FLUX CONTROL
FOR
IRRADIATION EXPERIMENTS

AEC Research and Development Report

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FLUX CONTROL
FOR
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By
W.K. McCARTY
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ABSTRACT

This report reviews various means which have been used to control experimental conditions in irradiation experiments, and presents the concept of local flux control.

The concept presented involves an irradiation facility, consisting of an annular can or circle of small tubes containing boron trifluoride, and a test element which is inserted within the facility. The density of the BF$_3$ may be varied, during operation, to control the flux at any single given point in the test element. The reactor coolant flows through the volume enclosed by the facility and around the outside of the facility, to cool it and the test element independently. The purpose of this concept is to permit irradiation at controlled flux, and hence at controlled burnup rate and temperature conditions.

This system removes cycle-to-cycle flux variation; and, while it cannot stop the axial movement of the peak flux in the irradiation position, it can maintain a constant flux at one point or a constant average flux, based upon the average output of a finite number of thermocouples. Based upon experimental data obtained from two experiments in the Engineering Test Reactor (ETR), advanced concepts are proposed which would permit local flux flattening or shaping, in addition to control of peak flux, to provide desired experimental conditions.
I. INTRODUCTION

A. PROBLEM

In the development of fuel elements and other components for high-temperature reactors, many tests and measurements can be made under closely controlled conditions in the laboratory. Extensive data have been taken, utilizing furnaces for prolonged time-at-temperature studies and programmed thermal cycling tests. Many other tests have been conducted in recirculating loops containing liquid metal or organics. These tests, however, have been conducted in the absence of a nuclear environment, so that the effects of fission and radiation damage are not a part of the tests. Until recently, it has not been possible to achieve a similar degree of control of experimental conditions in a reactor environment.

In the Sodium Reactor Experiment (SRE) and in the Organic Moderated Reactor Experiment (OMRE), it has been possible to perform experiments in a nuclear environment at high temperatures, but the available neutron flux level in these reactors has been insufficient for most irradiation objectives. Besides, system tests of the reactors have caused them to be of limited value for irradiation tests of materials. Hence, it has not been possible, in these reactors, to obtain experimental conditions which would be typical of those prevailing in advanced high-temperature reactors for extended periods of time. It has been necessary, therefore, to conduct experiments in the various testing reactors, all of which are water cooled.

The performance of tests of high-temperature reactor components and materials in a water-cooled reactor presents the immediate problem of surface temperature. To evaluate this problem realistically, one must first examine the nature of the high-temperature reactor itself; and, for this purpose, let us consider the sodium-cooled reactor.

One of the primary advantages of the sodium-cooled reactor concept is the combination of the relatively high boiling point (1616°F at 1 atm pressure) and heat capacity (~ 0.3 Btu/lb·°F) of the coolant. This makes possible the use of fuel elements with surface heat fluxes so high (up to 3 x 10^6 Btu/hr·ft^2) that they would cause boiling burnout in water-cooled reactors. We customarily
speak of fuel elements for sodium-cooled thermal reactors as having a peak power output of 20 to 45 kw/rod foot; although, in fast reactors, it could be higher.

Since these conditions are greater than normal for water-cooled test reactors, careful planning is required to permit testing of simulated fuel elements or fuel element sections at full power in these reactors. The surface heat flux must be reduced to provide an acceptable burnout safety factor, and the fuel temperature must be maintained at that of the high-temperature reactor. Both these conditions can usually be met by providing a thermal barrier which simultaneously increases the temperature difference between the water and the fuel and increases the surface area exposed to the water. Because of the high heat flux, the thermal barrier material must have a relatively high thermal conductivity; NaK is usually used for this purpose. Figure 1 shows a cross section of a typical high-heat-flux, high-temperature fuel irradiation capsule.

If the neutron flux is known, a test element using a thermal barrier can be designed to operate in a water-cooled reactor at the right temperature and power for the high-temperature reactor. Some compromise may be necessary

![Figure 1. Typical Capsule for High-Temperature, High-Heat-Flux Experiment](https://example.com/f1.png)
on rod diameter and enrichment, but the operating conditions will be correct. Unfortunately, the neutron flux is not known accurately enough, and it changes from cycle to cycle and during a cycle. The thermal barrier raises the temperature of the fuel in proportion to the heat flux, which ultimately means, of course, in proportion to the neutron flux. This means that, if the thermal barrier is to provide a differential temperature of 1000°F, which is typical, a 20% error in flux means a 200°F error in fuel surface temperature, and an even larger error in fuel central temperature.

Unfortunately, test reactor operators usually cannot estimate the neutron flux closer than ±20%. This can be improved by monitoring the flux just prior to insertion of the experiment; but the experiment is still subject to flux changes, due to burnup of the driver fuel elements and the resultant control rod motion, and due to the addition or removal of experiments in neighboring test positions. A change of flux of 50% is not unusual; and, in one instance in the Materials Testing Reactor (MTR), we experienced a flux increase of a factor of four, from one cycle to the next (the fluxes referred to here are all full power fluxes).

It has become clear that, with the variations in flux which prevail in all of the test reactors, some means of control is necessary for high-temperature reactor fuel experiments. Without controls, the burnup rate (or power) and temperatures vary over wide limits, and the phenomena observed in postirradiation examinations cannot be related directly to experimental conditions.

B. EXPERIMENTAL CONTROL

There are four parameters which can be varied, in order to control the temperature in a test specimen. These are: (1) the environmental temperature, (2) the heat transfer from the specimen to the environment, (3) supplementary heat in the test assembly, and (4) power generation within the test specimen. In-pile experiments have been conducted by Atomics International, utilizing each of these modes of control.

In Reference 1, Harrington presented a review of the state-of-the-art of in-pile experiment design, as it stood in the summer of 1961. Some of the information which he presented is repeated here, in order to show the need for the experimental techniques presented herein.
1. Environmental Temperature

If the parameter we wish to control is the surface temperature of the test specimen, one of the simplest means is to control the temperature of the reactor coolant. This is actually done in the test reactor, but the control is based upon some limiting condition in the reactor itself (e.g., the driver fuel element surface temperature), rather than in an experiment. A multitude of in-pile pressurized-water loops have been used to study components for pressurized-water reactors. These loops provide a capability for raising the coolant temperature about 400° F, and have the advantage of using the same coolant as the reactor. The temperature of the coolant can be controlled by heating or cooling the portion of the loop which is outside the reactor. During shutdown, these loops may be opened underwater, to insert or remove specimens.

Atomics International has built and operated two organic loops in the MTR; but, because of the expense and compatibility problems involved, has never chosen to install a sodium loop in a water-cooled reactor. Others have operated small NaK loops in-pile, but it was necessary to replace the entire loop to change specimens.

2. Heat Transfer

In applying the heat transfer mode of temperature control, most experiments have utilized a gas annulus, in which the composition of a flowing gas has been adjusted to vary the effective thermal conductivity. Such designs are described in Reference 1, as is a device developed by the author, utilizing a modified gas annulus having movable fins, which can change the effective annulus thickness. These heat transfer control techniques are only applicable to low surface-heat-flux experiments, up to a practical limit of perhaps 20 kw/rod foot. Other heat transfer control schemes may be developed which will lend themselves to higher heat fluxes.

3. Supplementary Heat

A third means of control is the addition of supplementary heat to the assembly. This is done by surrounding the specimen with an electrical heater. This technique works reasonably well for low- or intermediate-heat-flux experiments, although the heater materials which are available leave something to be
desired. For high-heat-flux experiments, however, the technique is severely limited. Let me cite an example:

Suppose we are to test a fuel rod at 40 kw/ft, with the fuel surface and fuel center at 1000 and 2000°F above the coolant temperature, respectively, in a flux with an uncertainty of ±20%. To be able to control the temperature, the rod must be designed to operate at temperature at 120% of nominal flux with no electrical heat. Simultaneously, the design must permit the same surface temperature at 80% of nominal flux and full electrical heat. Thus, it can be shown that the heat transfer must be planned for 48 kw/rod foot; and that, since nominal flux generates 40 kw/rod foot, the electrical heater must be capable of generating the equivalent of 40% flux, or 16 kw/rod foot, to maintain the surface temperature when the flux is 80% of nominal and the nuclear power is only 32 kw/rod foot. Furthermore, this heater must be packed into the NaK annulus in such a way that it will permit adequate heat transfer in the high flux case. Heaters which will provide an output of this magnitude and occupy such a small volume are not currently available. Also, perfect performance of the heater in controlling the surface temperature would still allow a ±200°F variation in center temperature.

4. Power Generation in Specimen

The control schemes discussed in the preceding sections have two failings in common, as applied to high-temperature, high-heat-flux experiments. First, they are limited in range of control, and cannot correct for the extent of change in neutron flux usually observed. Second, they control the temperature at only one selected point (e.g., fuel surface or fuel center). The specific power and fuel ΔT are still proportional to the incident neutron flux. If, on the other hand, the flux itself can be adjusted, both the temperature and specific power can be controlled.

It is possible to adjust the flux by varying the reactor power. This is a practical approach, where the reactor itself is the experiment. It is even feasible for the highest priority experiment in the reactor; but this procedure is likely to be deleterious to the other experiments. It is also possible to tilt the flux in the reactor (i.e., to increase the flux in one region). Since an increase in flux in one region causes a power increase in the driver fuel elements in that

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region, and since the test reactor power is normally held nearly constant, this flux increase must necessarily be accompanied by a flux decrease in other regions. Test reactor control systems are generally incapable of providing this control, for a particular experiment, without introducing effects which are deleterious to other experiments.

Atomics International has studied several techniques of local flux perturbation, to correct for flux gradients and to control the flux level.
II. LOCAL FLUX VARIATIONS

As discussed previously, our problems in controlling the temperature in a fuel irradiation originate with errors in flux prediction and changes in flux during the experiment. Another problem, not previously mentioned, is the flux gradient across the experimental position, both radially and axially. Furthermore, in the high-flux test reactors, the burnup of the fuel in the driver elements is so rapid that there is a significant axial shift of the flux peak during each cycle. This is caused by the gradual withdrawal of the control rods, to maintain a steady reactor power. This axial shift of peak flux is as large as 8 in., in one reactor.

Atoms International has designed and conducted in-pile experiments, employing both radial and axial motion, and employing neutron poisons, to obtain satisfactory fluxes in its experiments. Examples of these are discussed in the following sections.

A. VARIABLE POSITION

About seven years ago, a series of experiments, in which an axial motion was available while the reactor was at power, was designed and operated in the MTR. In this case, the fuel assembly was only 2 in. long, and could be adjusted to ride the peak of the flux curve. This was accomplished by means of an O-ring sealed screwjack at the reactor flange, and a flexible lead tube section going around a 40° bend. Eight inches of vertical travel was provided, and this resulted in a possible flux variation of a factor of two.

With the advent of the General Electric Test Reactor (GETR), a series of experiments was designed for irradiation there. AI took advantage of the flexibility offered by the reflector pool, and GE built the Radially Adjustable Facility Tube (RAFT) facilities, to accommodate this type of experiment. This facility provides radial motion within the reflector pool, and makes available a significant adjustment of flux level while maintaining water flow past the experiment. It is particularly advantageous in "sitting out" the flux peaking during xenon buildup at the beginning of the GETR cycle. This early cycle flux peaking can be a particular disadvantage to those experiments requiring a constant, or fairly constant, flux. While there were some mechanical problems with the
Figure 2. Axial Flux Plot

Figure 3. Cross Section of Experiment AI-20

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RAFT facilities, initially, they have been largely overcome; and AI is still using them with satisfactory results. This system depends for its successful operation upon the existence of a flux gradient through the reflector region where it is used. It can correct, with reasonable limits, for errors in predicting the flux and in calculating the flux perturbation caused by the assembly. However, the specimens are still subjected to a flux gradient.

GE Vallecitos has gone on to develop a second type of facility, the Vertical Adjustment Facility Tube (VAFT), which permits the following of the flux peak by an experiment, and a third, the V-RAFT, which combines both motions. The V-RAFT should make it possible to control both the location and height of the flux peak. The shape of the flux curve along the axis of the experiment would not remain quite the same, but should not deviate greatly. Since the axial flux distribution tends to become flatter as the cycle progresses, the use of this facility to maintain a constant peak flux at one position would bring about a small increase in flux toward the ends of the assembly as the driver fuel elements burn up.

B. USE OF SOLID NEUTRON POISONS

The use of neutron poisons in a region surrounding an experimental assembly, to control the flux to the experiment, has been studied extensively. Consideration has been given to both absolute flux intensity and flux distribution.

When a neutron absorber is introduced into a neutron flux, the flux is depressed. The amount of flux depression, or perturbation, is dependent upon many factors which will not be detailed here. An axial flux plot of a hypothetical case, in which a fuel capsule containing no significant absorber, except for the fuel, is inserted in a flux field, is shown in Figure 2. The shape of the flux curve over the central portion of the specimen is similar to the unperturbed flux curve. The flux peaks at the ends, however, due to the absence of absorber beyond the ends. This effect is called the "lightning rod effect." If the portion of the experimental assembly which has approximately constant macroscopic cross section (e.g., the fueled portion in a fuel experiment) is about the same length as the reactor core, the "lightning rod effect" is not usually very important, since the flux is low at the end. If this portion is short, however, or if
Figure 4. Flux Perturbation by Eccentric Depressor
there are segments with low cross section material between them, the effects of flux peaking at the ends can overshadow other effects of the experiment. To negate the "lightning rod effect," a practice has developed of placing extra pieces of fuel or poison discs (such as boral, boron steel, tungsten, etc) at the ends of the specimens. These absorber pieces do not necessarily eliminate the peak, but do push it away from the ends of the test specimens.

The steep flux gradients present in the reflector regions of high-flux test reactors are the cause of much concern. In addition to the problem of obtaining the desired peak flux intensity, the flux distribution can seriously affect the temperature distribution. As an example, consider a fuel capsule of the design shown in Figure 1, and assume that the fuel specimens are relatively long. Assume further that this capsule is placed in a steep transverse flux gradient. Since the flux is higher on one side than on the other, the power generation, and hence the temperature distribution will be eccentric. The temperature on the side of the rod toward the peak flux will exceed that of the opposite side, and the resultant thermal expansion will bow the rod. The bowing of the rod, in turn, changes the thickness of the thermal barrier, tending to lower the temperature at the thinner section and partially compensate by reducing the bowing somewhat. A significant element of doubt exists regarding the resultant geometry and temperature distribution and evaluation of the experimental results is subject to question.

In the design of an experiment series conducted in the Westinghouse Test Reactor (WTR) reflector, this problem was studied in some detail, and the resultant concept was quite successful in flattening the radial flux gradient. An eccentric stainless steel annular can filled with a copper-silver eutectic surrounded the test assembly, outside the water-cooling annulus. The poison assembly was oriented to position the thick portion on the side toward the reactor core, where the higher flux was located. A cross section of the experimental assembly is shown in Figure 3.

In this case, the unperturbed flux dropped a factor of three in 2 in. Figure 4 presents a plot of flux across the experimental position. Curve 2 shows that, without the poison, the flux on one edge of the fuel is twice that on the opposite edge, while the flux is almost flat across the fuel on the side away from the core. The location of the peak temperature must be well off center.
Curve 1 shows that the poison has gone a long way toward correcting this situation, in that the ratio of flux from one side to the other is now 1.25. If the poison annulus were blacker, this number would probably be quite close to 1. This technique has been used on a recent experiment in MTR, in which temperatures within 10°F of those desired were achieved.

C. USE OF LIQUID POISONS

Several organizations, including Atomics International, have studied the use of liquid poison systems for flux control. These systems usually involve aqueous solutions of cadmium or rare earth salts. If the liquid is flowing, the concentration can be changed, from time to time, to adjust the flux. Problems would be expected with radiolysis of water, requiring recombination or venting the hydrogen and oxygen and also solution makeup and recovery.

D. USE OF GASEOUS POISONS

The only gas which has received serious consideration at Atomics International as a neutron poison is boron trifluoride, although the use of He³ has been suggested. Diborane (B₂H₆) may be another possibility. In connection with the development of emergency reactor scram systems, a great deal of information has been accumulated at Atomics International on the physical and chemical properties of boron trifluoride. Smith⁶ extended the work on PVT relationships and McCarty⁷ conducted irradiations to study radiolysis and corrosion problems with BF₃.
III. BORON TRIFLUORIDE FOR FLUX CONTROL

A. BACKGROUND INFORMATION

In the fall of 1959, a study was undertaken of the use of BF$_3$ for variable flux control. Since boron can be obtained in the form of a gas, as boron trifluoride, it offered the possibility of changing the macroscopic cross section of an assembly containing BF$_3$, by changing the pressure and thereby changing the atom density of boron-10. The concept originally studied involved the use of an annular can, made of stainless steel, to contain the BF$_3$. Variation of the gas density would adjust the flux within the can. The annular can would be an irradiation facility, within which a model fuel element could be inserted and operated at a controlled power despite variations in incident neutron flux. While a higher incident flux is required with such a facility than for a comparable test without it, this concept offers the possibility of operating the fuel element at full power consistently and reaching the objective burnup in a shorter time.

The behavior of BF$_3$ in a neutron flux field is discussed in detail in Appendix A, and a discussion of the problems involved with the use of a boron trifluoride system is presented in Appendix D.

B. EXPERIMENT NAA-62

A facility was designed which would replace a 3-in. square "A-piece" in the ETR, and which would accommodate a 3-rod model fuel element. Each of the three rods was of the basic design shown in Figure 1, and contained 15% enriched uranium carbide fuel slugs of 0.400 in. diameter. The outside diameter of the fuel rods was 0.886 in., and they were equally spaced on a 1.090-in diameter circle. A BF$_3$ annulus, 0.171 in. thick, formed by an annular stainless steel can, surrounded the fuel element as shown in Figure 5. Aluminum spacers were provided to channel the reactor coolant along the fuel rods. The reactor coolant flowed through the volume enclosed by the annular can and around the outside of the can, to cool the annulus and the test element independently.

The first experiment of this series was installed in Position P-10 of the ETR for two cycles, starting in June 1961. The BF$_3$ density was controlled by
a valve manifold, using a cylinder of BF$_3$ as the source, and using a potassium hydroxide scrub system to react the used BF$_3$. It was intended that the density would be controlled automatically by solenoid valves, using the output of a thermocouple attached to the cladding of one fuel rod for a control signal, but internal leaks prevented their use. The leaks were thought to be caused by failure of the solenoid valves to seat properly. Subsequent examination showed the cause of the difficulty to be corrosion of the valves by HF formed by hydrolysis of the BF$_3$. The source of moisture was assumed to be the helium used for purging the system. The experiment was operated quite satisfactorily manually, using micrometer valves as seal valves.

During the first few hours of the irradiation, the BF$_3$ pressure was intentionally varied over a wide range to accumulate data on the effectiveness of the system in controlling flux. These data are presented in graph form in Figure 6.
Figure 6. Experimental Data, NAA-62-1
Following the failure of the original BF$_3$ density control system to function automatically, a different approach was taken for the second experiment. A system was developed which controlled the vapor pressure over a refrigerated reservoir containing liquid BF$_3$. It was a closed system which, during normal operation, had no moving parts. It provided rapid response to increases in incident flux, and was fail-safe in operation. The system is described in detail in Appendix C.

The second experiment, NAA-62-2, was operated in the ETR for five reactor cycles in the fall and winter of 1963-64. It was expected to operate with a constant burnup rate and a peak central temperature at the control point of $1600 \pm 25^\circ F$ at all times when the reactor power was in excess of 125 Mw. However, the reactor flux turned out to be only about half that desired, and the peak central temperature was only $1500^\circ F$ at full power with the BF$_3$ condensed. To demonstrate the control capability, the setpoint was lowered, and the test ran automatically with the central temperature at $1205 \pm 20^\circ F$ for a period of 10 days. The setpoint was then returned to its original value.

The degree of temperature control achieved in these experiments was gratifying. Two examples of the effect of BF$_3$ pressure change on central fuel temperature are given in Table 1 for each experiment.

### Table 1

**FLUX REDUCTION BY B$_{10}$F$_3$**

<table>
<thead>
<tr>
<th>Reactor Power (Mw)</th>
<th>Pressure (psia)</th>
<th>Central Fuel Temperature ($^\circ F$)</th>
<th>Flux Reduction (%)</th>
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<tr>
<td><strong>NAA-62-1</strong></td>
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<tr>
<td>120</td>
<td>12</td>
<td>1940</td>
<td>42</td>
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<tr>
<td></td>
<td>368</td>
<td>1180</td>
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<td>150</td>
<td>12</td>
<td>2200</td>
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</tr>
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<td></td>
<td>395</td>
<td>1590</td>
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<td><strong>NAA-62-2</strong></td>
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<tr>
<td>130</td>
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<td>1170</td>
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<td></td>
<td>715</td>
<td>795</td>
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C. EVOLVED CONCEPTS

There are several features of the BF$_3$ flux control system which can be improved in future experiments. Our calculations show that a significant improvement in control capability can be obtained by reducing the stainless steel and U$^{235}$ content. The flux perturbation factor is a function of the macroscopic cross section of the experimental assembly. Thus, the larger the proportion of the cross section contributed by the boron, the larger will be the effect of pressure variations in the BF$_3$.

1. Improvement in Neutron Economy

A recent study involved a flux-controlled system, for use with a long fuel cluster. The program has since been terminated, but not until detailed analyses had been conducted. In this case, one of the ground rules was that $K_{\text{eff}}$ must be between 1.0 and 1.2. The basic design of NAA-62, even without BF$_3$, had such a high macroscopic cross section that $K_{\text{eff}}$ was around 0.3 or 0.4. The design was modified to reduce the cross section. The most significant change was the replacement of the heavy-walled stainless steel annular can by a circle of 1/4-in. diameter by 10-mil wall stainless steel tubes; the resultant cross section reduction made it necessary to reduce the fuel enrichment as well. The desired value of $K_{\text{eff}}$ was achieved by this means, and a corollary dividend as well: the amount of boron-10 necessary to drop the flux 50% was reduced from $2 \times 10^{22}$ to $7.8 \times 10^{21}$ atoms per linear inch.

In this instance, a relatively high burnup was desired using 5% enriched uranium carbide. A relatively high depletion of U$^{235}$ would be expected; and hence, at constant flux, the power would drop off. The plot in Figure 7 shows the flux required in the fuel to maintain power as a function of burnup, and the flux available at two levels of boron density. This demonstrates how this system can be used to maintain constant power output in an experiment.

2. Axial and Transverse Flux Variation

Up to this point in the discussion of BF$_3$ flux control, we have not considered the axial flux distribution. If, in the case of the annular can of NAA-62, the boron density were uniform throughout, the shape of the axial flux curve would remain the same but the intensity would change as the boron density was changed. The boron density is not uniform, however, because the flux is
not uniform. In the regions of higher flux, the fission rate of boron-10 is higher; this results in a higher temperature and a lower boron density since the pressure must be uniform. This tends to accentuate the axial flux curve, increasing the peak-to-average flux ratio. The extent of this change would be a function of the absolute flux level, becoming more pronounced at higher flux levels. Similarly, the radial flux gradient would tend to be accentuated somewhat.

We have given some thought to shaping the flux curve to suit our purposes. If we were to locate more poison near the peak of the flux curve, we could reduce the peak-to-average ratio or even invert the curve, if we wished. If we were to locate more poison on the side of the assembly toward the reactor core, as we did with Ag-Cu alloy in an earlier example, we could reduce the radial flux gradient or even invert it. It then would follow that a barrel-shaped annular can (having a thicker annulus at the midpoint than at the ends) containing BF$_3$ should provide axial flattening, and that an eccentric can should provide radial flattening. However, it is not quite this simple.
The $BF_3$ gas itself is a heat source. The temperature is not only dependent upon the boron-10 fission rate, but also upon the geometry (i.e., upon the heat transfer). Calculations show that, other things being equal, a thicker annulus would have a higher peak temperature and a higher average temperature than a thinner one. If the thick annulus communicates with the thin annulus, some of the mass moves from the thick annulus to the thin one. Thus, the pressure equalizes but the density disproportionates. An eccentric annular can, placed in a field of uniform flux, would have a higher average temperature and lower average $BF_3$ density on the thick side than on the thin side, but would still have a higher mass of $BF_3$ per unit sector on the thick side. It then can be shown that a barrel-shaped annulus and an eccentric annulus will bring about the flux shaping expected; but, since the gas is free to move, the extent of change is not as great as would otherwise be expected.

The effect just discussed is simply the effect of thermal expansion of the gas. If, instead of the annulus, we proceed to the concept of a ring of tubes which makes more efficient use of the flux, we can shape the flux by having more or fewer tubes on one side than the other, rather than an eccentric annulus. Here, the heat transfer path is the same in each tube; and the temperature effect, while still present, is minimized. To flatten the axial flux curve, the tubes could be tapered, or fewer tubes could be used at the ends.
IV. CONCLUSIONS

The technique of controlling thermal neutron flux, using boron trifluoride as described here, has proved to be a very satisfactory means of achieving and maintaining the desired power and temperature distribution in an in-pile experiment. It is possible, using this technique, to achieve a desired exposure or burnup in a shorter time due to the ability to maintain full flux or power levels in the test element. The range of flux control observed in an experiment in the Advanced Sodium Graphite Reactor (ASGR) critical facility and two experiments in the ETR, and the ease with which the flux to the experiment was changed, demonstrate the value and flexibility of this concept. The refrigerated liquid BF$_3$ reservoir system has proved very satisfactory in adjusting the BF$_3$ density. However, it is necessary to purify the gas at frequent intervals, in order to retain the ability to condense the BF$_3$.

The range of flux control available is limited only by the available incident flux and the pressure capability of the facility. If the macroscopic cross section without BF$_3$ is minimized, the control range is maximized. Advanced concepts, using this system, will permit tailoring the flux to the requirements, flattening or shaping the flux, and holding a constant flux, power output, or temperature. Step changes or gradual changes in flux level could be introduced as desired.
APPENDIX A
BEHAVIOR OF BF3 IN A NEUTRON FLUX FIELD

Under irradiation, BF3 decomposed by two paths: the (n, α) reaction of boron-10, and radiolysis.

The (n, α) reaction,

\[ B^{10} + n \rightarrow α + Li^{7} \quad \ldots (1) \]

results in the net reaction,

\[ B^{10}F_3 + n \rightarrow Li^{7}F + He + F_2 \quad \ldots (2) \]

and the results of radiolysis are

\[ 2 \text{BF}_3 \rightarrow 2 \text{B} + 3 \text{F}_2 \quad \ldots (3) \]

It should be noted that Reactions 2 and 3 both produce more gaseous molecules than were present initially, resulting in a pressure increase in a closed system. None of the reaction products are radioactive, except for the short-lived α particles. Except for the elemental boron-10 formed in Reaction 3, the products all have low neutron capture cross sections, and further nuclear reactions are negligible. The elemental boron and lithium fluoride are solids; part of this material adheres to the walls of the vessel and the balance falls to the bottom. The material on the walls appears to blister and flake off. This problem can be minimized by using BF3 which is highly enriched in boron-10. While normal boron has a 2200 m/sec cross section of 755 barns, the cross section of boron-10 is 4017 barns. For this reason, we have used 90% enriched B\(^{10}\)F\(_3\) rather than normal BF\(_3\). This has resulted in a reduction of the required pressure by a factor of five and of the B\(^{11}\)F\(_3\) content by a factor of 43, for the same cross section.

BF\(_3\) departs significantly from the perfect gas law, PV = n RT; this is the reason for the work reported in Reference 6. The (n, α) reaction of boron-10
<table>
<thead>
<tr>
<th></th>
<th>Relative Volume</th>
<th>Scram</th>
<th>175 Mw</th>
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<tbody>
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<tr>
<td></td>
<td>Temperature (°F)</td>
<td>Density (g/cc)</td>
<td>Temperature (°F)</td>
</tr>
<tr>
<td>Case No. 1</td>
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<tr>
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<td>0.053</td>
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<td>70</td>
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</tr>
<tr>
<td></td>
<td>4.8</td>
<td>120</td>
<td>0.050</td>
</tr>
</tbody>
</table>

Reduction in BF$_3$ density = 27.8%

Case No. 2

Pressure at 175 Mw = 363 psi
Pressure scrammed = 270 psi

Reduction in BF$_3$ density = 24.4%

*Temperatures estimated to be between 500 and 650°F
†By differences
releases approximately 3 Mev of energy to the system; this brings about an increase in temperature and pressure in the gas as the neutron flux increases.

The gas annulus of NAA-62-1 communicated with the volume in the two tubes connecting it to the gas console. When the reactor flux increased, the increased temperature and pressure forced some gas out of the annulus into the connecting tubing. Data taken at two scrams are given in Table 2 to demonstrate this effect.

It is further estimated that, were the gas not permitted to expand from the annulus into the lead tube upon subsequent startup, the pressure in the annulus would have increased from 250 to 460 psi in Case No. 1, and from 270 to 495 psi in Case No. 2.

To help in understanding the performance of tubes and annuli containing BF$_3$ in a flux field, a series of calculations were performed. This is a very complex problem, and a number of simplifying assumptions were made. The two most significant assumptions were: (1) that the thermal conductivity of BF$_3$ is the same as that of CO$_2$, and (2) that the flux is the same across the BF$_3$ chamber. The first assumption was made in the absence of data for BF$_3$, but after comparing data for 12 other gases. The second assumption is obviously in error, but is not considered unreasonable for the qualitative conclusions drawn.

Data, calculated for the temperature and density as a function of diameter in a tube with 0.230 in. ID, are shown in Figure 8. Note that the effects of flux and pressure are both shown. The shapes of the density curves show that the bulk of the gas is near the wall with a significantly lower density in the center. This would lead one to wonder whether a large diameter tube would offer much more effective volume. To answer this question, a similar calculation was made for a tube with 0.430 in. ID. Density Curve B (Figure 8) is replotted, together with the new one, in Figure 9. In this case, to make the comparison easier, the percent of mass is plotted vs percent of volume, measuring from the outside in; and, for all practical purposes, the curves coincide. The total mass in the larger tube is more than in the smaller tube, but the density is less due to the higher temperature. However, the mass distribution in the tube is apparently the same.
Figure 8. Temperature and Density of BF$_3$ in 0.230-in. Diameter Tube Under Irradiation

Figure 9. Mass Distribution of BF$_3$ (200 psi at 1 x 10$^{14}$ n/cm$^2$-sec)
APPENDIX B
NUCLEAR CALCULATIONS

Using the attenuation equation for thermal neutron absorption in an infinite slab,

\[ I = I_0 \exp(-\Sigma_a X) \quad \ldots (4) \]

where:

- \( I_0 \) = initial intensity
- \( I \) = final intensity
- \( \Sigma_a \) = macroscopic absorption cross section (cm\(^{-1}\))
- \( X \) = annulus thickness (cm)

approximate calculations were made to determine the possible effectiveness of BF\(_3\), 90% enriched in B\(^{10}\), for flux control. \( I/I_0 \) is the fraction of neutrons not absorbed; therefore, \( 1 - I/I_0 \) is the attenuation, the fraction of neutrons absorbed. The macroscopic absorption cross section for thermal neutrons is proportional to the BF\(_3\) density (i.e., a function of temperature and pressure). Plots were made of neutron attenuation vs gas pressure. This information is shown in Figure 10. The attenuation calculated across the annulus was 70% at a maximum pressure of 475 psi at a gas temperature of 80°F. The gas temperature varies from place to place through the system due to power generation within the BF\(_3\), with a resultant density variation as discussed in Appendix A. In the high-flux region of the reactor, the gas temperature is known to be high and the density correspondingly lower, but the values are not known. In the absence of thermal conductivity data, the temperature and density cannot be calculated accurately. For this reason, the pressure is used in Figures 6 and 10, although it is recognized that the pressure does not adequately describe the boron-10 density.

Machine calculations (I\(_2\) code) were performed on the test assembly, by converting the assembly into a model having cylindrical cell geometry with different annular regions. The area of the annular regions corresponded to the

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Figure 10. Thermal Neutron Attenuation Across the BF\textsubscript{3} Annulus

Figure 11. Flux Perturbation in NAA-62
area fractions of each material in the actual assembly. The $I_2$ machine code solves the neutron transport equation to a $P_3$ approximation, by the Weil method, for the thermal neutron flux at mesh points through each region. By comparing the neutron flux value at the inner boundary to that of the outer boundary, the neutron attenuation across the annulus was determined as a function of BF$_3$ pressure. Figure 10 presents $I_2$ machine code data for gas pressures up to 438 psia, indicating an attenuation of 52% at a pressure of 438 psia. Figure 11 presents flux plots across the NAA-62 experiment assembly at two different BF$_3$ densities.

The neutron flux perturbation factor is the neutron attenuation factor from the outer boundary of the cylindrical cell model previously described to the center of the fuel in the cylindrical cell model. $S_n$ machine code calculations were used to determine the amount of neutron flux perturbation caused by the test assembly. The $S_n$ machine code gives a fourth approximation to the neutron transport equation, and is generally considered to give better results than the older $I_2$ code. The perturbation factor was calculated for BF$_3$ pressures of 1.47, 14.7, and 331 psia in the annulus, and the attenuation predicted for 331 psi was 58%. This result is also plotted on Figure 10, for comparison with the other data.

A nuclear mockup of this system was irradiated in the Sodium Graphite Reactor Critical Facility at the Atomics International Field Laboratories, Santa Susana, California. In order to experimentally determine the magnitude of thermal neutron attenuation which could be obtained across the 0.171-in. thick BF$_3$ annulus, the mockup was fueled with Th - 7.6 wt% U, 4.8 wt% enriched with U-235. The mockup was physically identical to the actual design, except for the difference in fuel enrichment and the fact that the coolant annuli around the fuel rods in the mockup were filled with aluminum. Natural BF$_3$ gas (18.8 at.% B$^{10}$) was used. Cadmium foils were inserted in special channels on each side of the annulus to determine the integrated thermal neutron flux for each datum point. The data were taken in the following manner: the annulus was pressurized, the reactor operated long enough to activate the Cd foils, the foils removed for counting, and the annulus then pressurized to a new value for another datum run. At the maximum test pressure of 575 psia, the neutron attenuation across the annulus was 27.5%. The 575 psia of natural BF$_3$ gas corresponds to 125 psi of 90% enriched gas at the operating temperature of 80°F.
The experimental data are plotted in Figure 10, for comparison with the various calculated data.

The amount of boron which is required to give a specific flux reduction is dependent upon the perturbation caused by the experimental assembly itself. In the NAA-62 experiments, it was determined that a boron-10 content of about $2 \times 10^{22}$ atoms per linear inch was required to reduce the flux by 50%. Calculations have been made for a similar 3-rod test, in which the neutron economy has been optimized by the removal of a significant amount of stainless steel and nickel and by a reduction in fuel enrichment. This resulted in the ability to reduce the flux by 50% with a boron-10 content of about $7.8 \times 10^{21}$ atoms per linear inch with a similar power output of the fuel.

The margin of control which is considered necessary is dependent upon the degree of confidence one has in the predicted flux and flux variations, and in the flux perturbation calculations. In using this control technique, one must ask for a higher flux than he really desires, in order to gain a margin of control. Several advantages accrue from making this margin narrow: (1) the neutron cost will be less, (2) there will be less effect on neighboring experiments, and (3) there will be less BF$_3$ required and hence fewer decomposition products to dispose of. Against this must be balanced the likelihood of underestimating the required flux and running cold, despite the flux control; one can compensate for a flux which is too high by increasing the BF$_3$ density.
APPENDIX C
REFRIGERATED BF₃ HANDLING SYSTEM

BF₃ is a condensable gas which will freeze in liquid nitrogen. Its vapor pressure curve is given in Figure 12. These properties make it easy to handle in a vacuum system, and easy to transfer from one vessel to another. We decided to make use of these properties to control the gas pressure. Since internal leaks in valves caused the automatic gas control system for experiment NAA-62-1 to malfunction, we resolved to eliminate as many valves in the operating system as possible, and were quite successful in doing so. The resultant system has no valves which must be operated during reactor operation, except for one special circumstance which will be discussed later. The effective system is shown in Figure 13.

The minimum temperature limit of the reservoir was arbitrarily set at -130°F. Initial laboratory experiments with this system utilized thermoelectric cooling to cool the reservoir, but it failed to reach that temperature. We then obtained a two-stage cascade refrigerator, with approximately a 200-w capacity, for this purpose. The temperature achieved by a refrigerator is predetermined by the design of the system and the selection of the refrigerant; it is not as easily adjustable as an electrical heater. For this reason, it was found necessary to oppose the refrigerator by electrical heaters, as discussed in the following paragraphs.

The BF₃ reservoir is shown schematically in Figure 14. It consists of 20 ft of Type 304 stainless steel tubing, 1/4-in. in diameter with a 0.028 in. wall, formed into a helix approximately 12 in. in diameter and 16 in. long, and with a return leg from the bottom to the top. The refrigeration coil and electrical heaters were wound with the reservoir coil and bonded to it with a thermal mastic. The return leg is neither heated nor cooled. Three thermocouples are attached to the refrigeration coil, two to the BF₃ coil and one to the return leg of the BF₃ coil. The entire assembly is cast in plastic foam to insulate it from its environment and to exclude moisture which could otherwise condense on the coil.

The refrigerant enters at the top of the coil and has its maximum effect in that region. To smooth out this effect, the power from the heaters is distributed more heavily at the top of the coil than at the bottom. As shown in Figure 15, the auxiliary heater is the same as the control heater, but only a
Figure 12. Vapor Pressure of BF$_3$
VALVE 2 (NORMALLY CLOSED)  VALVE 1 (NORMALLY OPEN)

HEATER  LIQUID BF₃  REFRIGERATOR  REACTOR

PRESSURE GAUGE
 struct: P
 PRESSURE TRANSUDER
 struct: T

Figure 14. Diagram of BF₃ Reservoir

BF₃ CONDENSES HERE
RETURN LEG 16 in.
BF₃ EVAPORATES HERE
12 in.

0-1000 psi PRESSURE GAUGE
0-3000 psi PRESSURE TRANSUDER

Figure 13. BF₃ Density Control System

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Figure 15. Wiring Diagram for Heaters

RESISTANCE VALUES

A 14.4 Ω, 0.72 Ω/ft
B 12 Ω, 1 Ω/ft
C 5 Ω, 1 Ω/ft

DOUBLE LINES REPRESENT RESISTANCE HEATERS
SINGLE LINES REPRESENT CONDUCTOR WIRES
WIRES 28, 31, AND 33 ARE NOT CONNECTED TO TERMINALS, BUT ARE INSULATED
portion of it is used. It operates at a higher power, because a higher voltage is applied. In the event of a failure of the control heater, the auxiliary heater may be used in its place. There are actually two emergency heaters. If one should fail, the other could be used in its place; or, if either the control heater or auxiliary heater should fail, one of the emergency heaters could be used for the auxiliary heater. The heat produced per foot of coil is shown in Table 3, for four regions of the reservoir coil.

**TABLE 3**

<table>
<thead>
<tr>
<th>Heaters on</th>
<th>Coil Length from Bottom (ft)</th>
<th>Power (w/ft)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>0 - 10</td>
<td>10 - 15</td>
</tr>
<tr>
<td>C</td>
<td>0-3</td>
<td>0-14</td>
</tr>
<tr>
<td>C + A</td>
<td>28</td>
<td>40</td>
</tr>
<tr>
<td>C + A + E</td>
<td>78</td>
<td>90</td>
</tr>
</tbody>
</table>

C = Control Heater  
A = Auxiliary Heater  
E = Emergency Heater

The leads to these heaters all terminate in a terminal box at the end of the refrigerator enclosure. The substitution of one heater for another may be made by changing the connections on terminal strips in this terminal box. The wires shown as insulated on Figure 15 terminate in the box also, but are simply unattached and insulated.

The control system consists of the refrigerator coil and electrical heater in the BF$_3$ reservoir and the associated instrumentation.

The refrigerator coil operates at capacity at all times during normal operation, and is opposed by the control heater operating at some intermediate power. The helical coil is chilled by the evaporation of the refrigerant, Freon 13, from the top down to the point where all the Freon has evaporated. The length of the coil which is thus cooled is determined by the power input to the control heater; with no power input, the entire coil will reach equilibrium at about -130°F.
A dynamic equilibrium exists in the reservoir coil, with BF$_3$ condensing in the upper portion of the coil, running down and evaporating in the lower portion of the coil, with the vapor going up the return leg and recondensing in the top of the coil. The BF$_3$ itself acts as a refrigerant in the lower part of the coil, opposing the control heater in that region. Whenever the power input to the control heater is adjusted by the control system, a new equilibrium BF$_3$ pressure is established in the annulus. Figure 14 shows this part of the system schematically. Figure 16 is a photo of the coil.

![Figure 16. BF$_3$ Reservoir](image)

The parameter which is actually controlled is the temperature at the measuring junction of a thermocouple in the NaK annulus surrounding one fuel rod. The signal from this thermocouple goes to a proportional control system which adjusts the control heater power between 0 and 215 w.

In the event that the temperature exceeds the control setpoint by 10° F, an auxiliary heater will be turned on automatically giving a step increase of 700 w. Should the temperature exceed the control setpoint by 30° F, an emergency heater will automatically apply an additional 1000 w. These "back-up" heaters are controlled by cam-operated control switches located in the fuel temperature recorder/controller. For maximum effect at this emergency control point, the application of power to the emergency heater can be accompanied by shutdown of the second stage of the refrigeration unit, if desired, thus providing for maximum BF$_3$ density in the annulus.
The required temperature range in the BF$_3$ reservoir is from -135 to +15°F. At -135°F, the BF$_3$ pressure is about 40 psia; and, at +15°F, all of the liquid BF$_3$ has evaporated, since the initial pressure loading is the equilibrium pressure at +15°F. Further heating has little effect on the pressure. As a precautionary measure, a thermostat was attached to the BF$_3$ reservoir to limit its peak temperature. This control turns off all heaters if the coil temperature exceeds 200°F, and will protect the system from overheating in the event of an open circuit of the controlling thermocouple.

The operation of the temperature control system and the interrelationship of its various components can be best understood by referring to the block diagram in Figure 17. The recorder/controller responds to the output of the fuel cladding thermocouple. The fuel temperature is recorded on a 0 to 1200°F scale. A retransmitting slidewire within the recorder produces a dc voltage which is proportional in amplitude to the amount of temperature deviation from the control setpoint. This voltage is zero at the control point temperature. Its polarity is determined by the direction of deviation. The voltage output from the retransmitting slidewire is used by the proportioning unit to control the power provided to the control heater by the power manipulator. This provides anticipatory control which minimizes overshooting the control temperature.

![Diagram of Control System](image)

Figure 17. Diagram of Control System
Appendix A refers to the buildup of helium and fluorine as a result of the $(n, \alpha)$ reaction and radiolytic decomposition. Neither of these gases is condensable in a temperature range which is desirable to control the $\text{BF}_3$ pressure. They tend to concentrate in the reservoir and, in time, would seriously hamper the pressure control system. The net effect is to increase the $\text{BF}_3$ content of the annulus, which is safe from a safety standpoint, but prevents the attainment of desired experimental conditions. It therefore is necessary to remove the helium and fluorine, from time to time.

The removal of helium and fluorine is not difficult using liquid nitrogen. The melting points and boiling points at atmospheric pressure of these gases are shown in Table 4.

<table>
<thead>
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<th>Gas</th>
<th>MP $^\circ$C</th>
<th>BP $^\circ$C</th>
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<tbody>
<tr>
<td>Helium</td>
<td>-272.2</td>
<td>-268.9</td>
</tr>
<tr>
<td>Fluorine</td>
<td>-223</td>
<td>-187</td>
</tr>
<tr>
<td>Nitrogen</td>
<td>-209.86</td>
<td>-195.8</td>
</tr>
<tr>
<td>$\text{BF}_3$</td>
<td>-127</td>
<td>-101</td>
</tr>
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</table>

It readily can be seen that at liquid nitrogen temperature, the $\text{BF}_3$ is frozen solid, the fluorine is liquid but with considerable vapor pressure, and the helium is still gaseous. The helium and fluorine can be pumped off, and the $\text{BF}_3$ will remain in a liquid nitrogen cooled vessel. Rather than pump fluorine through a vacuum pump, however, we chose to use a vessel filled with mossy tin and heated to $200^\circ$C to react the fluorine.

A gas handling system, incorporating all these features, was built and used for Experiment NAA-62-2. It is shown in Figure 18, and was found to be quite satisfactory. In addition to the facilities described previously, a small condensing bulb was added between Valves No. 2, 3, and 4; its purpose is to assist in recovering all the $\text{BF}_3$ from the purification part of the system since the reservoir can only reduce the pressure to a minimum of about 25 to 40 psia.
Figure 18. Gas System Flow Diagram
A statement appears earlier that, with one exception, no valves need be operated while the reactor is in operation. The simplified schematic shown in Figure 13 is that part of the system to the right of Valve No. 2 in Figure 18. The exception is the case which arises when the helium and fluorine content reaches the stage where there is an apparent inability of the control system to increase fuel temperature. This situation actually occurred once in NAA-62-2, and was corrected by a technique of overpressurizing with clean BF$_3$ with the refrigerator turned off (further reducing the fuel temperature), and then bleeding gas off into the purification system. The excess gas can be purified and returned to the supply bottle. The annulus was then repressurized; and, when the refrigerator was again turned on, the temperature returned to the desired value automatically.
APPENDIX D
PROBLEMS WITH THE USE OF BORON TRIFLUORIDE

Boron trifluoride is a chemical poison, and must be handled with care. When released into the air, it hydrolyzes instantly, even in the dryest climates, generating a white cloud containing boric oxide (B$_2$O$_3$) and hydrofluoric acid gas (HF). This gas is toxic and very corrosive. Because of the presence of B$_2$O$_3$, the gas is always visible, whereas HF alone would not be. Properly handled, BF$_3$ presents no problem. It is not corrosive, and will not react with metals, except alkali and alkaline earth metals.

Reactor operators have three primary questions regarding the use of BF$_3$:

1) What will the BF$_3$ do to the flux in adjacent positions?

   The adjacent flux will be lowered somewhat, dependent upon the macroscopic cross section, including the BF$_3$. If the experiment objective includes a constant flux, the flux variation in immediately adjacent positions will also be reduced.

2) What will be the effect on reactivity if the BF$_3$ escapes?

   The loss of BF$_3$ represents an introduction of reactivity. The amount of reactivity introduced is a function of boron-10 content, position in the reactor, and rate of loss of gas. In NAA-62-2, located in the first reflector row, sudden removal of all BF$_3$ was undetected in the reactor control room.

3) What if the BF$_3$ leaks into the reactor coolant?

   The BF$_3$ would immediately hydrolyze and form boric acid and hydrofluoric acid. The solution would be quite dilute, and would be negligible in a once-through system. In a recirculating system, some corrosion could result in the removal of some metal by the HF. Here again, the use of a separated isotope would be an advantage, for much less gas would be involved, and hence there would be less corrosion. The boric acid should be removed by the water purification system, as should the HF and corrosion products; but, for a short time, there might be a dilute "poison cloud" circulating through the reactor.

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The loss of any appreciable amount of BF$_3$ would result in an instantaneous flux increase in the experimental assembly. This could be detected readily, and the instrumentation could be connected to energize the reactor scram system. This safety feature was included in the NAA-62 experiments, and would probably be required by the reactor operator in the case of any other such experiment.

A question arises regarding the deposits of boron and lithium fluoride on the surface of the BF$_3$ chamber and the corrosion of the container walls. Since radiolysis involves no transmutation, the isotopic distribution of deposited boron is the same as that in the gas, and the boron-10 will subsequently burn out. The annulus from NAA-62-1 was sectioned and examined following irradiation.

The surfaces of the BF$_3$ containment vessel were dry. They were coated with a dark deposit, which was loose and flaky at the surface and granular near the Type 304 stainless steel walls. This deposit was noticeably heavier at the ends of the vessel, the lower end being most heavily coated. The inner tube of the vessel was more heavily coated than the outer tube. Although no actual measurements were taken, the deposits appeared to be no more than about 5 mils thick, except at the bottom where loose particles settled. Metallographic examination of the Type 304 stainless steel BF$_3$ containment vessel revealed no evidence of intergranular penetration, pitting, or other forms of corrosion. Postirradiation measurement of the vessel walls showed them to be within the tolerance of the as-built dimensions.

As discussed earlier, the purpose of the flux control devices is explicit in their name—to control the flux. If the reactor can deliver a flux close to that desired and maintain it there, the device will do little to change conditions. On the other hand, if there are wide variations in flux, the device tends to smooth these variations out for its own position and to a lesser degree for the immediately adjacent positions.
REFERENCES


9. F. Hudswell et al., "Corrosion Experiments with Gaseous Boron Trifluoride," AERE C/R 468 (June 1950)