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ISPO C-50 Phase I

DIVERSION ASSUMPTIONS FOR HIGH-POWERED RESEARCH REACTORS

F. T. Binford

Oak Ridge National Laboratory

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ISPO-C-50 Phase I  
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F. T. Binford

1. INTRODUCTION

1.1 Scope

This study was performed at the Oak Ridge National Laboratory pursuant to a request by the International Safeguards Project Office (ISPO) of the Brookhaven National Laboratory. The study, designated Task C-50 under the Program for Technical Assistance to International Atomic Energy Safeguards (POTAS), deals with diversion assumptions for high-powered research reactors — specifically, MTR fuel; pool- or tank-type research reactors with light-water moderator; and water, beryllium, or graphite reflectors, and which have a power level of 25 MW(t) or more.

The objective is to provide assistance to the IAEA in documentation of criteria and inspection observables related to undeclared plutonium production in the reactors described above, including:

- a. criteria for undeclared plutonium production,
- b. necessary design information for implementation of these criteria,
- c. verification guidelines including neutron physics and heat transfer, and
- d. safeguards measures to facilitate the detection of undeclared plutonium production at large research reactors.

Fulfillment of this objective consists of two phases, of which this study covers only the first.

## PHASE I

1. Define criteria which the IAEA may use to assess the credibility of undeclared production of significant\* amounts of plutonium annually within or at the periphery of the active core along with quantitative estimates of the production rate (taking into account both physical possibility and the ability to operate the reactor safely).
2. Suggest methods for making rough, quantitative, practical estimations of plutonium production for typical cases.
3. Specify the necessary reactor design information for assessing the capabilities for making practical estimates of plutonium production.
4. Suggest guidelines for verification, including neutron physics, heat transfer, and mechanical handling of radioactive components.

## PHASE II

It is contemplated that Phase II of the study will consist of providing assistance to the IAEA in determining the clandestine production potential for plutonium in six to ten large research reactors under IAEA safeguards. This will include identification of the necessary data to be used for calculations and the calculation of the reactor operating conditions and fuel usage necessary to produce one or more significant quantities of plutonium per year based on the available data.

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\*A significant amount of plutonium production has been defined by the IAEA to be 8 kg or more. In this work it has been taken to mean 8 kg of fissile plutonium ( $^{239}\text{Pu}$  and  $^{241}\text{Pu}$ ).

As a result of the Phase I study, the following reactors are regarded as probable candidates for the Phase II study:

BR-2, Mol, Belgium	100 MW
R-2, Nykoping, Sweden	50 MW
JMTR, Oarai-Machi, Japan	50 MW
HFR, Petten, Netherlands	45 MW
Tamuz-I, Bagdad, Iraq	40 MW
María, Swierk, Poland	30 MW

In addition to these, there are a few tank-type reactors now operating at lower power levels which might become candidates if their power were to be increased.

## 1.2 Method of Analysis

The reactors listed in the previous section are reasonably similar; however, they differ significantly in two respects which are relevant to their potential for producing plutonium. The first is power level and the second is their physical configuration which determines the amount of space in and around the reactor core into which fertile material can be loaded. Moreover, in each case there is a variety of loading patterns and operating cycles which could be used. While it may be possible to determine that pattern and that cycle which optimize production, it must be remembered that concealing the fact that plutonium is being produced is also an objective of the operator. Since use of optimum conditions may make the fact that plutonium production is going on quite obvious, the operator may elect to use less optimized conditions which are more susceptible to concealment.

For these reasons, examination of the potential to produce plutonium has been approached in a generic fashion. While sufficient information has been given to estimate the practical upper limit of

plutonium production in these reactors, accurate estimates of the production rate for various production schemes require calculations based upon the specific characteristics of the reactors under consideration and are reserved for Phase II of Task C-50. Similarly, sufficient information about the various operational elements which must be present during plutonium production is given so that upon observing them a well-trained inspector should be able to draw strong inferences concerning whether or not clandestine production is occurring. As in the case of production, an accurate quantitative estimate of the magnitude of these observables depends upon the production scheme being used.

Consequently, the body of this report provides a description of how plutonium can be produced in a research reactor with emphasis on the observable differences between the operational characteristics of a reactor which is producing plutonium and one which is not. The general principles of research reactor operation are described in Section 3. Section 4 discusses methods of plutonium production, while Section 5 deals with a number of ancillary considerations. In Section 6 consideration is given to those elements of the operation which provide evidence of plutonium production. Section 2 contains a summary of findings and the conclusions which may be drawn from them.

## 2. SUMMARY, CONCLUSIONS, AND RECOMMENDATIONS

During this study it became apparent that currently there exists no simple, straightforward method to detect the production of undeclared plutonium by merely monitoring the values of easily observable parameters. There are, however, a number of distinguishable differences between normal operation and operation in a plutonium-producing mode. Observation of these can provide evidence of plutonium production.

The potential for plutonium production can be estimated using neutron balance techniques and, if sufficient details are known, specific neutronic calculations.

## 2.1 Plutonium Production Potential

The maximum possible rate of  $^{239}\text{Pu}$  formation in research reactors of the type under consideration is about 0.95 g per megawatt-day, and the maximum actually achievable is 0.5 to 0.65 g per megawatt-day. It is unlikely that a plant factor greater than 85% can be achieved so that, in order to produce 8 kg of  $^{239}\text{Pu}$  per year, the reactor must operate at a minimum power level of 40 to 50 MW.

A second limitation is the quantity of fertile target material which must be present in the core. Although as much as 6.2 g of fissile plutonium can be produced per kilogram of  $^{238}\text{U}$ , this requires a very long irradiation (fluence of about  $8 \times 10^{21}$  n/cm<sup>2</sup>). Moreover, at this level the product is encumbered by the presence of large quantities of nonfissile plutonium. At a fluence of  $5 \times 10^{20}$  n/cm<sup>2</sup> (which only takes one-sixteenth the time), the target contains 2 g of fissile plutonium per kilogram and is of much better quality. In general, production is maximized by using many short irradiations rather than a few long ones. The time required to reach a given fluence and, hence, the quantity of  $^{238}\text{U}$ , for a given production rate, that must be present in the core, depends on the magnitude of the flux to which the targets are exposed. For a given configuration the flux is proportional to the reactor power. An accurate estimate of the quantity of fertile material which must be present at all times in order to produce 8 kg of plutonium per year must be obtained on a case-by-case basis; however, for a 50-MW reactor it probably lies between 1.5 and 2 metric tons. If uranium metal can be used as the fertile material, it may be possible to get that much into the flux; but, if currently available fuel fabrication techniques must be used, more than twice the volume occupied by metal would be required. Under the latter constraint, power levels approaching 100 MW would be necessary in order to produce a significant quantity of plutonium annually.

## 2.2 Specific Production Estimates

Detailed neutronic calculations based on specific fuel and target configurations are necessary if accurate production estimates are to be obtained. These are planned for Phase II of this study. For this purpose, it is necessary to obtain for each reactor sufficient information concerning the composition of each possible configuration of target and fuel elements to permit the effective fluxes and cross-sections in the individual elements to be computed. Production in each position as a function of fluence can then be estimated using an appropriate isotope or burnup computer code.

Because it is impossible to predict the production scheme which might be adopted for a given reactor, a number of configurations should be examined with fluence as a parameter.

Other parameters which must be considered include power level, permitted start-of-cycle excess reactivity, and fuel cycle length. The frequency with which targets are replaced and, hence, the annual target usage, will depend upon the neutron flux and the fluence to which they are irradiated.

## 2.3 Inspector Observables

Evidence of possible plutonium production may be supplied by observation of one or more of the following:

- a. The actual presence of undeclared plutonium detected by an interrogation device.
- b. The presence of targets containing a large quantity of irradiated or unirradiated  $^{238}\text{U}$  in the reactor or in storage.
- c. A substantial increase in core component traffic over that required for normal operation.
- d. The use of heavy core components containing pins or thick plates.

- e. A substantial increase in the number of fuel-like components loaded in and around the reactor core.
- f. A reduction in  $^{235}\text{U}$  consumption in the normal fuel without a concurrent decrease in power.
- g. Structural or engineering changes designed to increase the number of movable components located in and around the core.
- h. A reduction in the use of the reactor as a neutron source for experiments and radioisotope production.

#### 2.4 Conclusions

It may be concluded that for research reactors operating at a power level in excess of 40 to 50 MW sufficient neutrons are generated to permit the production of 8 kg or more of fissile plutonium annually, but that the capability for such production may be reduced to less than half of this by space limitations within and around the core unless the fertile material can be used in metallic form.

It may be further concluded that at this level of production, which for the reactors under consideration requires dedication of all or a large fraction of the reactor to production, there will be sufficient differences between normal operation and operation in the plutonium-producing mode so that they can be detected by an inspector or by surveillance devices, provided of course that access to the relevant areas in and near the reactor is available at the proper times. Production at lesser levels would be correspondingly more difficult to detect and easier to deliberately conceal.

#### 2.5 Recommendations

For the inspector to be able to recognize those changes in the operating pattern which may signify that plutonium is being produced,

he must be familiar with the normal operating pattern. He should, therefore, have a sound knowledge of the characteristics of the reactor for which he is responsible. Moreover, this knowledge should extend beyond the limits of the information contained in the "Design Information Questionnaire" and the "Facility Attachments" and should include a study of all available open literature documents concerning the reactor.

Although such devices may not exist at present, consideration should be given to the development of surveillance instruments which can monitor the reactor power level independent of the regular instrumentation and to the feasibility of the use of sophisticated instrumentation to detect the presence of plutonium fissions or the presence of large quantities of  $^{238}\text{U}$  in the reactor core by using noise analysis techniques.

Finally, because it is obviously much easier to conceal the annual production of a small quantity — one or two kilograms of plutonium — than that of a "significant" quantity, it is suggested the inspectors be alert to this possibility.

### 3. RESEARCH REACTOR DESIGN AND OPERATION

#### 3.1 General Characteristics of the Reactors

The research reactors listed in Section 1.1 have a number of features in common. They are all light-water-cooled, thermal reactors, moderated with water or with a combination of water and beryllium. Reflection is provided by beryllium, graphite, or deuterium oxide together with some water. All are located within or immediately adjacent to a large pool of water which provides shielding and facilitates the handling of fuel and other radioactive components which must be removed from or inserted into the reactor core and which permits easy access to the reactor for the purposes of maintenance, experiment servicing, or other necessary operations. The pool, which

is usually divided into two or more sections by means of removable partitions or dams, also provides shielded storage space for irradiated fuel and other radioactive components.

At present, all of these reactors employ plate-type, aluminum-clad fuel elements which contain uranium highly enriched in  $^{235}\text{U}$ \* (HEU) and have either an approximately square or a circular cross-section. Depending upon the reactor, the fuel elements will initially contain 250 to 400 g of  $^{235}\text{U}$ , of which normally about 50% is consumed during the life of the element. In some cases, a neutron-absorbing "burnable poison" is incorporated in the fuel to reduce its initial reactivity, permit heavier loading, and thus extend the fuel life. Current practice is to fabricate the individual fuel plates by a "picture-frame" technique with the fuel cores being made from uranium-aluminum alloy or by powder metallurgy using aluminum and uranium ( $\text{UAl}_x$ ) or aluminum and  $\text{U}_3\text{O}_8$ . They are then rolled and formed to the proper shape — rectangular plates for the square or box-type elements or concentric cylinders for the circular elements.

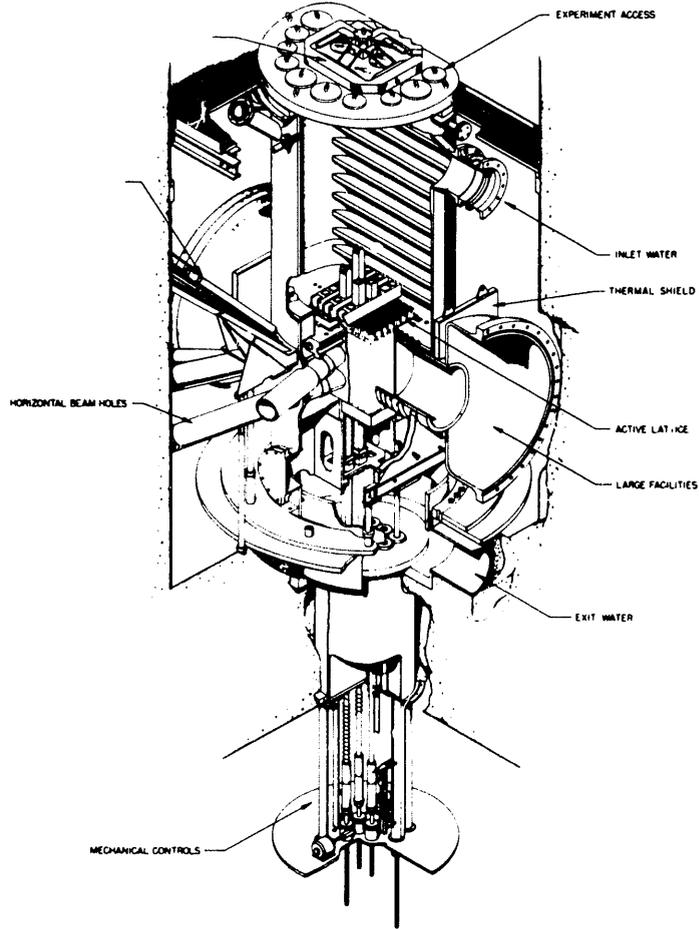
The fuel elements are maintained upright within the reactor core by means of end-mounted adapters which fit into horizontal grid plates designed to accommodate them. An operating core may contain 25 to 40 fuel elements, depending upon the reactor, although the grid plate may have space for perhaps twice as many. Those grid positions not occupied by fuel will contain reflector pieces, control elements, or, frequently, experimental rigs. A typical arrangement is shown in Figs. 3.1 and 3.2.

The cores of four of the reactors — BR-2, R-2, HFR, and JMTR — are contained in tanks or pressure vessels which are immersed in the reactor pools. Maria is a "pressure-tube" reactor in which each fuel element is enclosed in a separate tube, all of which are connected to an

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\*Conversion to low-enriched uranium (LEU) is currently being contemplated for most of these reactors.

ORNL-LR-DWG 13392AR2



PRESSURIZED-WATER LOOP THROUGH ACCESS OPENING IN REACTOR COVER PLATE

ORNL-LR-DWG 49839R

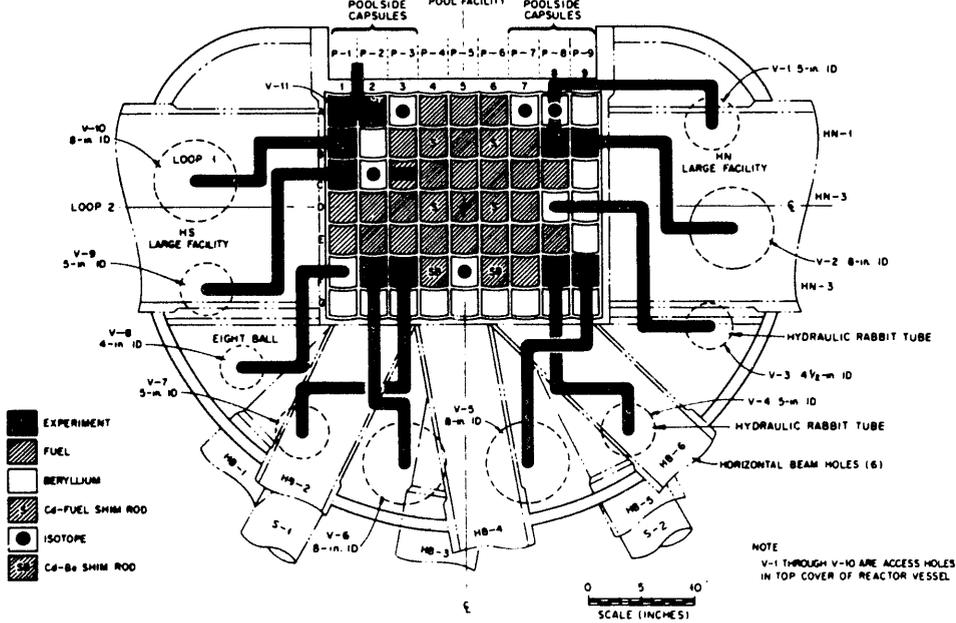
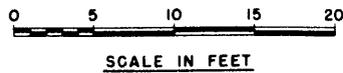
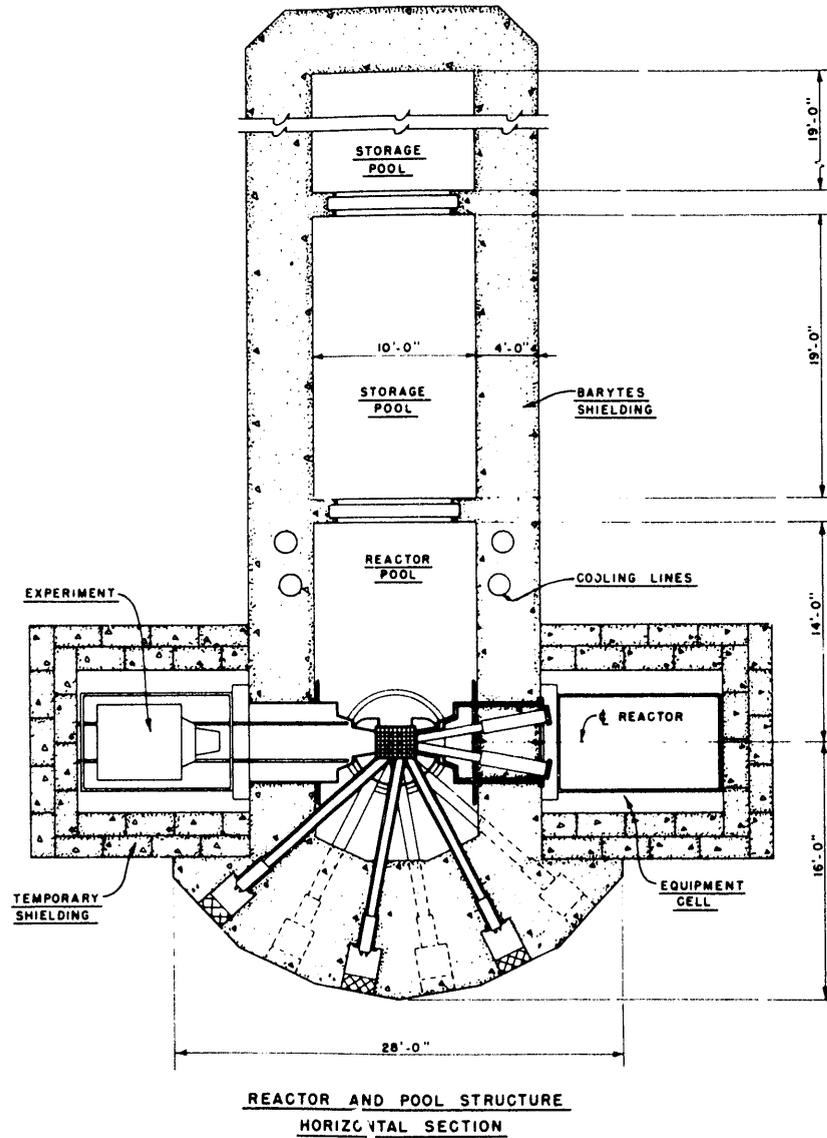


Fig. 3.1. Typical Tank-Type Research Reactor



C-11386-R1 NOT CLASSIFIED

Fig. 3.2. Research Reactor Showing Storage Pools

upper and lower plenum. Tammuz-I has a unique cooling system which includes a chimney directly connected to the pool. Aluminum is the material of construction used for the vessels and tubes.

The water moderator also serves as the primary coolant which removes the nuclear heat from the reactors. These reactors operate at low to moderate pressures — 3-14 atm. abs. — and the coolant flow rate is adjusted to maintain the exit temperature well below saturation. In general, it is also expedient to keep the fuel surface temperature and that of other in-core components near or below the coolant saturation temperature so that the coolant velocity must be sufficiently high to provide a heat transfer coefficient which is adequate for this purpose. With the exception of Tammuz-I, the primary coolant is pumped vertically downward through the reactor core. It is then passed through heat exchangers where the heat is given up to a secondary coolant, also water. Cooling water is supplied to the secondary side of the heat exchangers either from an external source of cold water such as a river or sea or is recirculated through cooling towers to waste the heat to the atmosphere.

A small fraction of the reactor heat escapes to the pool due to radiation and conduction through the reactor vessel. Also, heat is supplied to the pool water from the radioactive decay of fission products in stored irradiated fuel so that the pools are also provided with a cooling system.

Appropriate fractions of both the primary coolant and the pool water are circulated through demineralizing equipment in order to maintain water purity.

Reactor control is exercised by means of control elements or "rods" fitted into the reactor lattice among the fuel elements. These control rods contain a neutron-absorbing material or "poison" such as cadmium, hafnium, boron, or cobalt. The reactor is started up by raising the control rods so that the poison section is moved out above the core until a multiplication factor slightly greater than unity is achieved. When the proper power level is reached, the position of the

rods is adjusted to provide a multiplication factor of unity. The rods are further withdrawn throughout the operating cycle to compensate for reactivity lost as the fuel is consumed.

In most cases the control rods serve also as safety rods, being connected to their drives by a magnetic coupling which, when deenergized by a protection system signal, allows the poison section to fall by gravity into the core. The drives may be located either above or below the core, with the latter being the most common arrangement for high-powered reactors.

Four of the reactors — R-2, HFR, Tammuz-I, and JMTR — employ "fuel followers" which are modified fuel elements attached to the bottom half of the control rods. As the poison section is moved out of the top of the core, fuel is drawn in the bottom. This arrangement increases the reactivity worth of the rods. Depending upon the reactor, five to ten such rods are generally utilized. Neither BR-2 nor Maria employs fuel followers; and, in the latter case, control is exercised by using ten cylindrical control rods using  $B_4C$ . An additional six  $B_4C$  rods are used independently as safety rods. While there is no set rule, it is usual in reactors of this type for the control and safety system to be able to handle about twice the beginning-of-cycle reactivity.

The acknowledged mission of all these reactors is to generate neutrons which may be used for basic physical research, the testing of power reactor materials and components, production of radioisotopes, neutron radiography, activation analysis, and many other constructive purposes. For this reason, the reactors are equipped with a variety of experimental facilities to give access to the neutron field. These include beamports and thermal columns exterior to the core, as well as facilities for in-core experiments including circulating loops and flux traps. Usually, provision is also made to insert and remove samples from the reactor core during operation by means of one or more hydraulically operated "rabbit" tubes. Both the R-2 and HFR vessels and the Tammuz chimney have been designed so that relatively

large experiments can be positioned in the reactor pool exterior to the vessel but quite close to the core. These "poolside windows" are frequently used for shielding and neutron damage experiments.

In general, thermal neutron fluxes of the order of a few times  $10^{13}$  to a few times  $10^{14}$  are available in the experimental facilities with perhaps  $10^{15}$  being available over a small region in the flux traps of the higher-powered reactors. Fast neutron fluxes of a few  $\times 10^{14}$  are also available. The actual values of the fluxes and their spatial distribution depends upon the arrangement of the core components and the operating power level.

Containment is provided in all cases by either a pressure-tight building or by a controlled-leakage arrangement which permits only filtered air to leave the containment envelope.

More detailed descriptions of the reactors under consideration can be found in the publications listed in the attached bibliography.

### 3.2 Heat Transfer Considerations

Aside from the necessity of designing a chain-reacting device which will provide a source of neutrons in a configuration suitable for experimental uses and whose power level can be regulated in a safe and orderly fashion, the overriding considerations in research reactor design are those related to the removal of heat.

A 50-MW reactor similar to the Oak Ridge Research Reactor (ORR), illustrated in Figs. 3.1 and 3.2, will normally contain about 30 fuel elements and four to six shim safety rods. The total heat transfer area of such a core is approximately  $53 \text{ m}^2$  so that the average heat flux is about  $95 \text{ W/cm}^2$ . Because of variations in the power density, both radial and axial, the peak heat flux may be perhaps 2.5 times this or  $238 \text{ W/cm}^2$ . These values are nearly twice those found in a 3000-MW(t) pressurized water reactor. Thus, while the cores of high-powered research reactors are physically small, they are intense sources of heat.

To remove this heat, the fuel elements must be designed to present an adequate surface area to the cooling water, the coolant flow rate must be high enough to remove this heat without an excessive rise in temperature, and its linear velocity through the fuel elements must be sufficiently high to provide satisfactory heat transfer coefficient without producing a prohibitive drop in pressure. It is for these reasons that fuel elements assembled from thin plates are almost universally used in high-powered research reactors.

Bulk coolant flow rates in the reactors under consideration are generally in the range of 23 to 37 L/s per fuel assembly. The velocities vary from 6 to 20 m/s, while the pressure loss due to friction and shape changes varies from 1 to 2.5 atm. The heat transfer coefficients range from 3-6 W/cm<sup>2</sup>°C.

The foregoing values are valid for "normal" operating cores. While the maximum bulk coolant flows are essentially fixed by the capacity of the coolant pumps and the flow area of the individual elements is fixed by the fuel element design, the flow rate through the elements, and hence the velocity, depends upon the total available flow area, i.e., the number of fuel elements present. A relatively small fraction of the total flow does not pass through the fuel but is required to cool other core components such as reflector pieces, experimental rigs, core structural elements, etc. Nevertheless, to a first approximation, both the coolant flow per element and velocity are inversely proportional to the number of fuel elements present. Since the heat transfer coefficient varies directly as the 0.8 power of the velocity, it decreases with the increasing number of fuel elements.

On the other hand, the heat transfer area increases directly with the number of fuel elements so that, for a given power level, the average heat flux varies inversely as the number of fuel elements. However, because the ratio of peak to average power density may actually increase as the number of fuel elements increases, the larger core may not produce a lower peak heat flux. Moreover, although the

bulk coolant temperature rise is not affected by the number of fuel elements, because of the lower flow, the rise in the hottest channels may actually be as great or greater in the highest power density channels of a large core than it is in a small core. Thus, since it is the hottest channels which govern the reactor thermal-hydraulic behavior, it cannot be assumed a priori that increasing the number of fuel elements will always lead to an improvement in the worst heat transfer conditions.

The basic criterion which is widely used to dictate the thermal design of reactors of the type under consideration is:

"The combination of coolant flow, velocity, and inlet temperature shall be capable of preventing the onset of nucleate boiling anywhere in the core."

In some respects this criterion is conservative because the heat transfer coefficients associated with nucleate boiling are considerably higher than those associated with conduction. Thus, operation in this regime will permit an increase in heat flux (and, hence, power) without a proportional increase in fuel surface temperature. However, the nucleate boiling regime borders on a region of flow instability and is close to conditions which could lead to bulk boiling, film blanketing, and subsequent melting of the fuel. Also, the formation and collapse of bubbles may lead to fluctuations in the neutron signals feeding the reactor instrumentation, thus producing a "noisy" response which could mask developing malfunctions.

For the conditions considered, the onset of nucleate boiling usually requires that the fuel surface temperature exceed the saturation temperature by 5 to 15°C. An even more conservative criterion is to require that the fuel surface temperature nowhere exceed the saturation temperature.

All of the reactors were originally designed to one or both of these criteria. In most cases experience showed that the designs were

overly conservative and that by making modest changes it was possible to increase the original power level of the reactor by a significant amount. Such changes include increasing the capacity of the secondary cooling system, increasing the primary coolant flow, decreasing the primary coolant inlet temperature, increasing the heat transfer area of the fuel plates, and careful tailoring of the core configuration to minimize the peak power density. Beyond this, it may in some cases be possible to increase the reactor pressure to raise the saturation temperature, although this would be difficult if the system had not been originally designed with such a contingency in mind. In extreme cases, it might be considered worth the risk to operate in the nucleate boiling regime.

Calculation of the heat transfer conditions is accomplished using a series of "hot channel" and "hot spot" factors which account for variations in axial and radial power density as well as uncertainties in fuel density, channel width, and flow velocity. For reactors in which the coolant flow is downward, both the maximum coolant temperature and the minimum pressure will occur at the bottom of the active region of the fuel.\* Generally, the maximum power density (hence, maximum heat flux) will occur either slightly below the core centerline or at the bottom of the core, depending on the magnitude of the reflector peak caused by the bottom reflector. It is conservative, therefore, to assume that the maximum heat flux occurs at the bottom of the core where the coolant temperature is highest and the saturation temperature lowest. The hot spot and hot channel factors are then applied and the fuel surface temperature calculated. This is then compared to the saturation temperature or to a nucleate boiling correlation to determine if the criteria are met. Because the power density profile may change significantly throughout the operating

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\*In the case of Tammuz-I, in which the flow is upward, the maximum coolant temperature and minimum pressure occur at the top of the core.

cycle as a result of control rod movement and fuel consumption, it is often desirable to calculate the thermal conditions at various stages of the cycle.

### 3.3 Research Reactor Fuel

All of the reactors utilize fuel elements assembled from thin plates. These fuel plates consist of a fuel core containing uranium-bearing material which is clad on both sides and on the ends with aluminum metal. The plates are fabricated by forming fuel cores into the proper size and placing them in an aluminum "picture frame." Aluminum cover plates are placed on both sides of the framed core, the edges welded, and the whole assembly rolled to the proper thickness. The rolled plates are then trimmed to the correct size, straightened, and, where necessary, formed to the proper curvature. They are then assembled into finished fuel elements.

Four of the reactors — HFR, R-2, JMTR, and Tammuz-I — utilize "box-type" fuel which consists of a number of flat or slightly curved plates assembled in the form of a rectangular parallelepiped. The fuel used in BR-2 and Maria consists of six concentric cylindrical plates and, thus, has a circular rather than an approximately square cross-section. Typical fuel elements are shown in Fig. 3.3.

Three methods to produce fuel cores are presently in use. The oldest method (which is now used infrequently) is to produce a uranium-aluminum alloy billet by melting the two metals together. The billet is rolled into relatively thick plate from which the cores are punched. The amount of uranium which can be contained in such cores is limited because it is extremely difficult to roll alloys which contain a high percentage of uranium. Although higher percentages have been achieved, attempts to routinely produce fuel plates having alloy cores containing more than about 26 wt % uranium have not been very successful.

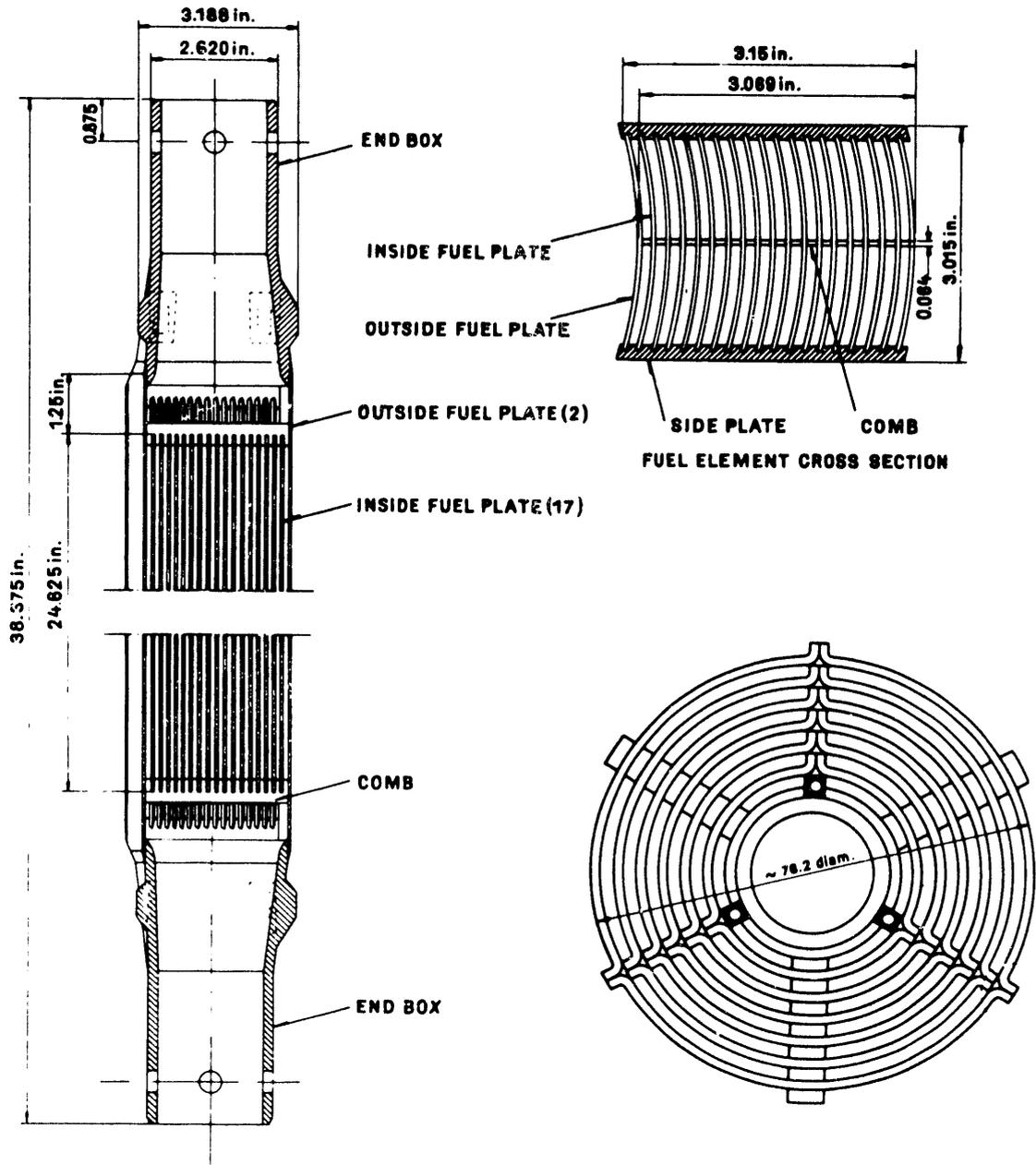


Fig. 3.3. Typical Research Reactor Fuel Elements  
 (taken from IAEA Directory of Nuclear Reactors, Vol. II)

The use of uranium-aluminum alloy cores has been largely displaced by cores fabricated by powder metallurgy techniques. These processes use a mixture of aluminum powder and either uranium metal or  $U_3O_8$  powder. The ingredients are blended in the desired proportions, shaped in a die, and hot pressed to form the core. The powder metallurgy methods have a number of advantages over the use of alloy: the plates are easier to roll, homogeneity of the fuel distribution can be better controlled — as can uniformity of the cladding thickness — and higher fuel densities are obtainable. Fuel cores containing up to 40 wt % of uranium are routinely produced by these methods.

An ongoing development program using  $U_3Si$  or  $U_3Si_2$  as the uranium-bearing materials is aimed at producing even higher density fuel's. This could yield uranium densities as high as 70 wt %.

Neither unalloyed uranium metal nor  $UO_2$  have been successfully used for research reactor fuel plate fabrication although both are used in pellet form in fuel pins. It is extremely difficult to roll the metal. Moreover, there must be a metallurgical bond between the cladding and the fuel core, both for heat transfer purposes and to prevent the formation of a void region into which fission gases could migrate and cause separation of the cladding, thus narrowing the coolant channels. Because the metal tends to swell upon irradiation, any such bond would be suspect.

The early natural uranium graphite-moderated reactors employed slugs (short cylinders) of uranium clad with either aluminum or a zirconium alloy and frequently bonded with silicon, but this technique is not suitable for plate-type fuel. More recently, the metal has been used in the form of thin cylindrical rods clad in a stainless steel tube. These rods or pins contain a heat transfer medium such as sodium or sodium-potassium alloy (NaK) to conduct heat from the fuel to the pin walls. While suitable as fast reactor fuel, the high thermal neutron capture cross-sections of steel and liquid metal virtually preclude the use of this type fuel in a thermal reactor.

Some fuel plates have been produced and operated using  $\text{UO}_2$  rather than  $\text{U}_3\text{O}_8$  as the fuel-bearing material for the powder metallurgy process. Because of chemical reactions which take place between  $\text{UO}_2$  and aluminum during the rolling process, plates containing  $\text{UO}_2$  tend to swell and blister and are far more difficult to produce than are those containing  $\text{U}_3\text{O}_8$ .

Uranium dioxide is routinely used in power reactor fuel pins and quite high-density fuel (88 wt % uranium) can be fabricated by using pins rather than plates in a fuel element similar in configuration to that of a TRIGA fuel element. The main disadvantage is that for a given fuel content the heat transfer area of the pin-type element is significantly less than that of the plate type.

The French have developed a unique method using  $\text{UO}_2$  which results in a high-density, plate-type fuel. Small lozenges of  $\text{UO}_2$  are produced by pressing the powder. These are individually wrapped in aluminum or zirconium foil, enclosed in cover plates, and rolled to produce the fuel plates. The wrapped  $\text{UO}_2$  lozenges look like a piece of wrapped candy, hence, the name "caramel" fuel.

A tabulation of the relevant characteristics of the various fuel types is given in Table 3.1.

#### 3.4 Operating Considerations

The operating cycle of a research reactor is governed by a number of factors which include: the needs of the experimental program, the reactivity limitations imposed by the control and safety system, the characteristics of the reactor fuel, the frequency with which both preventive and corrective maintenance must be performed, and economic considerations with respect to fuel usage.

Although the pattern may vary, operating cycles usually last for two to four weeks, followed by a shutdown for refueling and servicing

Table 3.1 Characteristics of research reactor fuel materials

Fuel type	Compound density (g/cc)	Current practice, <sup>a</sup> g/cc of uranium in fuel core	Maximum <sup>b</sup> g/cc of uranium in fuel core	Remarks
U <sub>3</sub> Si-Al cermet	15.2	-	7.0	Developmental, plates
U <sub>3</sub> SiAl-Al cermet	14.2	-	5.2	Developmental, <sup>c</sup> plates
U <sub>3</sub> Si <sub>2</sub> -Al cermet	11.9	-	5.5	Developmental, plates
U <sub>3</sub> O <sub>8</sub> -Al cermet	8.1	<1.3	3.2	Operational, plates
UAl <sub>x</sub> -Al cermet	6.37	<1.3	2.3	Operational, plates
U-Al alloy	-	<0.7	6.0 <sup>d</sup>	Operational, plates
UO <sub>2</sub> pellets	10.96	9.6	9.6	Caramel fuel or pins
U pellets	19.1	19.1	19.1	Pins, requires thermal bond

<sup>a</sup>Based on uranium concentrations in currently operating reactors.

<sup>b</sup>Best estimates of maximum values currently thought to be achievable.

<sup>c</sup>Not promising.

<sup>d</sup>Extremely difficult to produce this concentration; practical value may be much lower.

of the experiments, which may take anywhere from a few hours to several days. The length of the operating cycle depends upon the initial excess reactivity and the power level of the reactor.

For reactors of the type under consideration, the loss in excess reactivity due to fuel burnup is in the neighborhood of 0.009%  $\Delta k/k$  per megawatt-day, provided the fuel contains no burnable poison. If burnable poison is present, this value will be less during the early life of the fuel because the loss in fuel is compensated for by burnup of the poison. In both cases, there is a greater loss in reactivity during the first two or three days due to the growth of fission product poisons, principally  $^{135}\text{Xe}$ . ( $\sigma_{2200} = 3.6 \times 10^6$  b.)

Once the reactor has shut down because the excess reactivity has been depleted, it cannot be restarted and operated at power until it has been refueled. If the startup is to take place within a few hours of the shutdown, it will be necessary to replace a relatively large fraction of the fuel because of the growth of  $^{135}\text{Xe}$  following shutdown. During operation, the equilibrium  $^{135}\text{Xe}$  concentration will compensate for 3-4%  $\Delta k/k$ , but once shut down, this will increase rapidly due to the decay of its parent, the 6.7 h  $^{135}\text{I}$  fission product. The rate of increase is highly flux dependent as is the time required for the 9.1 h  $^{135}\text{Xe}$  to decay. For example, in a reactor which has been operating with an average thermal flux of  $1.0 \times 10^{14}$ , the xenon concentration will increase to a maximum value of about six times the operating equilibrium value over a period of about 10 h. It then requires an additional 37 h for the xenon to return to its original concentration. Curves showing the growth and decay of this isotope following shutdown after operation to equilibrium at various neutron flux levels are shown in Fig. 3.4. It should be noted that since the neutron flux varies from fuel element to fuel element, the reactivity effects are also different for different elements.

The operating cycles at a given reactor are generally repetitive with a fixed operating period, followed by a short shutdown for

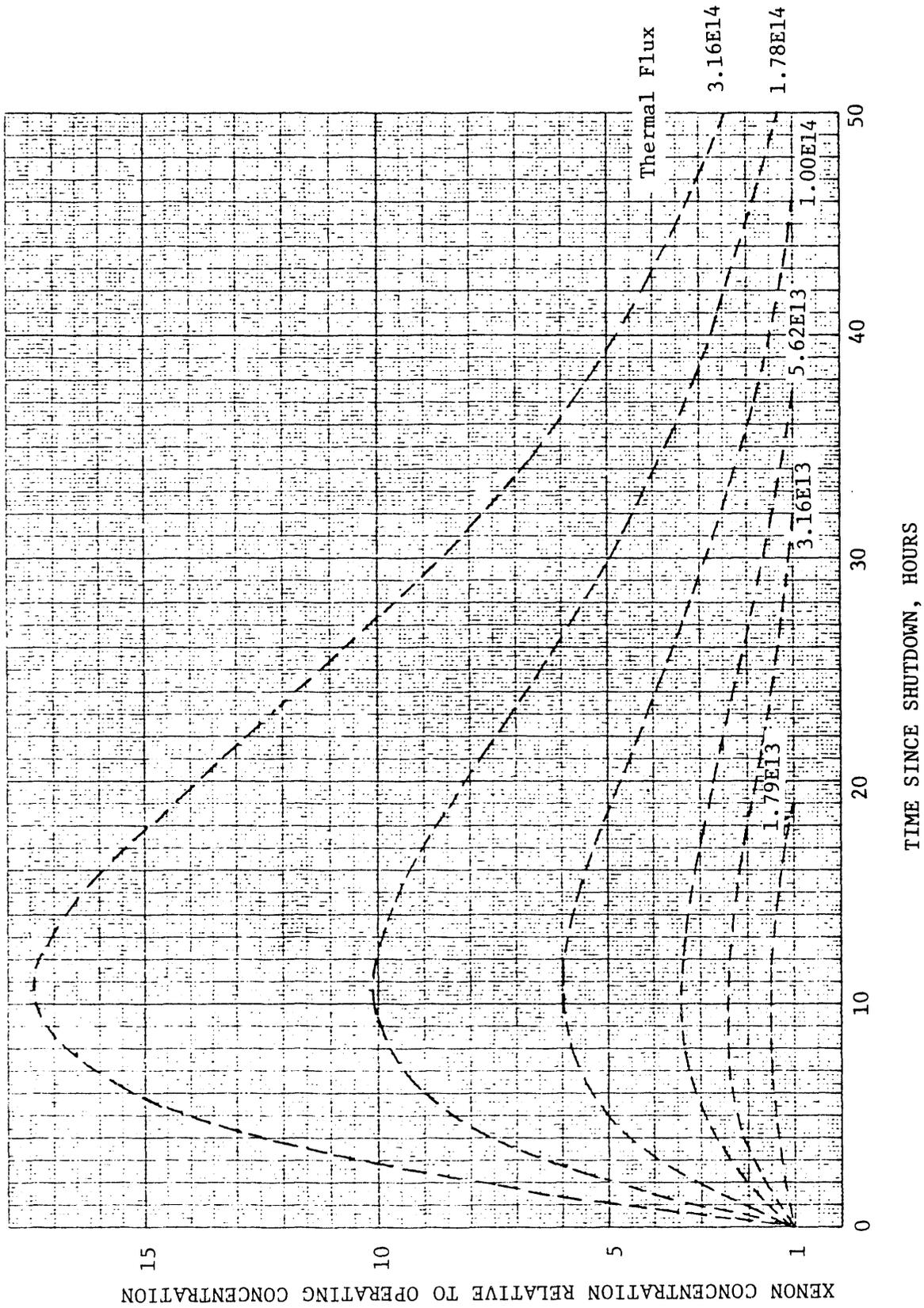


Fig. 3.4. Growth and decay of <sup>135</sup>Xe in a shutdown reactor with neutron flux as a parameter.

refueling and experiment servicing. Frequently, two or more long shutdowns are scheduled annually to permit major maintenance. In addition, there are occasional unscheduled outages due to malfunctions of one kind or another. Most of the reactors are able to achieve a plant factor of about 75%, although with careful planning plant factors of up to 90% are possible.

### 3.5 Fuel Management

Among the most important operating expenditures are those for manpower, utilities such as electricity and water, and the cost of the reactor fuel. The manpower cost is essentially constant whether the reactor is operating or not, and the fuel cost depends on the amount of energy produced per unit time together with the number of megawatt days which can be obtained from each fuel element. It follows that it is highly desirable from an economic standpoint to operate with the highest plant factor consistent with the needs of the experimenters and to obtain the maximum practical amount of energy from each fuel element before discarding it as spent.

Usually, all of the new fuel elements for a given reactor are identical in fuel content. However, except for the initial startup, the reactors are not operated with new fuel but with a mixture of partially spent and new elements. Generally, the core configurations are relatively constant from cycle to cycle. At the beginning of a cycle, there may be four or five new fuel elements in a 35-element core. The remaining elements will be in various stages of depletion and located in such a way as to provide the requisite excess reactivity and the appropriate neutron environment for the experimenters. During the operating cycle,  $^{235}\text{U}$  is consumed at a rate of approximately 1.24 g per megawatt-day, although this consumption rate is not uniformly distributed throughout the reactor core because the elements are not all operating at the same power level. For example, in a 14-d operating cycle, about 26 g of  $^{235}\text{U}$  will be consumed in an element operating at

1.5 MW, whereas only one-half that much would be consumed at a power level of 0.75 MW. At the end of the cycle, those elements which are regarded as spent are removed from the reactor, the remaining elements redistributed in the core, and new elements added to duplicate the previous start-of-cycle pattern.

Frequently, instead of redistributing the fuel in the core, some or all of it is removed and replaced with elements of similar  $^{235}\text{U}$  content which have been stored in the reactor pool following their use in previous cycles. This is done when it is desired to start the reactor up before there has been sufficient time for xenon to decay. It is often the practice to keep a preselected group of fuel elements on hand so that in the event of an unscheduled shutdown the core can be reloaded and the reactor restarted with a minimum of delay.

Of course, the core configuration is altered from time to time to fit the needs of the experiments. Under these conditions, there will be variations in the loading procedures until an "equilibrium" cycle is established for the new configuration.

It is highly desirable that as much as possible of the  $^{235}\text{U}$  originally present in fuel be consumed. It has been found that higher initial  $^{235}\text{U}$  loadings permit a higher fraction of the fuel to be consumed before the element is depleted to the point where it no longer contributes a useful amount of reactivity. Depletion of the  $^{235}\text{U}$  to about 50% of its original value is common; and, in some cases, this can be extended to 60% or 70% by the use of a burnable poison which allows a high loading by compensating for the extra reactivity. An ordinary 280-g element would thus be expected to produce about 115 MWd, whereas a 340-g element containing burnable poison might generate 175 MWd. The actual time over which this energy is generated may, however, be a year or more because of the practice of cycling the fuel in and out of the reactor. The amount of fuel consumption permitted in some reactors is arbitrarily restricted by safety regulations which limit the number of fissions permitted per unit volume of fuel core. In any case, an estimate of the average annual fuel element usage can be

obtained if the reactor power level, initial fuel weight, average burnup, and plant factor are known. For example, a 50-MW reactor using 280-g elements which can be burned to 50% and which has a 75% plant factor will consume  $50 \times 365 \times 0.75 \times 1.24 = 16,972$  g of  $^{235}\text{U}$  per year. This represents  $16,972 / (280 \times 0.5) = 121$  fuel elements per year.

Spent fuel elements must be stored for a period of time, usually three to six months, before being shipped to the reprocessing plant. This cooling period is required to allow the heat generation rate to decay to a point where the fuel can be handled in the shipping casks without the necessity for special heat-removal arrangements. The heat generation rate as a function of time after shutdown is shown in Fig. 3.5. The radiation levels are also considerably reduced; however, they are still extremely high so that massive biological shielding is still required.

The fuel is stored vertically in racks in the storage pool adjacent to the reactor. The water in the pool provides biological shielding and also cools the fuel by free convection. The racks are designed to provide a critically safe array either because of their geometry, by the inclusion of a neutron-absorbing material such as cadmium, or both. It is customary to provide criticality safety for fuel of at least the same  $^{235}\text{U}$  content as that in the new fuel elements. Typical storage racks are shown in Fig. 3.6.

Despite the high heat-generation rate, most research reactor fuel elements can be removed from the pool in air within two or three days after shutdowns, provided they are removed rapidly (to prevent blockage of the lower end by the pool water) and held vertically. In this position they are sufficiently cooled by free convection in air to prevent melting. They are, as pointed out above, extremely radioactive so that, if this is to be done, it must be done remotely or within a shielded space.

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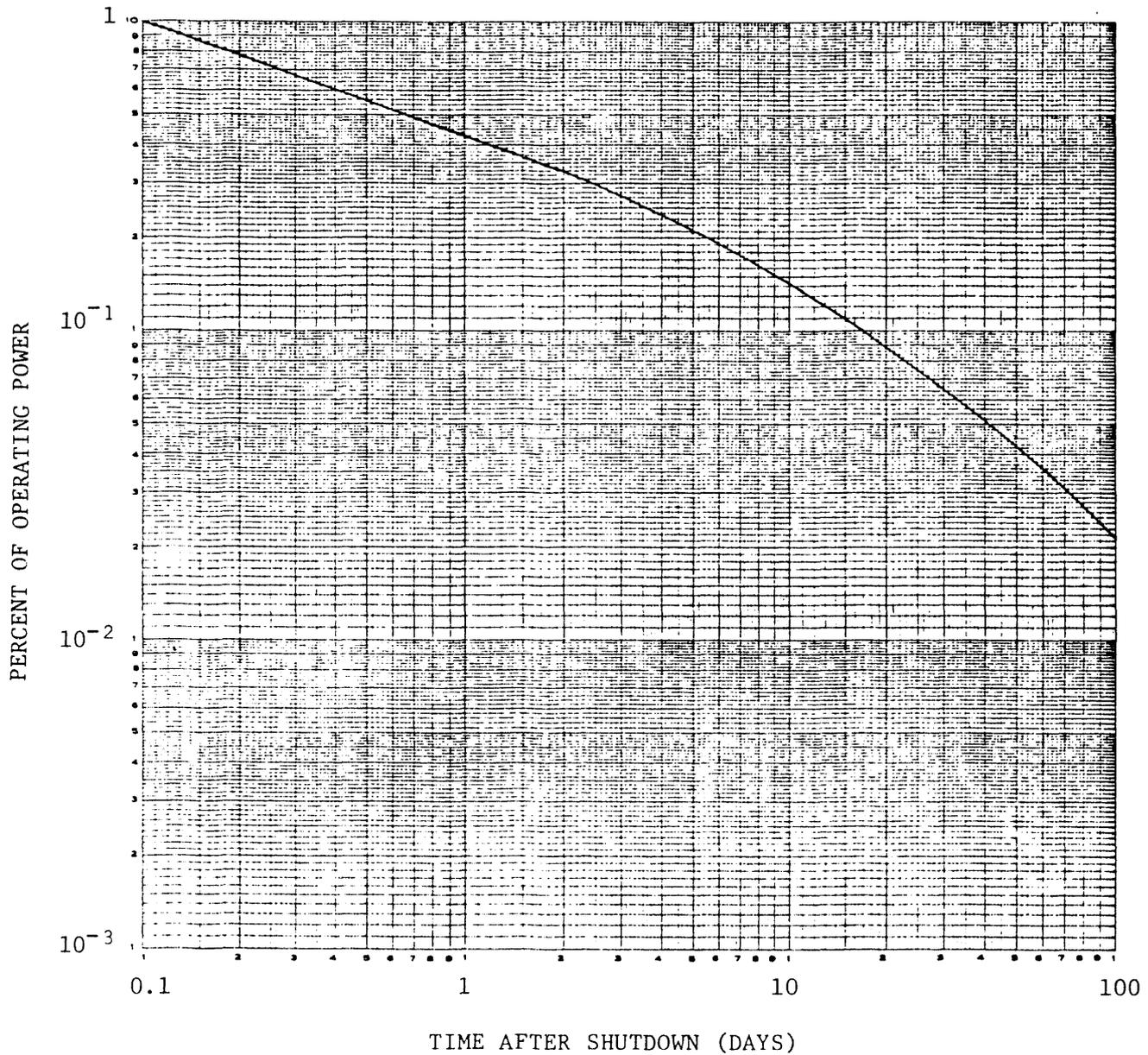


Fig. 3.5. Decay heat rate from typical research reactor fuel (based on 8 weeks of operation).

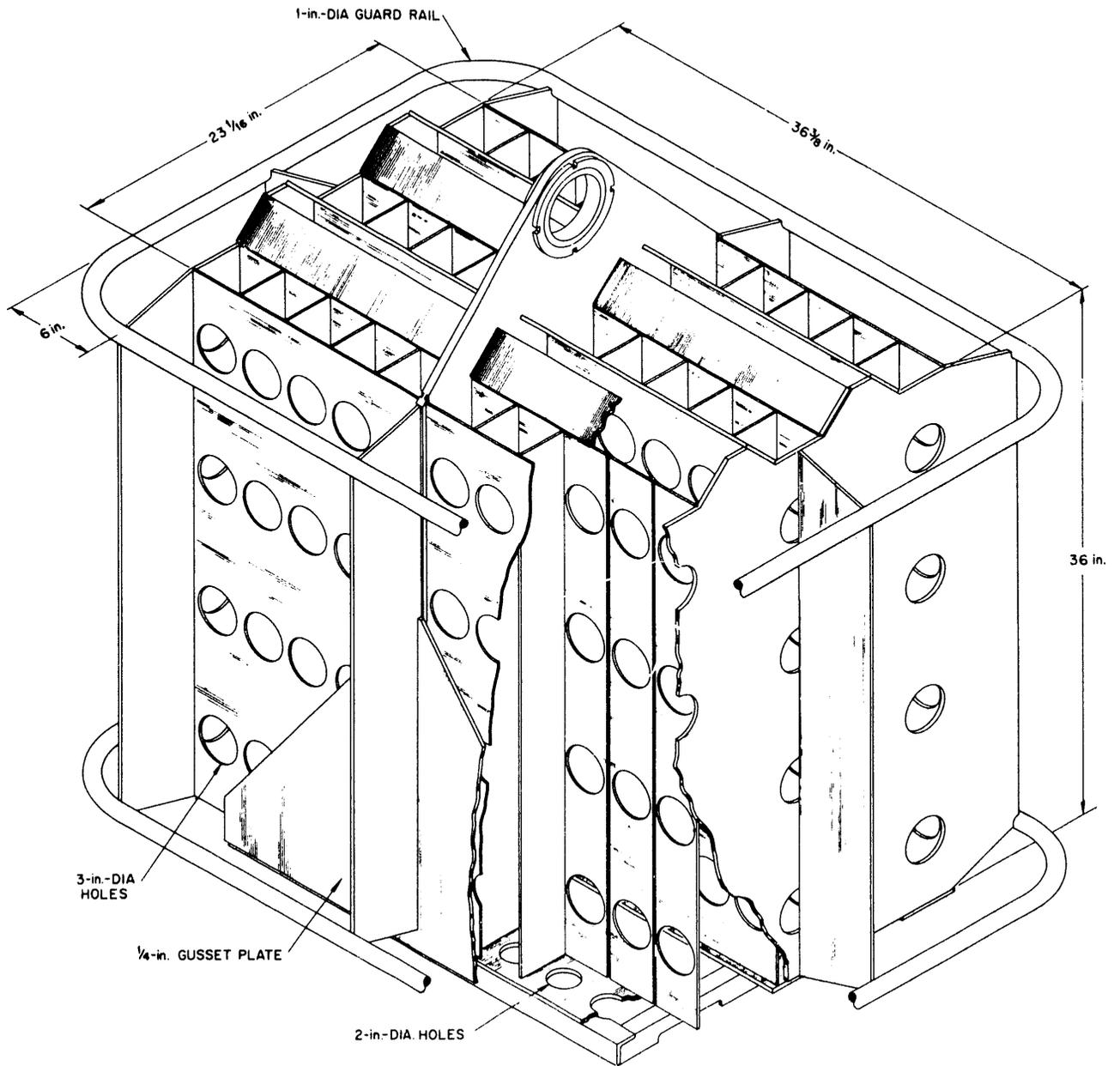


Fig. 3.6. Fuel Element Storage Rack

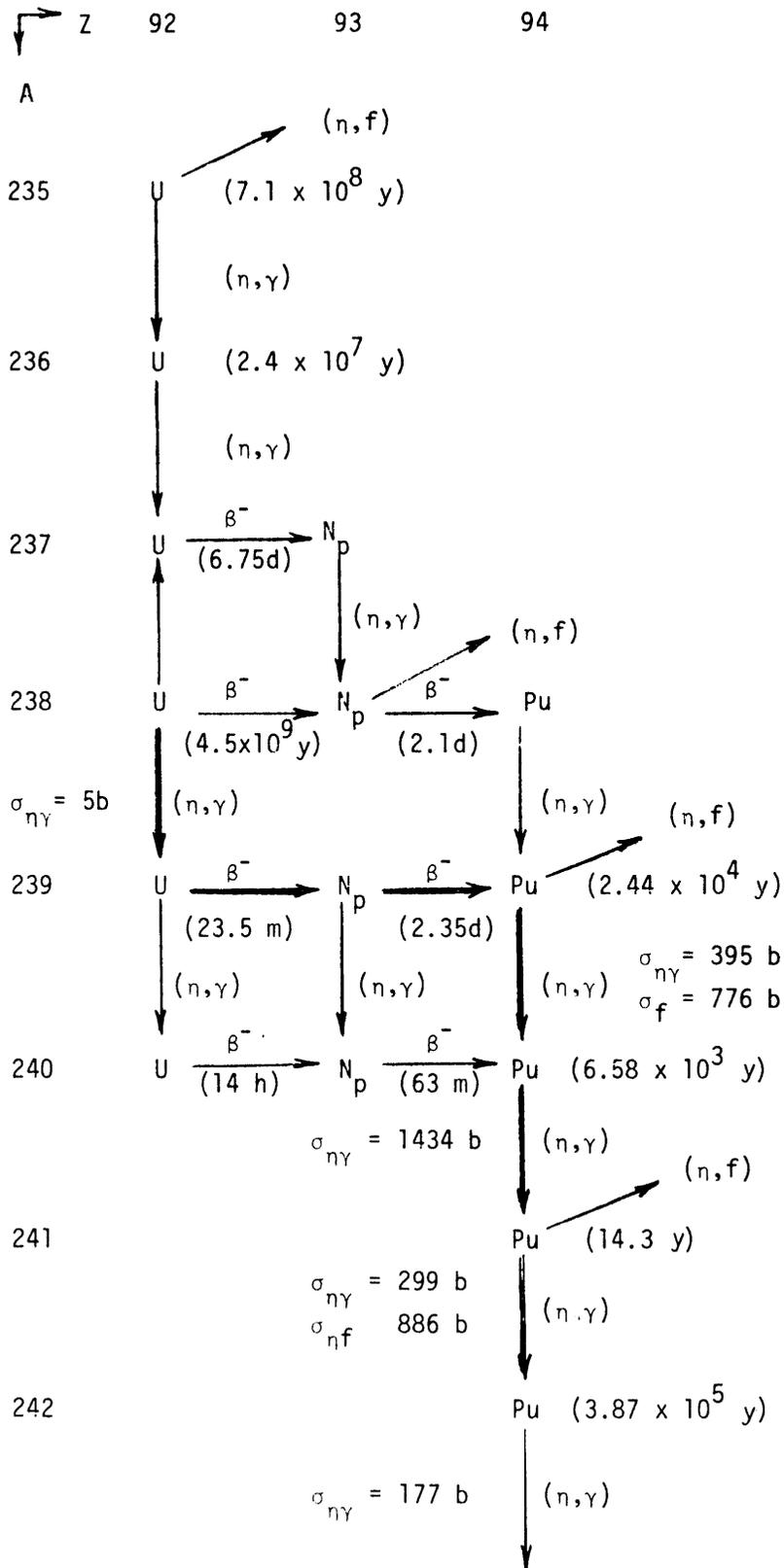
## 4. PRODUCTION OF PLUTONIUM

### 4.1 General Principles

Plutonium is produced by the absorption of neutrons in  $^{238}\text{U}$  to produce  $^{239}\text{U}$ , which decays with a 23.5-min half-life to  $^{239}\text{Np}$ , which in turn decays with a 2.35-d half-life to  $^{239}\text{Pu}$ . During the course of the irradiation, a number of other transuranium isotopes are produced, the most significant of which are  $^{240}\text{Pu}$ ,  $^{241}\text{Pu}$ , and  $^{242}\text{Pu}$ . The production and decay scheme is shown in Fig. 4.1.

Of the plutonium isotopes listed, only  $^{239}\text{Pu}$  and  $^{241}\text{Pu}$  have large fission cross-sections. They are thus the chief contributors to a plutonium chain reaction, with the former being by far the more important. Both  $^{240}\text{Pu}$  and  $^{242}\text{Pu}$ , which have small fission cross-sections, have an appreciable spontaneous fission decay mode, with the former generating about 900 neutron  $\text{g}^{-1}\text{s}^{-1}$  and the latter about 1600 neutron  $\text{g}^{-1}\text{s}^{-1}$ . Hence, their presence in significant quantities will result in some subcritical multiplication and the concurrent contamination of the plutonium with fission products. This causes an elevated radiation level from both neutrons and electromagnetic emissions, which makes subsequent processing and fabrication more difficult. Consequently, it is desirable to have a product which consists almost entirely of  $^{239}\text{Pu}$  and is relatively free of the heavier plutonium isotopes. This can be accomplished by using short irradiation times and frequent reprocessing. On the other hand, because less fissile plutonium is produced per unit weight of  $^{238}\text{U}$  in a short irradiation than in a long irradiation, the amount of target material which must be handled to produce a given quantity of fissile plutonium is larger in the former case than in the latter. Thus, some trade-offs must be made in order to determine the optimum irradiation cycle for a given facility.

The production characteristics are, to a very good first approximation, a function only of the fluence, i.e., the product of the flux



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Fig. 4.1. Plutonium production scheme (cross sections are effective values based on a typical research reactor spectrum. Heavy lines indicate principal production pathway.)

magnitude and the irradiation time. Curves showing the production per unit weight of target material and the fraction of nonfissile plutonium present are given in Figs. 4.2 and 4.3. The curves have been derived using the Origin code and a set of cross-sections developed for a typical water-moderated plate-type research reactor neutron spectrum; hence, they should be adequate to characterize the production rates to be expected provided the  $^{238}\text{U}$  targets are reasonably similar in configuration to the fuel. For situations where the neutron spectrum in the targets differs significantly from that found under the above conditions, the results would also be somewhat different.

Basically, then, what is required for plutonium production is a supply of neutrons and a suitable arrangement of  $^{238}\text{U}$  target material which is disposed in such a manner that it is capable of intercepting a large fraction of the neutrons not required to sustain the chain reaction.

#### 4.2 Reactor Power Requirements

The generation of 1 MW of thermal power requires  $3.121 \times 10^{16}$  fissions per second (200 MeV per fission). Since the average number of neutrons released by the fission of an atom of  $^{235}\text{U}$  is 2.47, the corresponding rate of neutron production is  $7.709 \times 10^{16}$  neutrons per megawatt-second.

Of these,  $3.12 \times 10^{16}$  neutrons are required to produce fission in order to sustain the chain reaction. In a typical water-moderated research reactor using highly enriched plate-type MTR fuel, only about 75% of the neutrons captured in the fuel elements and their associated moderator are captured by the  $^{235}\text{U}$ ; and only about 84% of these produce fissions. Hence, to maintain the chain reaction, a total of  $4.954 \times 10^{16}$  neutrons per second must be absorbed in the fuel. This

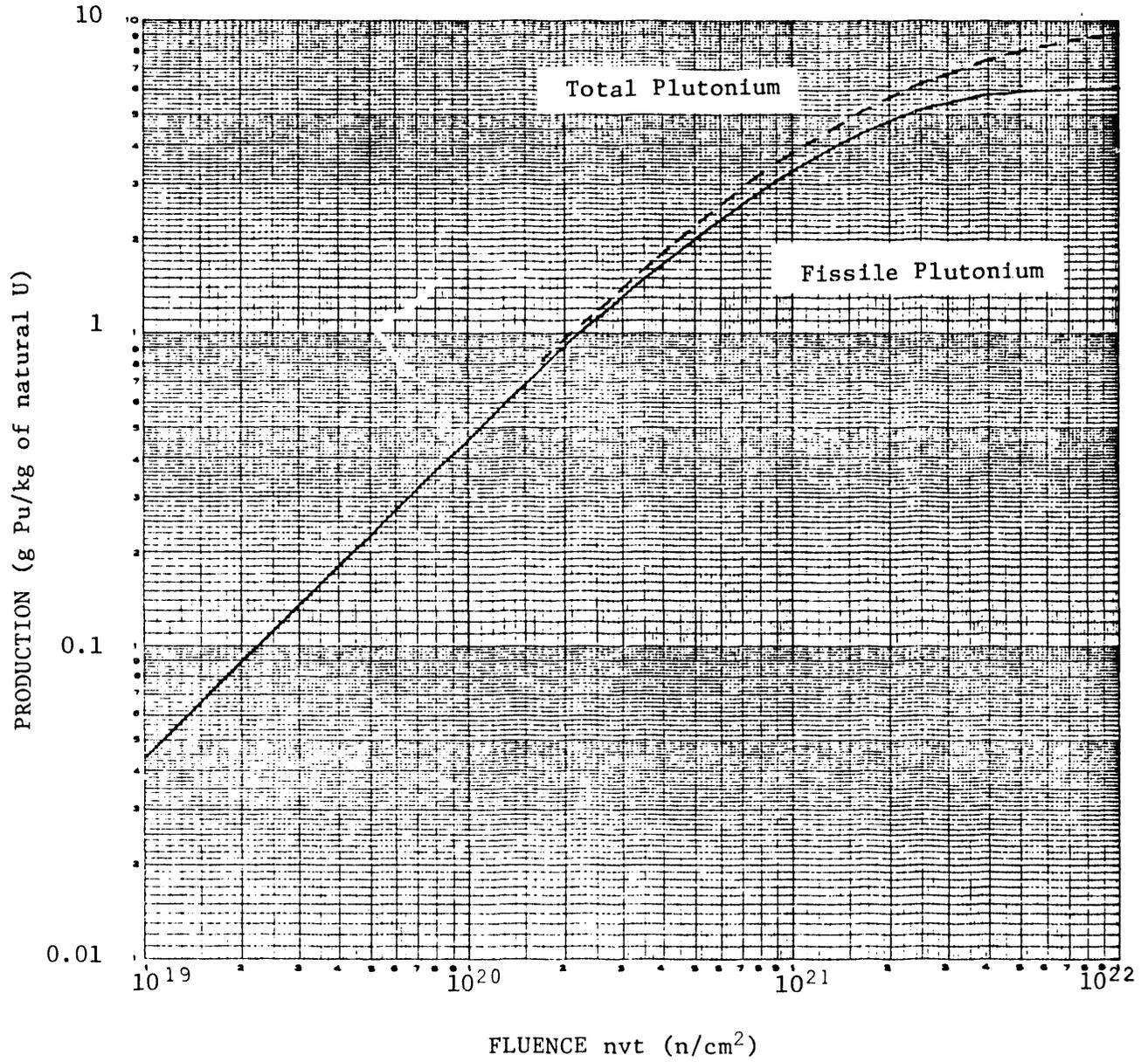


Fig. 4.2. Plutonium production as a function of fluence.

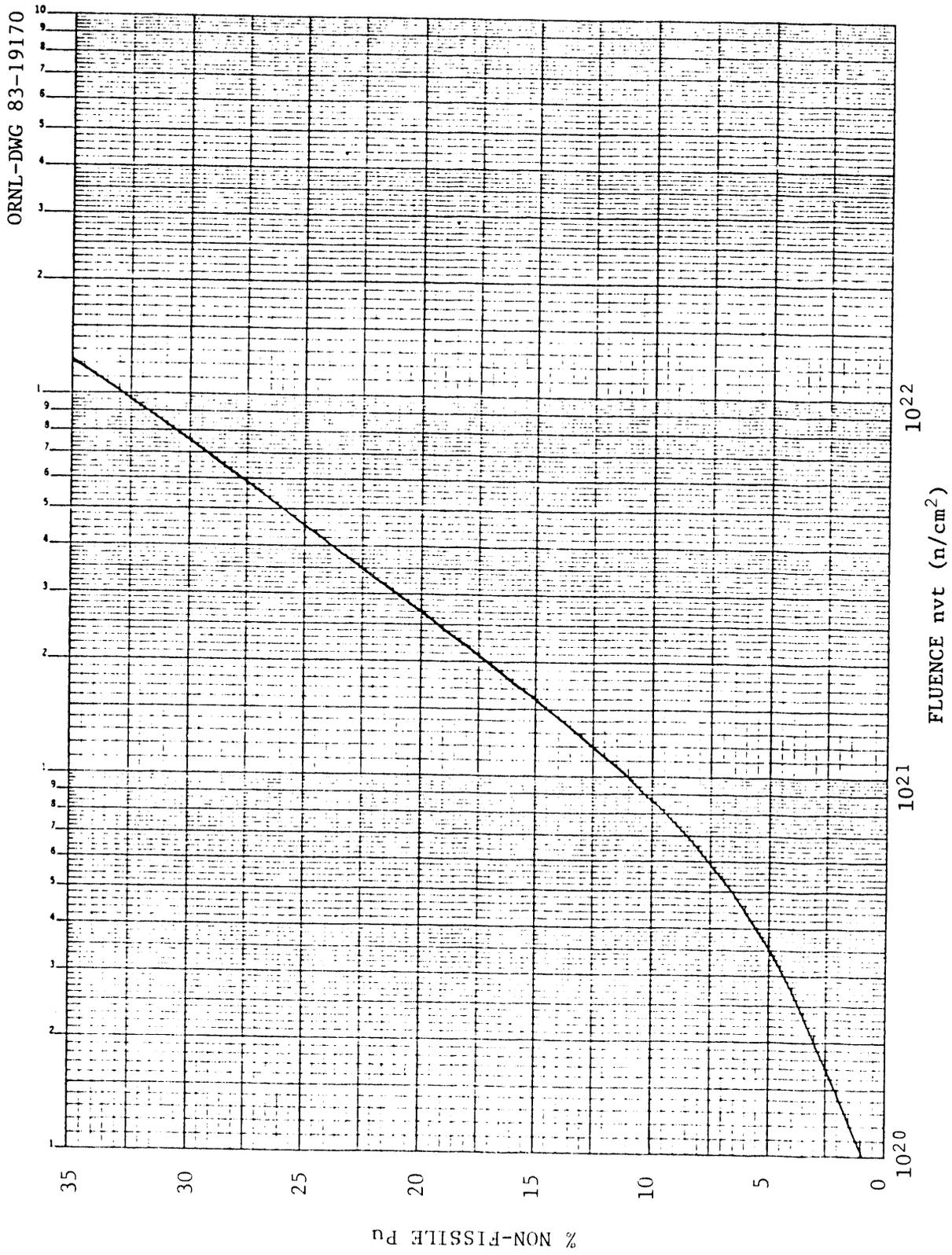


Fig. 4.3. Product purity vs fluence.

leaves  $2.755 \times 10^{16}$  neutrons per second (35.7%) available for other purposes. Of course, some of these are absorbed by the core structure and the control rods; and some of them leak out of the reactor so that the number of useful neutrons is less than this. For the moment, this fact will be neglected.

If it is now assumed that all of these spare neutrons are absorbed in  $^{238}\text{U}$ , it follows that the maximum production rate for plutonium is  $2.755 \times 10^{16}$  atoms per megawatt-second or 0.944 g per megawatt-day. This production rate has been calculated using, in addition to the assumption that all of the available neutrons are captured in  $^{238}\text{U}$  to form plutonium, the following assumptions: 1) there is no loss of plutonium due to fission or neutron capture which also implies that the presence of the plutonium formed in the  $^{238}\text{U}$  target has no effect on the probability of neutron absorption in  $^{238}\text{U}$ ; and 2) the only fissile material present is the  $^{235}\text{U}$  in the fuel and targets. These two assumptions would be true if the plutonium were removed as fast as it is formed; but as a practical matter, the targets must be allowed to remain in the neutron flux for a finite period of time so that there is a buildup of plutonium which is subject to both fission and radioactive capture. Moreover, the high cross-section plutonium isotopes compete with the  $^{238}\text{U}$  target for neutrons, thus reducing the production rate. On the other hand, fissions occurring within the target generate neutrons in the resonance energy range and thus increase the effective cross-section of the  $^{238}\text{U}$ .

Estimation of all of these effects requires a specific knowledge of the magnitude and energy spectrum of the neutron flux in the target region and will not be pursued further here, although the last two effects which are small and opposing can probably be neglected.

It may be concluded that in order to produce 8 kg per year of fissile plutonium an absolute minimum of about 8500 MWd are required. This translates into a 23-MW reactor operating for one year at a 100% plant factor with 100% use of all neutrons not required for fission.

As a practical matter, a certain amount of excess reactivity must be tied up in the control rods in order to permit operation to continue for a reasonable period of time and by absorption in fission product poisons, primarily  $^{135}\text{Xe}$ . Under the most optimistic circumstances, it is unlikely that more than two-thirds to one-half of the available neutrons could be utilized to produce plutonium; hence, about 15,000 MWd is actually required. Consequently, it is safe to conclude that in order to produce this amount of product in a year in a reactor operating at a realistic 85% plant factor, a minimum power level of about 40-50 MW is required. Moreover, the reactor would have to be dedicated almost entirely to the production of plutonium. Independent multigroup calculations of plutonium production in specific core configurations are in substantial agreement with this estimate.

#### 4.3 Target Considerations

Either natural uranium, which contains 0.72%  $^{235}\text{U}$ , or depleted uranium, which may contain 0.3 to 0.25%  $^{235}\text{U}$ , can be used as the target material.

Natural uranium has the advantage that it contains sufficient  $^{235}\text{U}$  so that a substantial amount of power, perhaps as much as 50% of the total, may be generated in the targets. Hence, using natural uranium targets will extend the life of the fuel elements. Because a substantial fraction of the power is generated in the targets, the heat load on the fuel elements themselves is reduced. This is desirable because some of the primary coolant is required to cool the targets, and this reduces the flow available to the fuel. Moreover, natural uranium is available from a wider variety of sources than is depleted uranium, which must be procured as a byproduct from one of the enrichment or reprocessing plants.

On the other hand, the power level and heat generation rate in natural uranium will be greater than that in targets made from

depleted uranium. Also, the fission product content in the irradiated targets will be higher by a factor of perhaps two or more. So far as plutonium production is concerned, there is virtually no difference between the two provided they are exposed to the same flux and spectrum; but, for a given reactor power level, the epithermal and fast flux in the natural targets should be higher than that in the depleted targets.

Three principles govern the design of the target elements: it is desirable to have each target contain as much  $^{238}\text{U}$  as possible, the dimensions of the target-bearing region must be such as to minimize self-shielding, and it is necessary that sufficient surface area be provided to permit removal of heat under the thermal-hydraulic conditions that prevail in the reactor being considered. The same heat transfer considerations that apply to reactor fuel also apply to the targets.

Of course, the exterior dimensions of the targets must be compatible with the dimensions of the reactor core and any blanket or other region where they are to be located. Moreover, it would seem desirable to have the targets appear similar to the fuel elements or to normally used experiment rigs so that they cannot be readily identified visually. Two kinds of target configurations suggest themselves: clad plates similar to those employed in the regular research reactor fuel elements or clad cylindrical pins much like those used in power reactors or in Triga fuel. The pins have the advantage that they can accommodate more  $^{238}\text{U}$  than can the plates. Moreover, if the denser materials such as metallic uranium or  $\text{UO}_2$  pellets are to be used, the pin geometry is probably mandatory. On the other hand, the surface area available for heat transfer will be substantially less than is the case for plates. For example, a 23-plate MTR-type fuel element having fuel core dimensions  $0.508 \text{ mm} \times 60.9 \text{ mm} \times 600 \text{ mm}$  has a total core volume of  $427 \text{ cm}^3$  and a heat transfer area of  $16,808 \text{ cm}^2$ . On the other hand, an array of 16 fuel pins having an active diameter

of 12.7 mm and a clad diameter of 13.7 mm of the same length has a total core volume of  $1416 \text{ cm}^3$ , but a heat transfer area of only  $4132 \text{ cm}^2$ .

An added disadvantage to the use of pin-type target elements is that upon close examination they can easily be distinguished from the regular plate-type fuel. A possible advantage is that these elements could probably be designed to permit removal of the pins in the reactor pool, with subsequent reuse of the other portions of the assembly. In this way, the pins could be handled individually without the necessity of moving entire target assemblies in and out of the reactor area. Similar considerations apply to the plate-type targets, but the design would be more complex.

The target elements could easily be produced using natural or depleted uranium by any of the techniques discussed in Section 3.3. The amount of uranium which can be loaded into the targets is, however, also limited by the considerations discussed in that section.

If metallic uranium is used, it is likely, as pointed out previously, that pin-type elements will be required. An array of 16 12.7-mm diam pins, 60 cm long, could contain as much as 23.2 kg of uranium. Similarly, arrays of 25 and 36 pins, 10 mm in diameter, could contain 22.5 kg and 32.6 kg of uranium, respectively. The heat transfer area per pin for the last two cases is about  $20.7 \text{ cm}^2$ . The use of  $\text{UO}_2$  pellets would reduce the loadings by about a factor of two.

Using soon-to-be-available powder metallurgy techniques, it is likely that uranium densities up to  $6\text{-}7 \text{ g/cm}^3$  can be achieved in plates. Using alloy, similar densities are possible, but whether this can be done successfully on a production basis is questionable. Another variable is the number and thickness of the fuel cores used. Thus, use of a 17-plate element having a core thickness of 1.73 mm would provide 2.5 times the core volume found in a 23-plate element which has a 0.51-mm-thick core. Of course, if the number of plates and their thickness in a target element differs from those in a normal fuel element, this could be observed by close examination. The heat transfer areas would also differ.

It is not possible to predict in advance what types of targets would be favored by an organization contemplating the production of fissile plutonium; but, as will become apparent, it is clear that in reactors of the power level under consideration, a very large amount of target material must be present. Thus, despite the fact that heat-transfer considerations and fabrication problems may prohibit it, it will be assumed that 25 kg of uranium is the maximum which can be contained in a pin-type element and 18 kg in a plate-type element. (The latter number is quite conservative, with 7 kg being more realistic.)

A very important point to note here is that an ordinary MTR-type fuel element weighs approximately 5.2 to 5.5 kg, of which 250 to 400 g is fuel. Hence, target elements containing large quantities of uranium can easily be distinguished from ordinary fuel by merely weighing them; and this can be done initially or underwater following irradiation. Even if 20% enriched uranium is substituted for the present highly enriched uranium in the fuel, the weight differences will be easily detectable. A reduction in the target weight to anything close to the normal fuel element weight would render them virtually useless for production purposes.

With respect to "fuel-like" targets, it must be realized that if these targets are placed within the reactor vessel they will partake of the primary coolant flow. Since the individual target elements are generating power at a rate considerably lower than the fuel or "driver" elements, they require less cooling than the fuel; however, the flow through them will divert coolant flow which normally passes through the fuel. It may, therefore, be necessary to restrict the flow through the targets by using orifices or some other means. If the targets are located outside the reactor vessel, some means of cooling them must be provided. This may be either by forced or natural convection. In the latter case it will be necessary to suppress the upward flow of water before it reaches the surface of the pool. This is to prevent  $^{16}\text{N}$ , which is formed by the  $^{16}\text{O}(n,p)^{16}\text{N}$  reaction in the cooling water, from reaching the surface and creating a radiation hazard.

Although it is possible to position specially designed targets in experimental facilities such as beam holes, the space available is so limited that at the production rates to be expected they would generate only a small fraction of the plutonium which could be obtained from targets in and near the reactor core.

#### 4.4 Production Strategies

The most obvious and probably the only viable scheme for producing large quantities of plutonium in a research reactor is to position as much  $^{238}\text{U}$  target material as possible in and around the reactor core within the reactor vessel and to irradiate for an appropriate period of time. In order to obtain accurate production information, it is necessary to perform a multigroup calculation on each specific fuel and target configuration. Nevertheless, using the data in Figs. 4.2 and 4.3 together with some reasonable assumptions concerning the magnitude of the neutron flux in the target elements, it is possible to come to some general conclusions concerning expected production rates.

It can be seen from Fig. 4.2 that the maximum possible production is about 6.2 g of fissile plutonium per kilogram of target\* and that this occurs at a fluence of about  $8 \times 10^{21}$  nvt. On the other hand, at a fluence of  $5 \times 10^{20}$  nvt, 2 g of fissile plutonium are produced per kilogram of target so that by the consecutive irradiation of 16 1-kg targets to  $5 \times 10^{20}$  nvt it is possible to produce 32 g of plutonium in the same time. Moreover, at the lower fluence the nonfissile impurity is only about 6.5%, whereas in the first case it is over 30%. On the other hand, at the lower fluence more than three times as much target material is needed per gram of product.

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\*Although Fig. 4.2 is based on the use of natural uranium target, the results would be the same for depleted uranium provided the neutron spectrum was the same.

In general, higher production rates (and a higher quality product) are obtained by irradiation to lower fluences with frequent changes of the target elements. This procedure has the obvious disadvantage that it requires more target material than do longer irradiations. The effects of irradiation time can best be illustrated by an example.

Consider a 50-MW research reactor which has space for 100 components in its grid. Assume that 30 of these components will be fuel "drivers" or shim rods and that the other 70 are target elements, each containing 25 kg of natural uranium. Of the 70 targets, 18 are located in the reactor core and the other 52 are arranged in a blanket around the core. The configuration is shown in Fig. 4.4. To perform the calculation properly, the flux in each target element should be known, but for illustrative purposes it will be assumed that the average flux in the in-core targets is  $7.75 \times 10^{13}$  and that in the blanket targets it is  $2.5 \times 10^{13}$ .

To achieve a given fluence,  $F$ , at a flux,  $\phi$ , the time required is  $F/\phi$  seconds. Consequently, the number of cycles per year is  $3.15 \times 10^7 \phi/F$ . If  $M$  is the weight of uranium per target, then the amount of plutonium produced per target per year is  $3.15 \times 10^7 \phi M P(f)/F$ , where  $P(f)$  is the production in grams of plutonium/kilogram of uranium given in Fig. 4.2. The annual number of targets required per target position is the same as the annual number of cycles and the annual target weight requirement is just  $3.15 \times 10^7 \phi M/F$ . Summing over all targets gives the total annual plutonium production. The results for the specific example cited above are given in Table 4.1.

A further elaboration of this example is instructive. Assume that the production strategy is to produce 2 g of fissile plutonium per kilogram of target. This requires a fluence of about  $5 \times 10^{20}$  n/cm<sup>2</sup>. Using the same fluxes as before and assuming an 85% plant factor, the targets in the core must be replaced about 4.15 times per year, and those in the blanket 1.31 times per year. Let  $M$  be the weight of <sup>238</sup>U

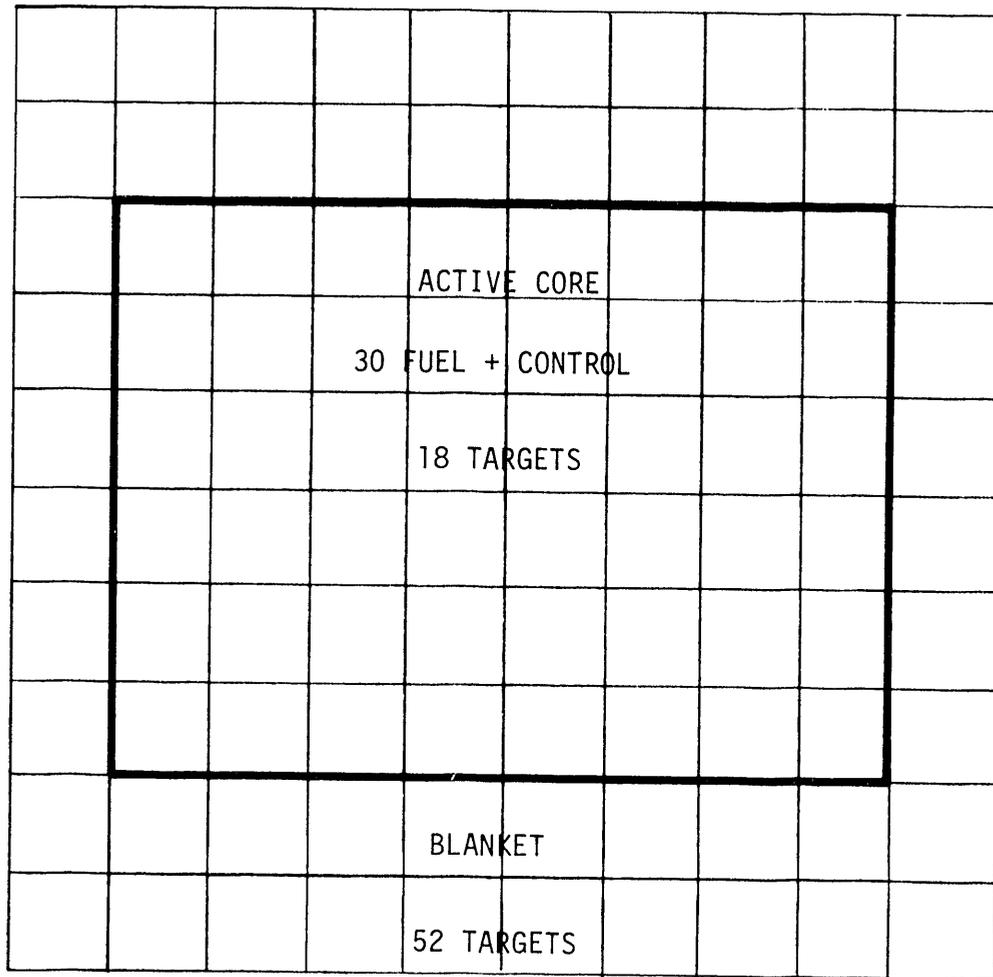


Fig. 4.4. Typical production core.

Table 4.1 Plutonium production in a typical research reactor

Fluence n/cm <sup>2</sup>	Based on 100% plant factor				Adjusted for 85% plant factor					
	Target location	Cycles per year	Targets per year	Weight U per year, tonnes	Fissile Pu per year, kg	Nonfissile Pu, %	g fissile Pu per kg U	Total fissile Pu per year, kg	Total Targets per year	Total weight U per year, tonnes
10 <sup>20</sup>	In core	24.41	439.4	10.985	5.273	1	0.48	8.659	722	18.04
	blanket	7.875	409.5	10.238	4.910					
2 x 10 <sup>20</sup>	In core	12.21	219.7	5.493	4.943	3	0.90	8.118	361	9.02
	blanket	3.938	204.8	5.119	4.607					
5 x 10 <sup>20</sup>	In core	4.88	87.89	2.197	4.284	7.5	1.95	7.037	145	3.61
	blanket	1.575	81.90	2.048	3.993					
10 <sup>21</sup>	In core	2.44	43.95	1.099	3.625	11	3.30	5.950	72	1.805
	blanket	0.788	40.95	1.024	3.375					
2 x 10 <sup>21</sup>	In core	1.22	21.97	0.549	2.609	17	4.75	4.284	36	0.902
	blanket	0.394	20.48	0.512	2.431					
5 x 10 <sup>21</sup>	In core	0.488	8.79	0.220	1.318	26	6.00	2.165	14.5	0.361
	blanket	0.158	8.19	0.205	1.229					
10 <sup>22</sup>	In core	0.244	4.39	0.110	0.659	33	6.01	1.082	7.2	0.180
	blanket	0.0788	4.10	0.102	0.614					
2 x 10 <sup>22</sup>	In core	0.122	2.20	0.055	0.319	40	5.82	0.524	3.7	0.090
	blanket	0.0394	2.05	0.051	0.297					

Power level - 50 MW

Flux in 18 in-core targets 7.75 x 10<sup>13</sup>

Flux in 52 blanket targets 2.5 x 10<sup>13</sup>

25 kg natural U per target

in the targets (kilograms). Then, to produce 8 kg of fissile plutonium we must have

$$(18 \times 4.15 \times 2M) + (52 \times 1.31 \times 2M) = 8000$$

so that that weight of the fertile material must be 28 kg per target and a total usage of about four metric tons annually.

If a fluence of only  $2 \times 10^{20}$  n/cm<sup>2</sup> is used, then only 0.9 g/kg is produced. The in-core targets must be cycled 10.4 times per year and the blanket targets 3.35 times per year. To produce 8 kg of fissile plutonium, we have

$$(18 \times 10.4 \times 0.9M) + (52 \times 3.35 \times 0.9M) = 8000$$

so that  $M = 24.6$  kg per target, but the total annual <sup>238</sup>U usage increases to about 9 tonnes.

In general, for a given rate of annual production the weight of target material which must be present in the reactor decreases as the fluence employed (and, hence, the production per kilogram) decreases; but the total weight of target material required annually increases dramatically. This constitutes an important practical limitation on the production rates which can be achieved. It has been tacitly assumed above that the targets can be fabricated using uranium metal; but, as discussed in Section 3.3, there is considerable doubt whether or not this can be done successfully. If, as is likely the case, conventional fuel fabrication techniques must be used, the capacity of the targets to contain fertile material will be reduced by a factor of at least two. Under these conditions, a power level of about 100 MW would be required in order to produce a significant quantity of fissile plutonium. Nevertheless, the possibility exists that some method for utilizing metal in the targets can be devised so that the use of the heavy targets cannot be ruled out entirely.

It is here reemphasized that the foregoing is an example used to illustrate a method for estimating production rates. Its results should not be construed to be representative of any given reactor, although their order of magnitude is probably correct in general. To get reliable estimates, configurations and neutronic parameters specific to the reactor under consideration must be used.

While it is impossible to predict what strategy would be followed in any particular situation, certain advantages and disadvantages are immediately obvious. To begin with, to attain the highest production rates requires the handling of a very large amount of target material. Also, it can be seen from Table 4.1 that the production of 8.66 kg of plutonium requires five times as much target material as that required to produce 7.04 kg, an increase in production of only about 23%. It would seem that prudent management would dictate use of the higher fluence value and lower production rate, not only from an operating standpoint, but also because the reduced amount of target handling would be easier to conceal. It is true that higher fluences produce a lower purity product, but as long as the value does not exceed about  $10^{21}$  this should not be a serious disadvantage. As a practical matter, then, for this particular example one would expect a production rate in the neighborhood of 7 kg of fissile plutonium per year, which implies that 140 to 150 targets must be handled annually.

During normal operation without targets, a 50-MW reactor with an 85% plant factor will generate 15,512 MWd per year. Since 1 MWd corresponds to the consumption of about 1.24 g of  $^{235}\text{U}$ , the total  $^{235}\text{U}$  annual requirement is about 19.2 kg of  $^{235}\text{U}$ . If this is supplied by fuel elements initially containing 285 g of  $^{235}\text{U}$  and which are burned down to an average of 50%, about 135 fuel elements are required annually. For the case where plutonium is being produced, about 145 target elements must be handled per year. In the example, there are always 70 natural uranium target elements in the core; and these contain initially a total of 12.6 kg of  $^{235}\text{U}$ , which provides about 30% of the total reactor power. Thus, for the core containing targets, only

about 35 MW is generated in the fuel so that the fuel requirement is reduced to about 95 elements annually; and the total number of core components which must be handled is 240, compared to 135 in the normal core.

If depleted uranium is used as the target material, the power generated in the targets will be less, by a factor of two or three, so that change in fuel element usage will be less pronounced; however, the total number of components handled will be greater.

It must be emphasized again that the foregoing is only an example and that each reactor system must be examined using values of the various parameters which are appropriate for that system. However, there are two important general conclusions which can be drawn: 1) in a plutonium-producing reactor, the number of core components which must be handled will be substantially greater than the number which must be handled in one not producing plutonium; and 2) for the same amount of generated energy, the plutonium producer will require less fuel than the nonproducer.

Changes in power will, of course, affect the production rate. If a reactor similar to that described in the example were to operate at 100 MW, 8 kg of fissile plutonium could be produced annually by irradiating 15 in-core targets and 26 blanket targets to a fluence of  $10^{21}$ . This would require that only about 115 targets or 2.9 tonnes of uranium be handled annually. In this case, the fuel consumption would be about 80% of that in the nonproduction reactor and the total number of core components handled would be about 40% greater.\*

In the foregoing discussion, it was tacitly assumed to be physically possible to locate 70 target elements in and around the reactor core in such a way that they are exposed to the fluxes specified. This may not always be true. For example, the ORR has a  $9 \times 7$  grid

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\*It is here assumed that 200 g of  $^{235}\text{U}$  can be consumed per fuel element.

which allows space for a total of only 63 core components. Additional targets could be placed outside the lattice to increase the total number to perhaps 85; however, this would require significant design modifications to provide cooling. Moreover, a blanket consisting of two rows of targets is about the maximum practical because the flux in a third outer row would be too low to produce much plutonium.

It is worth noting that targets located within the active lattice are more effective than those located on its periphery. In the example, the peripheral targets contained nearly 75% of the target material but contributed less than 50% of the production. Thus, configurations in which targets and fuel are interspersed would be favored. It may be possible to enhance the flux in the target elements by scattering them throughout a large active lattice rather than by disposing of most of them in a blanket. The effect of this and other such options can only be estimated by making calculations specific to the configurations under consideration.

It is clear, however, that it is possible to produce a significant quantity of fissile plutonium in a research reactor operating at a power level of 40 to 50 MW or above, that the potential production rate is proportional to the power level, but that it is limited by the physical size of the reactor core and its surroundings. To produce this much depends upon the ability to utilize uranium metal as the target material, and it is questionable whether or not this can be done successfully. If it is necessary to use some other uranium-bearing material as the target, then either higher powers, a great many more targets, or a combination of the two would be required. For example,  $\text{UO}_2$  pellets would require a power level of about 100 MW or else about twice the number of targets, whereas the  $\text{U}_3\text{Si}_2$  cermet would demand power levels of the order of 200 MW, or four times the number of targets.

## 5. ADDITIONAL CONSIDERATIONS

### 5.1 Ancillary Facilities

It was concluded in the previous section that it is possible to produce significant quantities of fissile plutonium in a research reactor provided that the power level is high enough to generate an adequate number of neutrons and that it is physically possible to expose sufficient  $^{238}\text{U}$  to them.

There are, however, two other ingredients which must be present. The first of these is the capability to fabricate or otherwise acquire a supply of target elements, and the second is the existence of a processing plant capable of separating the product from several tonnes of uranium which are highly contaminated with fission products. In addition, provision must be made for the temporary storage of irradiated targets which have to be shielded and, initially at least, cooled.

Although the existence of target fabrication and processing facilities are essential to the production of plutonium, a discussion of their characteristics is beyond the scope of this study. It should be noted, however, that they need not be located at the reactor site. Targets could be fabricated elsewhere and shipped to the site, and the irradiated material could be transferred to a remote processing site. The latter would require the use of massive shipping containers for shielding. Storage and handling of the targets at the reactor, whether or not fabrication and processing are done off-site, are potentially important observables which could provide evidence of plutonium production and will be treated in the next section.

### 5.2 Target Handling and Storage

Unirradiated uranium targets, like unirradiated fuel elements, can be stored and handled unshielded with the only safety concern being criticality. Thus, they can be stored in an area remote from the

reactor and transported to it quite easily. On the other hand, once irradiated they must be provided with biological shielding and sufficient cooling to prevent damage from overheating.

Both the heat generation rate and the radiation level depend upon the irradiation time, operating power level, and decay time following the irradiation. Figure 3.5 shows the heat generation rate to be expected in a typical target as a function of time after reactor shutdown. This curve is based on an irradiation time of eight weeks at constant power. For shorter irradiation periods, the fission product power will fall off more rapidly with time, less rapidly for longer irradiations; however, this curve should serve to give an idea of the time behavior of the heat generation rate. More specific estimates can be obtained by using the data in ANSI Standard 5.1.

A very crude estimate of the unshielded radiation level can be obtained by multiplying the current power level in watts by the factor 3.3. This gives an estimate of the radiation level in roentgens per hour at the midplane of the target element 3 m from the vertical centerline. Thus, for a target which had been irradiated for eight weeks at 250 kW and allowed to decay for ten days, the heat generation rate would be 350 W and the unshielded radiation level would be about 1150 R/h at 3 m.

While from a heat transfer standpoint it may be possible to remove the targets from the reactor pool shortly after shutdown and rely on free convection in air to remove the heat, it is clear that they cannot be approached unshielded without the risk of serious injury or death. Consequently, it is highly unlikely that irradiated targets will be removed from the reactor pool in other than shielded containers or through some sort of opening in the pool wall which leads to a canal or other storage area. In any case, the targets will have to be permitted to decay for some period of time prior to processing; and this is most conveniently done underwater to provide cooling as well as shielding. If the storage takes place in the reactor or adjacent fuel storage pool, the number of core elements so stored will

increase over that to be expected if only fuel were present. If the elements are removed to some remote storage area shortly after discharge from the reactor, merely accounting for the number of components stored at the reactor would not be conclusive. In either case, if a large number of uranium target elements are being handled, there will be a concurrent increase in the number of core components being transferred in and out of the reactor.

### 5.3 Reactor Power Increases

For the reactors under consideration, by far the most important limitation on power level is the ability to remove heat. So far as the fuel elements are concerned, they would, in nearly all cases, withstand a considerably higher specific fission rate provided the temperature could be controlled. This is also true of the smaller pool-type reactors. Basically, sufficient cooling implies that at the maximum heat flux produced by the operating power level, the primary coolant velocity will be adequate to develop a heat transfer coefficient high enough to maintain the fuel surface temperature, at most, a few degrees above the saturation temperature, and that sufficient primary coolant be forced through the reactor at a low enough inlet temperature so that the maximum temperature of the fuel remains below the point where softening and loss of structural integrity can occur. Moreover, the capacity of the secondary system which receives heat from the primary system and wastes it to the environment must be adequate to handle the heat load.

In order to increase the power level of an operating research reactor, there are a number of variables which can be manipulated. In the case of tank-type reactors which have a closed primary-coolant system, an increase in the operating pressure will produce a corresponding increase in the saturation temperature, thus allowing a higher surface temperature. A pressure increase may be accomplished by using a positive displacement pump to force water into the system, by

using a steam pressurizer in which a bubble of high-pressure steam located in a vessel attached to the primary system and controlled by electric heaters is used to pressurize the system, by altering the primary pump characteristics to increase the discharge pressure, or by increasing the hydrostatic head on the system. In general, pressure increases are limited because of the design of the reactor vessels which usually are not capable of withstanding very high pressures.

An increase in primary coolant flow will increase the velocity through the coolant channels in a reactor of fixed configuration. This causes an increase in the heat transfer coefficient, which varies as the 0.8 power of the velocity, and results in a higher permissible heat flux at the same surface temperature. Higher flow also reduces the coolant temperature rise for a given power level. It does, however, increase the pressure loss in the system. Lowering the primary coolant inlet temperature will also increase the permissible heat flux.

The heat flux may be lowered by increasing the heat-transfer area. The most obvious way to accomplish this is to increase the number of plates in the fuel elements. Thus, substituting 23-plate elements for 19-plate elements would increase the heat transfer area per element by about 20%. In this connection it should be recalled that in a reactor heavily loaded with natural uranium targets a substantial portion of the power is developed in the targets; and, since there is an increase in the number of core elements, the heat-transfer area is also increased. Thus, only an increase in primary and secondary coolant flow may be required to support a modest increase in power.

For reactors already operating near their thermal capacity, one or more of the foregoing changes will be needed if the power is to be increased significantly. On the other hand, many reactors — particularly older ones — were overdesigned from a thermal hydraulic standpoint; and their power can be increased without any substantial changes. In most cases, this has already been done where possible.

Regardless of how the power level increase is accomplished, the most reliable method of determining the actual power is by means of a calorimetric balance on the primary and secondary cooling systems, utilizing the coolant mass flow rates together with the inlet and exit temperatures. Of course, more  $^{235}\text{U}$  will be consumed per unit time at a higher than at a lower power. But, because in a plutonium-producing reactor some fraction of the power is generated in the targets, the rate of consumption of  $^{235}\text{U}$  in the fuel itself may not be a reliable indication of the total power level.

With the exception of BR-2 which is already operating at 100 MW and Tammuz-I which was overdesigned to the extent that it could probably operate at 70 to 75 MW (although modification of the secondary cooling system might be required), none of the reactors under consideration could be operated at power levels much higher than 50 to 60 MW without major modifications to the cooling system to provide additional heat removal capacity. Some of the lower-powered research reactors could be upgraded to this level. The requirements for this would have to be determined on a case-by-case basis.

#### 5.4 Core and Target Configurations

In order to produce a significant quantity of fissile plutonium, it must be possible to dispose a large quantity of  $^{238}\text{U}$  in and near the reactor core. The number of in-core positions which are available for fuel, experiments, reflector pieces, control rods, or targets varies with each reactor but generally ranges from about 55 to about 100, of which approximately 30 to 35 positions are required for fuel and control rods. In some cases as many as 50 to 100 positions normally occupied by reflector pieces or experiments are located outside the core, but near enough to it to provide useful positions for target irradiations. It may be concluded that there is probably enough available space in and around each of the reactors to accommodate an

adequate inventory of target material, provided metal targets and the proper fuel configuration are used. An accurate estimate of the production rates to be expected and the annual amount of target material requirements can only be obtained by a calculation based on the particular configuration under consideration.

Use of the core grid positions as target locations presents no great difficulty since the targets can be inserted and removed in the same manner as ordinary fuel elements, which are generally handled through a hatch in the reactor vessel especially designed to allow easy access to the fuel area. If positions designed as more or less permanent locations for reflector pieces are to be used for target locations, it may be that special tools and handling procedures will be required to cycle the targets in and out of the reactor. In extreme situations, modification of the fuel access hatch may be needed. In any case, the reflector pieces which originally occupied those positions must be removed and disposed of.

## 6. OBSERVABLES INDICATIVE OF POSSIBLE PLUTONIUM PRODUCTION

### 6.1 Background Information

Unfortunately, at present there exists no specific method to conclusively determine whether or not a given research reactor is being used to produce undeclared plutonium. Although there are distinct differences in the neutronic behavior of plutonium fission and that of  $^{235}\text{U}$  fission and between the characteristics of reactors containing only a small amount of  $^{238}\text{U}$  and those containing a very large amount, as yet no instrumentation has been developed to exploit these differences. Therefore, in order to detect such production, it is necessary to take a diagnostic approach and search for those symptoms which must be present if plutonium is being produced.

In adopting this approach, a number of factors should be kept in mind. To begin with, it should be obvious that any attempt to produce

undeclared plutonium would be accompanied by a determined effort to conceal the fact that it is being done. Since it can be assumed that the reactor operator is just as aware of the significance of the observables which are symptomatic of plutonium production as is the inspector, he can be expected to make every effort to prevent the inspector from discovering them. Denial of access to these observables may itself be a meaningful symptom.

Secondly, it must be realized that it will be far easier to conceal the annual production of a small quantity of plutonium — a kilogram or two — than that of a significant quantity. In the latter case, the reactor will have to be virtually dedicated to plutonium production with a consequent substantial alteration in its operating pattern, whereas in the former any changes would be less obvious. Similarly, a "one-shot" production campaign in which the reactor is used only long enough to produce one significant quantity and then returned to normal operation would be easier to conceal than conversion to plutonium production and operation in that mode for a protracted period of time.

The observation of one or more symptoms that plutonium is being produced does not necessarily mean that a significant quantity is being produced; however, if the reactor is capable of such production, it may be presumed that it is a possibility.

Finally, it must be realized that changes in the reactor operating regime which arise for perfectly legitimate causes are not unusual. Although operating cycles are generally more or less repetitive, changes in experimental programs, necessary repairs, planned improvements and upgrades, improvements in fuel design, and many other contingencies result in altered operating cycles and procedures. It is worth noting, however, that such changes are normally undertaken to improve the efficiency of the process, either from an operational or an economic standpoint. Thus, any change which obviously decreases efficiency should be viewed with suspicion.

In order to arrive at a meaningful diagnosis from observation of the various symptoms which indicate "off-normal" operation and possible plutonium production, it is necessary that the inspector be sufficiently familiar with the operation of the reactor for which he is responsible that he can readily recognize the difference between "normal" and off-normal" operation. It is for this reason that the earlier sections of this report have been devoted to a generic discussion of research reactor operation and plutonium production. The inspector should be cognizant of the specific values of the relevant parameters of the reactor and should be thoroughly familiar with its normal operating characteristics.

It is also necessary that the inspector have access, either through direct observation or surveillance devices, to the observables which may reveal the symptoms. If it were possible for the inspector to directly examine all of the components which enter or leave the vicinity of the reactor core, then there is no doubt that the irradiation of  $^{238}\text{U}$  targets could be readily detected; but, if he has no access at all, detection becomes difficult, if not impossible. The extent to which the inspector has access to the observables is a matter which is negotiated between the IAEA and the state in which the reactor is located and will not be addressed here. Nevertheless, it is obvious that, in a situation where undeclared plutonium production is occurring, efforts would be made to curtail or deny such access. It must be appreciated, however, that there may be perfectly justifiable reasons associated with national or industrial security for withholding certain types of information.

The more obvious symptoms indicative of plutonium production have been developed in the previous sections. They will be summarized and methods of observing them discussed in the remainder of this section. These will be treated in a generic fashion because specific quantitative results for any given reactor will depend to a considerable extent on the physical configuration of that reactor.

## 6.2 Direct Methods of Detection

The most compelling evidence that plutonium is being produced or that its production is being contemplated is the presence at the reactor of irradiated  $^{238}\text{U}$  targets or of fresh, unirradiated targets containing the fertile material. Even though they may have an external appearance similar to normal fuel elements, they can easily be identified by an inspector if he has access to them.

The simplest procedure is to weigh them and compare the weight with that of the regular reactor fuel. It was shown in Section 4 that, in order to produce a significant quantity of plutonium, the reactor must be heavily loaded with  $^{238}\text{U}$  so that the target elements must contain a large quantity of uranium and will weigh a great deal more than the fuel elements. The weight of a regular fuel element can be obtained by weighing it, obtaining the weight from the shipper's invoice, or calculating it using the dimensions of the element.

By use of suitable interrogation devices which detect their radiation, it should be possible to distinguish between an element loaded with natural or depleted uranium and one fueled with enriched uranium, whether or not it has been irradiated.

Visual examination of the interior of a core component will reveal if it has a configuration identical to a fuel element or whether pins or thick plates — which admit higher uranium loadings — are being used. Certain legitimate experiments which are designed to test power-reactor fuels may utilize pins contained within a box or shroud shaped like a fuel element. While pins which contain fuel having a low  $^{235}\text{U}$  enrichment will indeed generate plutonium upon irradiation, the presence of a few such experimental devices does not imply that a significant quantity of plutonium is being produced, but it could serve to confuse the issue. Likewise, other experimental devices which contain fertile material but do not resemble the fuel could be

present; however, because of the large amount of  $^{238}\text{U}$  required, it is doubtful that such devices alone could be used to produce a significant quantity of plutonium.

Regrettably, it is quite possible that the inspector will not even know that the targets exist, let alone have access to them. If the operator intends to conceal the fact that he is producing plutonium, he would plan to store the fresh targets in a location inaccessible to the inspector and load the reactor at a time when the inspector is not present. The irradiated targets would be removed and also stored at a location also inaccessible to him. This would require either transport in shielded casks or through some sort of canal leading from the reactor pool to the concealed shielded storage area. Such a procedure can be countered by providing surveillance to detect it as discussed in Section 6.4.

Direct visual inspection of the reactor core would reveal whether it contains a normal complement of fuel, experiment rigs, control rods, and reflector pieces, or whether it is largely loaded with fuel-like components, many of which could be targets. The visual inspection would have to take place when the reactor is shut down just prior to the removal of fuel for reloading or just before startup after reloading. Of course, the operator could shut down, unload targets, and load a normal core before the inspector arrives and then reload with targets after he leaves; however, this would greatly increase the amount of fuel handling, which should be detectable by surveillance devices. Having an inspector on the site during each shutdown to inspect the core has the additional advantage that the extra fuel and target manipulations would extend the shutdowns, which occur every two to four weeks, and thus cut into the operating time.

If a significant amount of plutonium is to be produced, a great many more core components must be loaded and unloaded annually than is the case if only fuel is being handled. While it is not contemplated that an inspector be physically present at all times when core components are being inserted or removed, the number handled could be

observed by appropriate surveillance devices. An idea of the magnitude of the discrepancy to be expected can be obtained using the principles outlined in Section 4.

### 6.3 Indirect Methods of Detection

If the reactor is loaded with a large amount of natural uranium, a substantial fraction of the power will be generated in the targets. Hence, for a given reactor power, the rate of consumption of  $^{235}\text{U}$  in the fuel itself will decrease. Since the amount of  $^{235}\text{U}$  received at the plant as fresh fuel and the amount of  $^{235}\text{U}$  returned for reprocessing is presumably known, it is possible to determine the rate of fuel consumption provided the amount of  $^{235}\text{U}$  in the inventory of fresh and partially spent fuel is also known. A comparison of this with the power level could, in principle, indicate whether a significant amount of energy is being generated from sources other than fuel. Because of the almost universal practice of cycling partially spent fuel in and out of the reactor and the cooling time required before shipping, most of the reactors maintain an inventory of over a hundred spent and partially spent elements in storage so that this would require interrogation of a large number of components to determine the amount of  $^{235}\text{U}$  present and may not be practical. Over an extended period of time, however, a trend which indicates the consumption of less  $^{235}\text{U}$  than is required to produce the energy generated would probably be detected. This symptom could be negated by a surreptitious increase in power so that the fuel itself would actually operate at its original power level with the target elements supplying the increase.

There are a number of subtle changes in control rod action during startup which could occur due to the burnup and growth of  $^{135}\text{Xe}$  in a combination of fuel and target elements, as distinguished from fuel alone; however, an evaluation of these changes would require an intimate knowledge of the core history and, without the aid of sophisticated instrumentation, is probably beyond the capability of the inspector.

Where, as must be the case when a large amount of plutonium is to be produced, the reactor is virtually dedicated to this production, there will be a reduction in the availability of the reactor for experimental purposes. If most or all of the available core and peripheral positions are occupied by fertile targets, clearly they cannot be used for experiments, nor can the neutrons captured by the targets be furnished to the experiments. Consequently, the use of the reactor for the production of radioisotopes, materials testing, physical research, or any of the purposes for which it was originally designed will be diminished, probably by an observable factor.

#### 6.4 Surveillance

Two types of surveillance are desirable: first, continual observation of the reactor vessel and its immediate surroundings in order to monitor the number and kind of components entering and leaving the vessel and the storage pool; and, second, continuous monitoring of certain reactor parameters to verify the operating schedule and power level. To be trustworthy, the surveillance devices should be tamper-indicating and completely independent of the normal operating instrumentation. Methods for optical surveillance of the reactor are already available, and there exist interrogation devices, including ordinary scales, which could distinguish between natural and enriched uranium components. Methods to independently measure reactor power might require some development.

With respect to surveillance of the reactor vessel and pool, the desiderata are to be able to count all of the components which enter or leave the vicinity of the reactor and to examine each one, either by weighing or by the use of an interrogation device. To count or examine is probably not difficult, but to make the devices tamper-indicating or, better still, tamperproof against a determined adversary, may not be so easy.

It has been emphasized several times that one of the characteristics of significant plutonium production is the use and handling annually of considerably more core components than would be the case for normal operation. Hence, if the irradiated targets were merely stored along with the regular depleted and partially depleted fuel, the inspector could presumably count them and, together with a knowledge of the amount of fuel received and the amount shipped for reprocessing, he could determine if a discrepancy exists.

The best method of monitoring the reactor power level is to obtain a continuous readout of the mass flow rate of the primary coolant and its inlet and outlet temperature, thus obtaining a heat balance across the core or, failing that, obtaining values of the same parameters for the secondary cooling system. These signals are available in the reactor control room; but, if concealment is intended, the reliability of the signals received by the monitoring device could be questionable. It would be better if independent sensors were used. There are other indicators which could be employed to estimate the power level, such as the direct radiation at some point distant from the reactor, the intensity of the Cerenkov radiation, or the examination of the reactor noise signature. However, some development would be required to produce a tamper-indicating device capable of measuring power using these parameters.

In the absence of gross contamination, the main radioactive components in the primary cooling system of a reactor which utilizes aluminum-clad fuel and deionized water coolant are the 15.4-h  $^{24}\text{Na}$  and 7-s  $^{16}\text{N}$ . To a first approximation, the rate of production of both of these is proportional to the power level so that their concentration in the primary coolant exit is also proportional to the power level, provided the flow remains constant. Increases in the concentration of these isotopes could be indicative of a power increase. The concentrations of both can be altered by changing the total flow and, in the case of  $^{24}\text{Na}$ , by changing the fraction of primary coolant passing through the deionization system.

### 6.5 Structural and Engineering Changes

There are two types of structural and engineering changes which might be required to convert an existing research reactor for the production of significant quantities of plutonium.

The first of these are those changes necessary to permit an increase in reactor power. These changes, which were discussed to some extent in Section 5.3, deal primarily with modifications to the heat removal systems. To substantially increase the capacity of these systems, it is first necessary to enlarge the capability of the secondary system to waste the heat to the environment by increasing the number or capacity of the primary to secondary heat exchangers and by increasing the supply of secondary cooling water, lowering its temperature, or both. This last usually means an increase in the number of cooling tower cells or a larger supply of cooling water if a natural source such as a river, lake, or sea is used as the source. Changes of this nature are not trivial and require considerable time to bring about. They should be quite easy to detect.

Possible alterations to the primary system include increased flow which would require modifications to or possibly replacement of the primary pumps, the installation of pressurizing equipment, and perhaps modification of the fuel to provide greater heat transfer area.

In any event, if a substantial increase in pumping demand is put into effect, there should be a concurrent increase in the use of electric power. Of course, if the system was initially overdesigned from a thermal-hydraulic standpoint, no significant changes in the design of the system would be needed in order to increase the power to its permitted maximum value and only if previously idle pumping equipment were utilized to accomplish the power increase would there be an increase in electricity usage.

The foregoing is directed primarily at tank-type reactors already operating in the 10- to 20-MW range. In order to increase the power of the small (2 to 5 MW) open-pool-type reactors to power levels

anywhere near that required to produce a significant quantity of plutonium, the modifications to the cooling system would be so extensive that it would probably be impractical to try to conceal them. It might be possible to achieve power levels of 12 to 15 MW in these reactors, but this would entail a major upgrade of the cooling system.

Modifications of the second type are those which may be necessary to permit the loading of the required amount of  $^{238}\text{U}$  into the reactor. For those reactors in which the core lattice is large enough to accommodate both the requisite amount of target material and sufficient fuel to permit operation, no significant changes beyond possible orificing of the targets to provide proper flow distribution will be necessary. A few of these were discussed in Section 5.4. In some cases where the lattice is small but where more or less permanent reflector material is located within the reactor vessel, it may be necessary to remove the reflector so that it can be replaced with fertile targets. Whether or not this will require a change to the existing reactor grid plate will depend on the specific reactor considered. However, unless the grid structure was originally sufficiently oversized so that it can handle the additional weight of the targets, modifications will be required to strengthen it. Changes in the hatches through which fuel is handled may be needed if the new target locations are inaccessible through the present openings. Specially designed handling tools could also be needed.

In some reactors it may be possible to place target material outside the reactor vessel. The most likely locations are on the "poolside faces" of those reactors which have them and around the "chimney" of reactors of the Tammuz-I or Osiris type.

Properly designed targets in these locations can probably be cooled by free convection; however, some method must be devised to prevent the  $^{16}\text{N}$  produced in the cooling water from diffusing to the top of the reactor pool where it would present a radiation hazard. Targets in these locations should be visible but could be disguised to resemble experiments normally expected to be there.

The location of experimental facilities such as thermal columns and beam ports could be used to contain ex-core fertile material with the thermal column locations, because of their size, probably being the more useful. To use these locations, special target configurations would have to be designed and fabricated and adequate cooling provided. When irradiated, procedures for shielding and perhaps cooling the targets during their withdrawal and transfer to storage would have to be devised. Because the potential production rate in these facilities alone is small and because of the elaborate provisions required to accomplish it, irradiation of target materials in ex-core experimental facilities is probably not of serious concern relative to the production of a significant quantity of plutonium.

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In order for the inspector to become sufficiently familiar with the reactor under his jurisdiction, he should study as much of the open literature concerning it as he can. The two best sources of information are the Safety Analysis Reports which are usually prepared during the design and construction phases and the periodic operating summaries normally issued by the operating organizations. While these documents are generally not restricted, their distribution may be quite limited so that they are sometimes difficult to obtain.

A variety of information published in the open literature may be found by searching abstracts such as Nuclear Science Abstracts and its successor, Energy Research Abstracts, both published by the Technical Information Service of the U.S. Department of Energy.

For convenience, a number of key references which describe the six reactors listed in this report are given below.

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\*These references all refer to the Osiris reactor, which is quite similar to Tamuz-I.

†This description applies to the USSR reactor MR, which is quite similar to María.

## INTERNAL DISTRIBUTION

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