reactivity in the control element, $\delta k/k$ was assumed to be proportional to the density in the control element, relative to the density in the control element at 100 percent power, that is,

$$\frac{\delta k}{k} = -K_c\left(\frac{\delta\rho}{\rho_L}\right)_c = -K_c\left(\frac{\rho - \rho_L}{\rho_L}\right)$$

(All symbols are defined in appendix E.)

(2) The reactivity worth of the distribution line was evaluated as a function of

(a) The size of the distribution line

(b) The flux in the distribution line relative to the flux in the control element

(c) The gas density in the distribution line

(d) The relative worth of the control element K_c

(3) The combined power coefficient for the control element and distribution line was obtained by statistically weighting the worths of each.

(4) The relation for density of gas in the control element and distribution line is expressed in terms of the gas temperatures by the perfect gas relation.

(5) The relation between reactivity changes and reactor power was obtained by heat-transfer calculations which express gas temperature in the control element and in the distribution line as a function of reactor power.

Reactivity Worth of Control Element

The worth of the control element to the cell was assumed to be due to the change in cell thermal utilization only and as shown in reference 7 can be expressed approximately as

---

[3]Reactivity change per unit change in reactor power $(\delta k/k)/\delta P$.

$$\frac{\delta k}{k} = -\frac{fA_c\varphi_c\delta\Sigma_{a,c}}{\Sigma_{a,F}A_F\varphi_F} = -\frac{f\Sigma_{a,c,L}A_c\varphi_c}{\Sigma_{a,F}A_F\varphi_F}\frac{\delta\rho_c}{\rho_{c,L}} \tag{B1}$$

where the subscripts F and c refer to the fuel region and control element; the subscript L represents conditions at 100 percent power.

When all the gas is removed from the control system, $(\delta\rho/\rho_L)_c = -1.00$ and

$$\frac{\delta k}{k} = \frac{f\Sigma_{a,c,L}A_c\varphi_c}{\Sigma_{a,F}A_F\varphi_F} = K_c \tag{B2}$$

The value of $K_c = 7.44$ percent ($10.7 from table I, p. 8). The power coefficient for the control element is

$$\left(\frac{1}{k}\frac{\delta k}{\delta P}\right)_c = -K_c\frac{1}{\rho_{c,L}}\frac{\delta\rho_c}{\delta P} \tag{B3}$$

Reactivity Worth of Distribution Duct

The change in reactivity worth of the distribution duct was also assumed to be due to the change in f and can be expressed as

$$\left(\frac{\delta k}{k}\right)_d = \frac{\delta f}{f} = -\frac{f\Sigma_{a,c,L}A_c\varphi_c}{\Sigma_{a,F}A_F\varphi_F}\left(\frac{\delta\Sigma_{a,d}A_d\varphi_d}{\Sigma_{a,c,L}A_c\varphi_c}\right) \tag{B4}$$

where the subscript d refers to the distribution duct.

The term $f\Sigma_{a,c,L}A_c\varphi_c/\Sigma_{a,F}A_F\varphi_F$ represents the ratio of neutron absorptions in the control element to total absorptions and is equal to K_c (from eq. (B2)). Therefore,

$$\left(\frac{\delta k}{k}\right)_d = -K_c\left(\frac{A_d}{A_c}\right)\left(\frac{\varphi_d}{\varphi_c}\right)\left(\frac{\rho_d}{\rho_{c,L}}\right)\left(\frac{\delta\rho_d}{\rho_d}\right) \tag{B5}$$

The power coefficient due to the distribution duct is

$$\frac{1}{k_d}\frac{\delta k_d}{\delta P} = -\left[K_c\left(\frac{A_d}{A_c}\right)\left(\frac{\varphi_d}{\varphi_c}\right)\left(\frac{\rho_d}{\rho_{c,L}}\right)\right]\frac{1}{\rho_d}\frac{\delta\rho_d}{\delta P} \tag{B6}$$

Combined Power Coefficient for Cell

The power coefficient for the combined control element and distribution duct is the sum of the statistically weighted worths of each:

$$\frac{1}{k}\frac{\delta k}{\delta P} = \frac{\left(\frac{1}{k}\frac{\delta k}{\delta P}\right)_d \tau_d \varphi_d^2 + \left(\frac{1}{k}\frac{\delta k}{\delta P}\right)_c \tau_c \varphi_c^2}{\tau_d \varphi_d^2 + \tau_c \varphi_c^2} \tag{B7}$$

This equation can be expressed as

$$\frac{1}{k}\frac{\delta k}{\delta P} = \frac{-K_c\left[1 + \frac{A_d}{A_c}\frac{\tau_d}{\tau_c}\left(\frac{\varphi_d}{\varphi_c}\right)^3 \frac{\delta\rho_d}{\delta\rho_c}\right]\frac{1}{\rho_{c,L}}\frac{\delta\rho_c}{\delta P}}{\left[1 + \frac{\tau_d}{\tau_c}\left(\frac{\varphi_d}{\varphi_c}\right)^2\right]} \tag{B8}$$

In this equation densities, fluxes, and volumes are unknown.

Relation Between Gas Densities and Temperatures

The change in gas density in each region was calculated from the perfect gas law:

$$\delta p_d = \mathscr{R}(\rho_d \delta T_d + T_d \delta\rho_d) \tag{B9}$$

$$\delta p_c = \mathscr{R}(\rho_c \delta T_c + T_c \delta\rho_c) \tag{B10}$$

Since the control element and the distribution line are connected, the equilibrium pressures in the two systems are equal:

$$\delta p_c = \delta p_d \tag{B11}$$

On the basis of the conservation of mass in the two systems,

$$\delta(\rho_d \tau_d) + \delta(\rho_c \tau_c) = 0$$

or

$$\frac{\delta\rho_d}{\delta\rho_c} = -\frac{\tau_c}{\tau_d} \tag{B12}$$

where τ represents the system volume. From equations (B9) and (B10),

$$\left[T_c - T_d\left(\frac{\delta\rho_d}{\delta\rho_c}\right)\right]\frac{1}{\rho_c}\,\delta\rho_c = \left(\frac{\rho_d}{\rho_c}\,\frac{\delta T_d}{\delta T_c} - 1\right)\delta T_c \tag{B13}$$

But from equation (B12), $\delta\rho_d/\delta\rho_c = -\tau_c/\tau_d$; therefore,

$$\left[T_c + \left(\frac{\tau_c}{\tau_d}\right)T_d\right]\frac{1}{\rho_c}\,\delta\rho_c = \left(\frac{\rho_d}{\rho_c}\,\frac{\delta T_d}{\delta T_c} - 1\right)\delta T_c \tag{B14}$$

When equation (B14) is divided by the power change δP, and the resultant equation is solved for the power coefficient of density $(1/\rho_c)(\delta\rho_c/\delta P)$,

$$\frac{1}{\rho_c}\,\frac{\delta\rho_c}{\delta P} = +\frac{\dfrac{\rho_d}{\rho_c}\,\dfrac{\delta T_d}{\delta T_c} - 1}{T_c + \left(\dfrac{\tau_c}{\tau_d}\right)T_d}\,\frac{\delta T_c}{\delta P} \tag{B15}$$

When the relation for $\delta\rho_d/\delta\rho_c$ and $(1/\rho_c)(\delta\rho_c/\delta P)$ from equations (B12) and (B15), respectively, are substituted into equation (B8),

$$\frac{1}{k}\,\frac{\delta k}{\delta P} = -K_c\left[\frac{1 - \dfrac{A_d}{A_c}\left(\dfrac{\varphi_d}{\varphi_c}\right)^3}{1 + \dfrac{\tau_d}{\tau_c}\left(\dfrac{\varphi_d}{\varphi_c}\right)^2}\right]\left\{\frac{\left[\left(\dfrac{T_c}{T_d}\right)\left(\dfrac{\dfrac{\delta T_d}{\delta P}}{\dfrac{\delta T_c}{\delta P}}\right) - 1\right]\dfrac{\delta T_c}{\delta P}}{\left[1 + \dfrac{T_d}{T_c}\left(\dfrac{\tau_c}{\tau_d}\right)\right]T_c}\right\} \tag{B16}$$

Relation Between Gas Temperature and Reactor Power

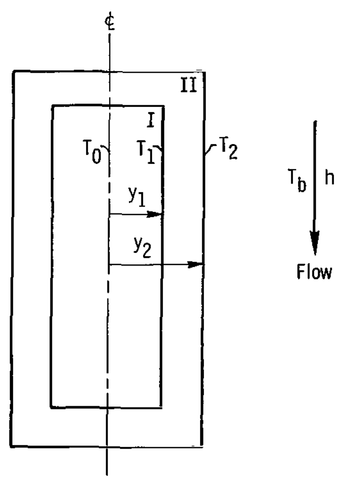
The average gas temperature in the control element and in the distribution line are required to determine the values of T_c, T_d, $\delta T_c/\delta P$, and $\delta T_d/\delta P$ in equation (B16).

The average gas temperature in the control element (fig. 32) was calculated as a function of reactor power (neutron flux) by using the heat-transfer model discussed in appendix D. The temperature in the control element was calculated for the following conditions:

Annulus outside diameter, in. (cm)	0.764 (1.94)
Annulus inside diameter, in. (cm)	0.690 (1.75)
Length of control element, in. (cm)	42 (107)
Coolant inlet temperature, $^{\circ}R$ ($^{\circ}K$)	656 (365)
Gas pressure, psia (N/cm^2)	61 (42)

The temperature of the gas in the distribution line was calculated from the following model:

(1) Heat transfer is one dimensional as shown in sketch (a).



(a)

(2) Constant thermal conductivity k and volumetric heat generation q are in regions I and II.

(3) Heat is generated in the helium 3 in region I because of the ${}_2He^3\left({}_0n^1, {}_1p^1\right){}_1T^3$ reaction.

(4) Heat is removed from the gas by conduction and is transferred to the coolant flowing with a constant bulk temperature T_b and constant convective heat-transfer coefficient h.

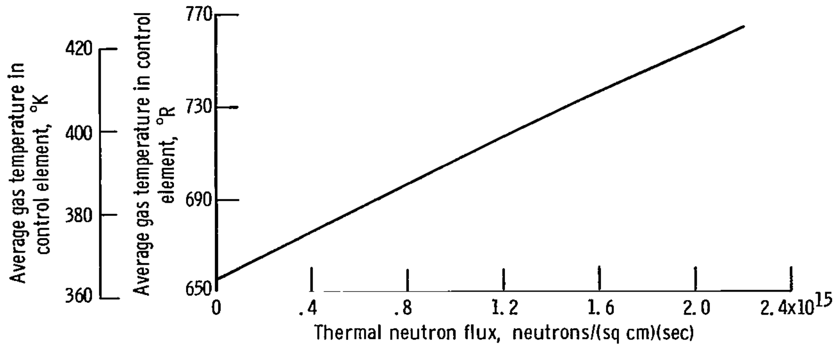


Figure 32. - Average gas temperature in control element as function of thermal neutron flux. Control annulus inside diameter, 0.690 inch (1.75 cm); control annulus outside diameter, 0.764 inch (1.94 cm); helium 3 pressure, 61 psia (42 N/sq cm); water inlet temperature, 656° R (365° K).

The temperature in the gas is

$$T = T_1 + q_g \frac{(y_1^2 - y^2)}{2k_g} \tag{B17}$$

The average gas temperature $\overline{T}$ is

$$\overline{T} = \frac{\int_0^{y_1} T\,dy}{y_1} = T_1 + \frac{q_g y_1^2}{3k_g} \tag{B18}$$

The temperature rise across the distribution line wall and convective film are

$$T_1 - T_2 = \frac{q_m\left(y_2^2 - y_1^2\right)}{2k_m} + \frac{(q_g - q_m)y_1(y_2 - y_1)}{k_m} \tag{B19}$$

$$T_2 - T_b = \frac{qy_1}{h} + \frac{q_m(y_2 - y_1)}{h} \tag{B20}$$

From equations (B18), (B19), and (B20),

$$\overline{T} = T_b + \left[\frac{qy_1}{h} + \frac{q_m(y_2 - y_1)}{h}\right] + \left[\frac{q_m\left(y_2^2 - y_1^2\right)}{2k_m} + \frac{(q_g - q_m)y_1(y_2 - y_1)}{k_m}\right] + \frac{q_g y_1^2}{3k_g} \tag{B21}$$

The bulk coolant temperature adjacent to the distribution line T_b can be expressed in terms of the inlet temperature to the reactor $T_{b,i}$ and the coolant temperature rise across the reactor core ΔT_c.

$$\Delta T_c = \left(\frac{\Delta T_{c,L}}{\varphi_{c,L}}\right)\varphi_c \tag{B22}$$

where $\Delta T_{c,L}$ (30^{o} K) is the average temperature rise across the core at 100 percent power and $\varphi_{c,L}$ (1.5×10^{15}) is the average neutron flux at 100 percent power.

$$T_b = T_{b,i} + \Delta T_c$$

The volumetric heating rate in the helium 3 from appendix D is

$$q_g = 1.17\times10^{-10}\, S\rho_g\varphi_d, \qquad \mathrm{W/cu\ cm} \tag{D18}$$

where gas density ρ_g is in grams per cubic centimeter.

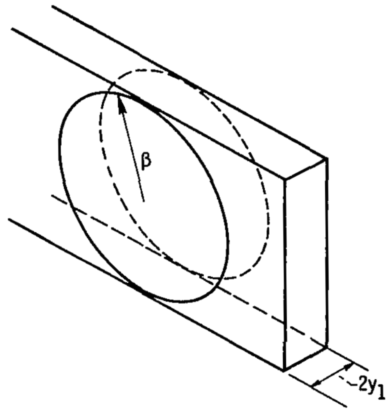
The heat generated in the wall of the distribution line q_m is due to (1) gamma attenuation in the wall q_γ and (2) absorption of those protons which are generated in the gas but not absorbed in it q_p.

$$q_\gamma = 8.03\times10^{-14}\,\varphi_d, \qquad \mathrm{W/cu\ cm}$$

$$q_p = \left[1.17\times10^{-10}\,\rho_g(1 - S)\right]\left(\frac{y_1}{y_2 - y_1}\right)\varphi_d, \qquad \mathrm{W/cu\ cm}$$

$$q_m = q_\gamma + q_p, \qquad \mathrm{W/cu\ cm} \tag{B23}$$

The fraction of the energy generated in the ${}_2He^3\left({}_0n^1,\ {}_1p^1\right){}_1T^3$ reaction which is absorbed in the gas was calculated from the model discussed in appendix D. In this model (sketch (b)) it is assumed that the average distance which a proton or triton



(b)

travels $\bar{\beta}$ is the radius of a sphere the volume of which is equal to the volume in which the particle can be stopped in the distribution lines. This volume can be approximated by a disk the radius of which is β and the thickness of which is the thickness of the distribution line $2y_1$.

$$\frac{4}{3}\pi\bar{\beta}^3 = \pi\beta^2 2y_1$$

$$\bar{\beta} = \left(\frac{3}{2}y_1\beta^2\right)^{1/3}$$

The average range of the particle is

$$\rho_g\bar{\beta} = \rho_g\left(\frac{3}{2}y_1\beta^2\right)^{1/3}$$

$$\beta = \frac{\psi}{\rho_g}$$

where ψ is the range of the particle in helium.

$$\rho_g\bar{\beta} = \rho_g^{1/3}\left(\frac{3}{2}y_1\psi^2\right)^{1/3}$$

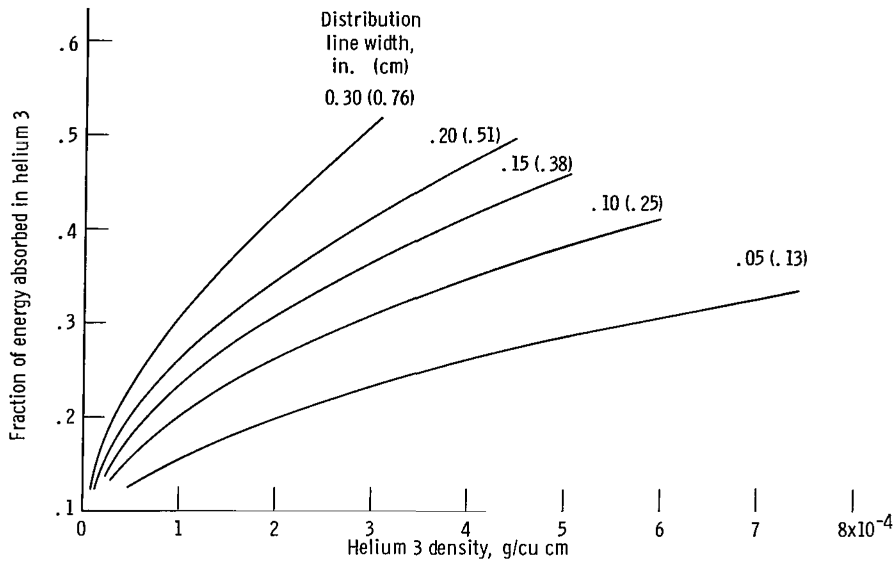


Figure 33. - Fraction of energy absorbed in helium 3 distribution ducts.

The energy of the particle can be expressed as

$$E = \left(E_s^{1.75} - 483\, \rho\bar{\beta}\right)^{0.571}$$

where E_s is the initial energy of the particle (MeV) (see eq. (D22)). The fraction of kinetic energy of the particles absorbed within the gas S is

$$S = \frac{(E_s - E)_p + (E_s - E)_{tr}}{E_{s,p} + E_{s,tr}} \tag{B24}$$

where the subscripts p and tr refer to the proton and the triton.

$$S = 1 - \frac{\left[E_{s,p}^{1.75} - 483\, \rho^{1/3}\left(\frac{3}{2}\, y_1 \psi_p^2\right)^{1/3}\right]^{0.571} + \left[E_{s,tr}^{1.75} - 483\, \rho^{1/3}\left(\frac{3}{2}\, y_1 \psi_{tr}^2\right)^{1/3}\right]^{0.571}}{E_{s,p} + E_{s,tr}}$$

The fraction of the energy absorbed is shown in figure 33 as a function of the distribution system thickness $2y_1$ and the gas density ρ. The following values of $E_{s,p}$, $E_{s,tr}$, ψ_p and ψ_{tr} were used in the calculations of figure 33:

Initial energy of proton, $E_{s,p}$, MeV (J) . . . . 0.573 (0.918×10^{-13})
Range value of proton, ψ_p, g/sq cm . . . . 7.55×10^{-4}
Initial energy of triton, $E_{s,tr}$, MeV (J) . . . . 0.191 (0.306×10^{-13})
Range value of triton, ψ_{tr}, g/sq cm . . . . 0.866×10^{-4}

The flux normalized heat generation rates in the helium 3 and in the distribution line wall were obtained from equations (D18) and (B23) and figure 33. The heat generation rates are shown in figure 34 as a function of the helium 3 density and gas distribution line width. The distribution line wall thickness $y_2 - y_1$ was assumed to be 0.060 inch (0.15 cm).

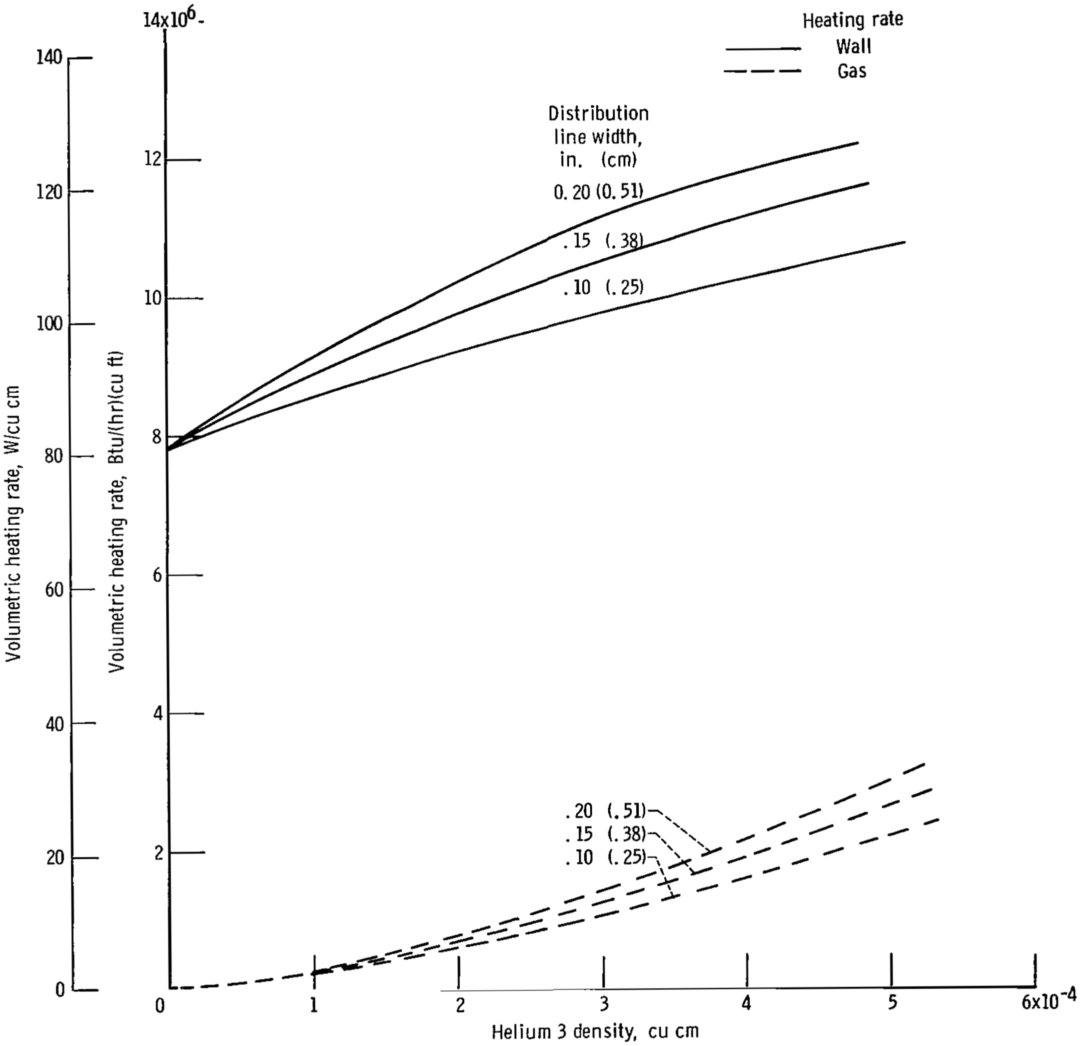


Figure 34. - Heating rates in distribution duct. Normalized to flux of 10^{15} neutrons per square centimeter per second; wall thickness, 0.060 inch (0.15 cm); aluminum walls; gamma heat contribution, 7.8×10^6 Btu per hour (81 W/cu cm).

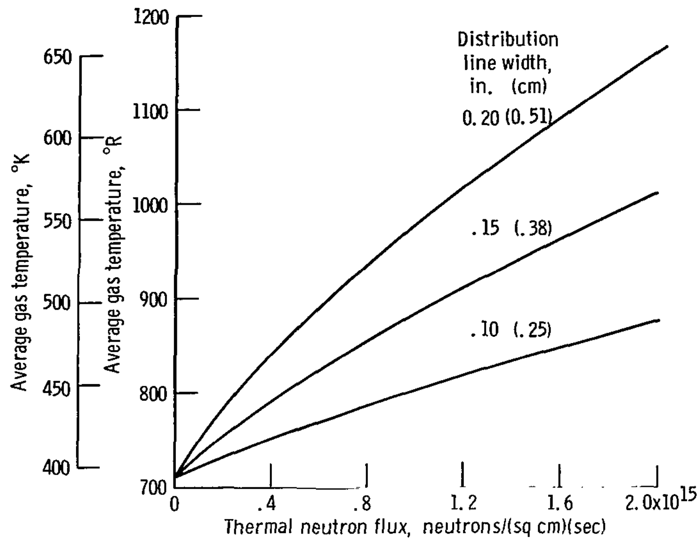


Figure 35. - Effect of neutron flux level on helium 3 temperature in distribution duct. Pressure, 61 psia (42 N/sq cm); water inlet temperature, 656° R (365° K).

In figure 35, the average gas temperature in the distribution line is shown as a function of distribution line thickness and neutron flux in the gas with the use of equations (B21) and (B22) and values of q_g and q_m obtained from figure 34. The gas temperature and density are related by the perfect gas relation $\rho = p/\mathscr{R}T$. The values shown in figure 34 are based upon the following parameters:

Helium 3 pressure, psia (N/sq cm) 61 (42)
Coolant inlet temperature, $T_{b,i}$, ^{o}R (^{o}K) 656 (365)
Helium 3 thermal conductivity, k_g, Btu/(hr)(ft)(^{o}F) [J/(sec)(cm)(^{o}K)] . 0.126 (0.00218)
Aluminum thermal conductivity, k_m, Btu/(hr)(ft)(^{o}F) [J/(sec)(cm)(^{o}K)] . . . 115 (1.99)
Forced convection heat transfer, h, Btu/(hr)(sq ft)(^{o}F)
[J/(sec)(sq cm)(^{o}K)] 3360 (1.91)

Calculation of Power Coefficient for Reference System

The power coefficient for the control element plus distribution line was calculated for operation at 100 percent power and is shown in figure 5 (p. 13). The values shown were calculated from equation (B16) with the use of the following values:

(1) A_d/A_c: The ratio of the distribution line area to the control-element area taken in a plane perpendicular to the axis of the reactor is

$$\frac{A_d}{A_c} = \frac{18.48\ t}{\pi\left[(1.94)^2 - (1.75)^2\right]} = 8.38\ t$$

where t is the width of the distribution line (cm).

(2) τ_d/τ_c: The ratio of the actual volume of the distribution line to the actual volume of the control element τ_d/τ_c is

$$\frac{\tau_d}{\tau_c} = \frac{4.62\ A_d'}{58.9} = 0.0786\ A_d'$$

where A_d' is the flow cross-sectional area of the distribution duct (sq cm). This area is also so sized that the flow disparity among control elements on a distribution duct is less than 5 percent. The calculation was based upon a laminar, steady state, isothermal compressible flow model. The result yields the following requirement:

$$D_{e,d}^2 A_d' = \frac{4t^3h^3}{(t + h)^2} = 1.27\ cm^4$$

where

$D_{e,d}$ equivalent diameter of distribution line, cm

h height of distribution line, cm

(3) K_c: The worth of helium 3 in the control element at 100 percent power is 7.44 percent; therefore, K_c is 7.44 percent.

(4) $\delta T_c/\delta P$: The temperature coefficient in the control element is obtained from the slope of figure 32. Where 100 percent power corresponds to an average flux of 1.5×10^{15} neutrons per square centimeter per second at 1500-megawatt reactor power. The value of $\delta T_c/\delta P$ is approximately 0.049° R per megawatt (0.027° K/MW).

(5) $\delta T_d/\delta P$: The temperature coefficient in the distribution line is obtained from the slopes of figure 35 as a function of the relative flux in the distribution line and the size of the distribution line.

(6) T_c and T_d: The average temperatures in the control element and distribution lines at 100 percent power are obtained directly from figures 32 and 35. The value of T_c is approximately 730° R (405° K).

APPENDIX C

PREPARATION OF NUCLEAR CROSS SECTIONS AND NUCLEAR CALCULATIONS

Throughout the nuclear calculations performed in this study, a six-group energy split was used to represent nuclear cross sections for neutron interactions. This six-group energy split is outlined in table V.

The neutronic calculations for a study such as this one must be sufficiently short that a range of parameters may be investigated. For this reason, most of the calculations were performed in one dimension. The main working program used in this analysis was TDSN, a two-dimensional discrete angular segmentation transport approximation program written at the Lewis Research Center (ref. 8). This program was used for one- and two-dimensional calculations with the use of the S_4 discrete ordinate approximation.

The reactor was divided into several axial regions. As many as five axial regions were used to represent the neutron reflector at the inlet end of the reactor. The volume fractions and atom densities of each region were calculated for a unit cell area in the reactor. The material atom densities were then employed in programs (refs. 9 and 10) to calculate a neutron spectrum and transport-corrected P_0 cross-section values averaged over this spectrum.

All calculations including a core region assumed a uniform fuel distribution throughout. Different core region cross sections used involved changes in helium 3 density only. Microscopic cross sections and region atom densities were used to generate cross-section values for the regions within the unit cell set up for radial calculations. These radial calculations were performed with the use of the computer programs RP-1 and FLAIR. RP-1 is a multigroup, multiregion, one-dimensional diffusion theory program that was used to produce flux-weighted cross sections to be used in the central material region of FLAIR. FLAIR is a triangular-mesh, multigroup, multiregion,

TABLE V. - SIX-GROUP ENERGY STRUCTURE

Group	Energy range (a)	Lethargy range
1	0.821 - 10.0 MeV	0 - 2.5
2	5.531 keV - 0.821 MeV	2.5 - 7.5
3	454 eV - 5.531 keV	7.5 - 10
4	3.06 - 454 eV	10 - 15
5	0.414 - 3.06 eV	15 - 17
6	0 - 0.414 eV	17 - ∞

[a]1 MeV = 1.602×10^{-13} J.

diffusion code written for problems of hexagonal geometry, Neither FLAIR nor RP-1 is capable of handling the cell correctly. RP-1 is required to flux-weight the cylindrical fuel region, while FLAIR is needed to include the triangular symmetry of control elements. A combination of the two programs is thus necessary for the cell analysis. The final output from FLAIR and RP-1 provides flux-weighting factors for each material zone within the unit cell. These factors, in turn, are used to calculate smeared cross sections for a unit cell.

The present cross-section data available on the library tapes for the GAM-II (ref. 9) and TEMPEST (ref. 10) programs do not include cross sections for helium 3. However, reference 11 indicates that the absorption cross section is inversely proportional to the neutron speed. The absorption cross section at 2200 meters per second for helium 3 is listed as 5500 barns (5500×10^{-28} sq m). The combination of these two pieces of information provided energy dependent cross sections for helium 3.

Locating the gas distribution system within the core region of a reactor eliminates the problem of depletion of reflector worth due to poison passages through the reflector. However, the local flux depression and reactivity effects produced by the distribution system within the core region must be investigated.

Cross sections used to represent the distribution system in a one-dimensional axial calculation were flux-weighted with the results of the two-dimensional r, z calculation of figure 10 (p. 19). The r, z geometry assumes that the distribution system completely encircles a fuel element. This system encircles only about 80 percent of the fuel assembly (fig. 2, p. 9). Therefore, the final cross sections for this axial region were 80 percent of the r, z flux-weighted cross section and 20 percent of those from a no-poison cell.

The neutron spectrum existing within the helium 3 plenum is not easily determined. The plenum is surrounded by water in the reactor, and an approximation of the spectrum can be made by mixing the helium with the surrounding water and calculating the spectrum that would exist in an infinite medium of that composition. The amount of surrounding water to be included in this mixture is uncertain. Calculations made with TEMPEST, a neutron thermalization program, indicate, however, that spectral hardening effects vary slowly over a large range of poison concentrations. (The average thermal group cross section is shown in figure 36 to illustrate this effect). Thus, a spectrum was generated with approximately three diffusion lengths of water surrounding the helium tube and was used for all helium 3 plenum calculations. This reference composition, marked by the line at the right in figure 36, was 13.79×10^{-5} atoms of helium per molecule of water.

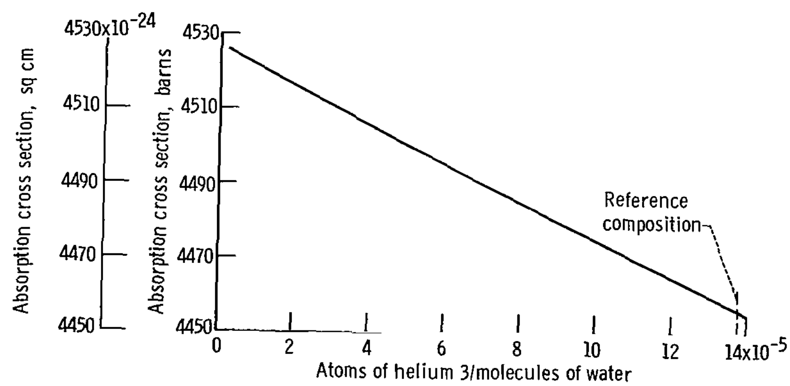


Figure 36. - Spectral effects on helium 3 microscopic cross section.

APPENDIX D

HEAT-TRANSFER MODEL

This model provides a method of calculating axial temperature and density distributions in a stagnant gas enclosed within an annulus formed by two concentric tubes. Heat is generated in the gas and in the tube walls. The gas and tubes are cooled by water flowing inside the inner tube and outside the outer tube. One-dimensional (radial) steady-state heat transfer is assumed for calculating a radially averaged gas temperature. The axial distribution of radially averaged gas temperature is determined by successive radial heat-transfer calculations at different axial positions. The axial gas density distribution, for any gas pressure, is calculated with the use of the perfect gas relation and the gas temperature distribution.

The model also possesses the following characteristics:

(1) The volumetric heating rate q_g and thermal conductivity k_g of the helium 3 are independent of radial position but are allowed to vary axially.

(2) The volumetric heating rate q_m and thermal conductivity k_m for the inner and outer tubes are independent of radial position but are allowed to vary axially.

(3) Heat is removed from the gas by conduction only.

(4) Heat transfer in the axial direction is neglected.

The calculation procedure outlined subsequently includes calculation of the radially averaged temperature at any axial position, calculation of axial gas temperature distribution, calculation of axial gas density distribution, and calculation of total mass of gas in the annulus.

Radially Averaged Gas Temperature

The radial temperature distribution through the gas is based upon the solution of the Poisson equation in cylindrical coordinates:

$$\frac{\partial^2 T}{\partial r^2} + \frac{1}{r}\frac{\partial T}{\partial r} + \frac{q}{k} = 0 \qquad \text{(D1)}$$

$$T = -\frac{q}{k}\frac{r^2}{4} + C_1 \ln r + C_2 \qquad \text{(D2)}$$

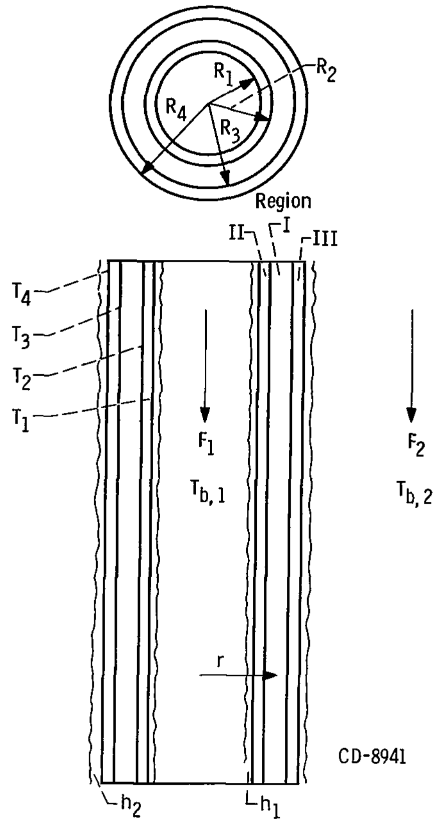


Figure 37. - Heat-transfer model of control element.

The boundary conditions in the gas, which is defined as region I in figure 37, are

$$\left.\begin{aligned} T &= T_2 \quad \text{at} \quad r = R_2 \\ \frac{\partial T}{\partial r} &= 0 \quad \text{at} \quad r = R_M \end{aligned}\right\} \text{for Region I}$$

where R_M is the radius of maximum temperature.

$$T = T_2 + \frac{q_g}{4k_g}\left[R_2^2 - r^2 + 2R_M^2 \ln\left(\frac{r}{R_2}\right)\right] \tag{D3}$$

The average gas temperature in the annulus is

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