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RADIOISOTOPE PRODUCTION IN POWER REACTORS

A. F. Rupp
J. A. Cox
F. T. Binford

OAK RIDGE NATIONAL LABORATORY
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ORNL-3792

Contract No. W-7405-eng-26

ISOTOPES DEVELOPMENT CENTER

RADIOISOTOPE PRODUCTION IN POWER REACTORS

A. F. Rupp, Directors Division
J. A. Cox, Operations Division
F. T. Binford, Operations Division

MAY 1965

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
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RADIOISOTOPE PRODUCTION IN POWER REACTORS

A. F. Rupp J. A. Cox F. T. Binford

ABSTRACT

The possibility of producing large amounts of radioisotopes in 18 power reactors, together with some of the problems of establishing such a program, was investigated. Because of limitations of flux, irradiation time, and possible uses, seven radioisotopes – ^{14}C, ^{60}Co, ^{170}Tm, ^{210}Po, ^{238}Pu, ^{242}Cm, and ^{244}Cm – were chosen for study. The amounts and specific activities which might be produced in all 18 reactors were calculated, and the approximate costs were estimated for one particular reactor.

INTRODUCTION

This study on the technical and economic aspects of the production of radioisotopes in power reactors was undertaken at the request of the USAEC Division of Isotopes Development. A group from Booz-Allen Applied Research, Inc.,[1] published a report in 1960 on the same subject. In the present study, a somewhat different approach has been made, taking into consideration the new information available and the changes that have occurred in the reactor industry since that time.

Although it is clear that consideration of possible markets for the large amounts of radioisotopes that could be produced by power reactors would be very important and valuable, it was not included in our study. Surveys are being made by the AEC and other governmental groups to collect and analyze the quite meager data that now exist on possible markets for very large quantities of radioisotopes.

Another important consideration left out of this study is the role that may be played by the huge AEC production reactors. It is believed that these reactors will tend to complement the function of commercial reactors, including power reactors, by producing large enough amounts of radioisotopes to develop and nurture markets in their early stages when suitable quantities of such radioisotopes could not conceivably be made available by any commercial reactors. Studies are being made by the AEC on the proper role for the production reactors in large-scale radioisotope production; and, for this reason, these reactors are not considered in this report.

The general characteristics of power reactors, together with the limitations listed above, were used to determine those radioisotopes that appeared to be the most likely candidates for production in power reactors. Information concerning rates of production as a function of neutron flux and irradiation time is also presented. These data, together with a knowledge of the operating cycles and the details of the reactor construction, were used to estimate the annual production in each of the 18 reactors studied. It is emphasized that these estimates are only a first approximation. They are primarily for the purpose of gaining some idea of the production potential and for use as a basis for deciding upon the advisability of

[1] *A Study of the Feasibility and Economics of Radioisotope Production in Power Reactors*, Booz-Allen Applied Research, Inc., AECU-4355 (February 1960).

further study. The radioisotope-production capacity in a reactor cannot be accurately determined without a careful engineering and economic study of the particular reactor system. Moreover, it is clear that because of the competitive aspects of the radioisotope business the number of power reactors actually engaged in such production will have a considerable effect on economic feasibility of doing similar work in any other reactor; assessment of such effects is beyond the scope of this study.

One large modern reactor (Malibu) was examined in somewhat more detail than the others, and an attempt was made to estimate the actual production costs, as well as the production; however, no attempt was made to relate costs to potential revenue. Although it is believed that the results obtained are fairly good approximations, a great deal more detailed work would be required to obtain really good estimates.

FACTORS TO BE CONSIDERED IN POWER REACTOR RADIOISOTOPE PRODUCTION

In principle, any neutron-generated radioisotope can be produced in a power reactor, but actually only a few cases appear to be of interest. Several hundred different radioisotopes are now being produced, but the markets are so limited that in most cases their production is not economically attractive. In contemplating power reactor production of radioisotopes, it becomes evident that megacurie markets would be required in most cases, while many of the radioisotopes are now distributed from Oak Ridge only in millicurie to microcurie quantities – a difference in level of 10^6–10^{12}. Research-type high-flux reactors are best suited for such general radioisotope production, and there exists, at the present time, enough research reactor capacity to take care of the world markets quite easily for a long time.

Short-lived radioisotope production which requires that targets be inserted and removed from the reactor on a frequent schedule would be impractical for power reactors unless special facilities were designed and built into them. This would require penetrations of the reactor pressure vessel, and it is highly unlikely that income from production of short-lived radioisotopes would justify such an expenditure.

Among the considerations which also limit the number of radioisotopes to be studied are the neutron fluxes available in power reactors, the temperature and pressure existing in the reactor core, the operating cycle, and, in some cases, safety. Finally, it goes without saying that the primary purpose of a power reactor is to generate electricity – in all cases the production of radioisotopes would be strictly a secondary function. Thus, any production scheme which would materially interfere with this primary purpose or would result in a significant decrease in electrical capacity could not be considered.

CHARACTERISTICS OF POWER REACTORS

There are, at present, in the United States 23 nuclear power stations either operating, under construction, or firmly committed for construction. Of these, five – Piqua, Fermi, EGCR, HTGR, and the Hanford New Production Reactor (NPR) – were not examined. Four of these are experimental reactors built to demonstrate the feasibility of particular systems, while the NPR is owned by, and operated for, the

Table 1. Power Reactors Studied

Short Name	Full Name	Type	Power [Mw (thermal)]	Location
BONUS	Boiling Water Nuclear Superheat Project	Nuclear superheat BWR	50	Punta Higeura, P.R.
La Crosse	La Crosse Boiling Water Reactor	BWR	165	Genoa, Wisc.
Pathfinder	Pathfinder Atomic Power Plant	Nuclear superheat BWR	189	Sioux Falls, S.D.
Humboldt Bay	Humboldt Bay (PG&E) Power Plant Unit No. 3	BWR	165	Humboldt Bay, Calif.
Dresden	Dresden Nuclear Power Station	BWR	700	Morris, Ill.
Big Rock Point	Big Rock Nuclear Power Plant	BWR	157	Big Rock Point, Mich.
Elk River	Elk River Reactor	BWR	58.2	Elk River, Minn.
9 Mile Point	Nine Mile Point Nuclear Station (Niagara Mohawk)	BWR	1538	Oswego, N.Y.
Shippingport	Shippingport Atomic Power Station	PWR	231	Shippingport, Pa.
Oyster Creek	Oyster Creek Nuclear Power Plant (Jersey Central)	BWR	1600	Ocean County, N.J.
LADWP	Malibu Nuclear Plant	PWR	1473	Los Angeles, Calif.
Indian Point	Consolidated Edison Thorium Reactor	PWR	585	Indian Point, N.Y.
Yankee	Yankee Nuclear Power Station	PWR	540	Rowe, Mass.
Saxton	Saxton Nuclear Experimental Reactor Project	PWR	20	Saxton, Pa.
Conn. Yankee	Conn. Yankee Atomic Power Station	PWR	1473	Haddam Neck, Conn.
So. Calif. Edison	San Onofre Nuclear Generating Station	PWR	1210	San Clemente, Calif.
CVTR	Carolinas-Virginia Tube Reactor	PWR Heavy water	65	Parr, S.C.
Hallam	Hallam Nuclear Power Facility, Sheldon Station	Sodium graphite	240	Hallam, Neb.

USAEC primarily as a plutonium production facility despite the fact that it also produces electrical power. It is interesting to note that the NPR reverses one of our premises: it is primarily a radioisotope-production reactor (the radioisotope being ^{239}Pu), with power production as a secondary mission.

The remaining 18 power reactors (Table 1) may be conveniently divided into four categories: boiling water, pressurized water, heavy-water moderated, and sodium-graphite. The nine boiling-water reactors range in power from 50 to 1600 Mw (thermal); the seven pressurized-water reactors vary from 20 to 1473 Mw (thermal); and two special reactors include a 65-Mw (thermal) heavy-water-moderated pressure tube reactor and a 240-Mw (thermal) sodium-graphite reactor.

No attempt was made to go into the details of power reactor characteristics, but suffice to say that in general they are large reactors, fueled with tons of slightly enriched uranium, operating at fairly high temperature and pressure, operating for long periods of time without a shutdown, and having well-thermalized neutron fluxes of about 1 to 3 ($\times 10^{13}$) neutrons cm^{-2} sec^{-1}. For the most part, these are not the characteristics desired in a radioisotope-production reactor, so it is clear that concessions in radioisotope-production techniques would be necessary to fit this activity into power reactor operation. In the first approximation, it appears that such compromises can be made and the production of large amounts of certain radioisotopes in power reactors is feasible.

RADIOISOTOPES SUITED FOR POWER REACTOR PRODUCTION

Keeping in mind the several limitations previously set forth, the radioisotopes that are of primary interest for production in power reactors should meet the following criteria.

1. They should be relatively long-lived if irradiations are to be fitted into fuel cycles.
2. They can be produced with reasonable efficiency by long irradiations at relatively low flux, particularly in cases where the target material is bulky.
3. Their demand is sufficiently great to justify a high rate of production.
4. Irradiation for their production will not adversely affect the reactor operation.
5. If valuable target material is required, it can be used efficiently under the reactor irradiation conditions.

In applying the third criterion, one is led immediately to consider two classes of products: those that have use in food and industrial irradiators, and those that are of potential use as isotopic power sources. At present, there appear to be no other radioisotope applications for which large potential markets exist. For gamma irradiators, the most likely neutron-produced radioisotope candidate is ^{60}Co. This radioisotope is fairly long-lived (5.3 years), can be produced with adequate specific activity by long irradiation, and also satisfies the other criteria. Among the isotopic power sources, ^{210}Po, ^{238}Pu, ^{242}Cm, ^{244}Cm, and ^{170}Tm appear to meet at least some of the criteria. It should be noted, however, that some of the target materials required for the production of these radioisotopes are very scarce and expensive; and one must consider whether the efficiency of target utilization is sufficiently high to warrant their production in low fluxes.

RADIOISOTOPES SELECTED FOR STUDY

Six radioisotopes were selected on the basis of the criteria previously listed, and these together with a seventh, ^{14}C, were considered for production. Further information is given in the Appendix (pp. 22 to 35).

Isotopes Primarily for Use as Power Sources

Plutonium-238. – This 89-year nuclide is produced by irradiation of ^{237}Np (2.14×10^6 years), which is produced as a by-product of reactor operation through the reaction $^{235}U \xrightarrow{n,\gamma} {}^{236}U \xrightarrow{n,\gamma} {}^{237}U \xrightarrow{\beta-} {}^{237}Np$ and by $^{238}U \xrightarrow{n,2n} {}^{237}U \xrightarrow{\beta-} {}^{237}Np$. In the production reaction, ^{237}Np captures a neutron to produce 2.1-day ^{238}Np, which beta decays to ^{238}Pu. Unfortunately, ^{238}Pu tends to further capture a neutron, producing fissionable ^{239}Pu, thus destroying the product nuclide at a fairly high rate. For this reason, it would be desirable to remove the ^{238}Pu from the neutron field as soon as possible after its formation. Plutonium-238 decays by alpha emission to ^{234}U (2.5×10^8 years) and emits virtually no penetrating radiation; it is thus ideal for an isotopic power source. However, it ranks 6th in the radioisotope toxicity list[2] and therefore must be handled with care.

Polonium-210. – Natural bismuth is made up of 100% ^{209}Bi, the target atom for ^{210}Po production. Bismuth-209 captures a neutron to produce 5-day ^{210}Bi, which beta decays to 138.4-day ^{210}Po. The production cross section is fairly low (0.012 barn), so that a relatively large amount of material must be irradiated to produce the desired quantities of ^{210}Po. Polonium-210 decays by alpha emission to stable ^{206}Pb. It also emits practically no penetrating radiation, making it a highly desirable short-lived isotopic power source. It ranks 37th in the radioisotope toxicity list.

Curium-242. – Curium-242 (163 days) is produced by irradiating the transplutonium isotope ^{241}Am (458 years), which results in a rather complicated group of reactions involving both secondary neutron capture and fission of several of the product nuclides. The production cross section is high, and the goal is to convert as much of the ^{241}Am into ^{242}Cm as possible (40–50%). Curium-242 decays by alpha emission to ^{238}Pu and has relatively little penetrating gamma radiation, but it does emit fast neutrons from spontaneous fission. Curium-242 has an extremely high power density, which makes it of especial interest for thermionic isotopic power. Curium-242 ranks 35th in the radioisotope toxicity list, and it decays to 6th-ranking ^{238}Pu.

Curium-244. – Plutonium-239 is used as the starting material for production of 17.6-year ^{244}Cm in another series of complex reactions involving secondary neutron capture and fission of several intermediate product nuclides. In a reactor which is producing ^{239}Pu, some of the transplutonium isotopes also accumulate after long irradiation, such as 152-year ^{242}Am and 17.6-year ^{244}Cm. Plutonium produced by the power reactors themselves may be of particular interest as target material for ^{244}Cm production because of the accumulation of higher plutonium isotopes (^{240}Pu, ^{241}Pu, ^{242}Pu) during the long power reactor fuel cycles. Curium-244 decays by alpha emission to ^{240}Pu (6760 years). There is a small amount

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[2] *A Basic Toxicity Classification of Radionuclides*, International Atomic Energy Agency, Vienna, 1963.

of gamma radiation, but the fast neutrons from spontaneous fission are by far the most important in shielding considerations. Curium-244 ranks 20th in the radioisotope toxicity list and decays to 5th-ranking ^{240}Pu.

Thulium-170. – Only slight interest has been shown in 127-day ^{170}Tm, but it was included because of the large amount that can be produced relatively easily, its relatively weak gamma radiation (0.084 Mev), and its low toxicity. Natural thulium, which is a relatively scarce rare earth, is composed of 100% ^{169}Tm, the target atom. Thulium-170 is produced by simple neutron capture (80 barns) and beta decays to ^{170m}Yb, from which the 0.084-Mev gamma ray is emitted in going to the ground state, stable ^{170}Yb. Most of the radiation power is in the beta emission (0.88 and 0.97 Mev), and it is therefore at least a candidate as an isotopic power source. Thulium-170 is of medium toxicity, ranking 73rd in the radioisotope toxicity list.

Radioisotopes for Use as Gamma Irradiation Sources

Cobalt-60. – This 5.3-year nuclide is practically the only candidate for gamma irradiation sources among the radioisotopes produced by neutron irradiation. The only serious competitor is fission-product ^{137}Cs. Natural cobalt is made of 100% ^{59}Co, the target atom for production of ^{60}Co. The production cross section (29 barns) is fairly high, and little of the product is lost by further neutron capture. Cobalt-60 decays to stable ^{60}Ni by fairly weak beta emission (0.31 Mev), and most of the decay energy appears as two strong gamma rays (1.17 and 1.33 Mev). For this reason, it is attractive as an irradiation source, but is of limited interest as an isotopic power source. However, the gamma rays can be absorbed in heavy materials, and the resulting heat energy can be used directly for various heating applications. Cobalt-60 ranks 45th in the radioisotope toxicity list.

Radioisotopes Primarily for Research and Instrumentation

Carbon-14. – There are many radioisotopes that one could consider, but the applications demand only a small amount of material; for example, some isotope gages which serve an important function in industry as continuous-film-thickness measurement devices require only a few millionths of a curie of radioisotope. For illustrative purposes, we selected the extremely important radioisotope 5680-year ^{14}C for the test calculations. Carbon-14 is produced by a transmutation (n,p) reaction on ^{14}N, which is usually in the chemical form of beryllium or aluminum nitride. Carbon-14 decays with a 0.155-Mev beta to stable ^{14}N. It is a relatively nontoxic radioisotope (203rd) and is, of course, of no interest as a heat or radiation source. However, ^{14}C is important in physical and biological research; and, because of its long half-life and its production by transmutation reaction, it is a reasonable candidate for power reactor production.

The total amount of ^{14}C which could be produced in the 18 power reactors is about 4000 curies/year (shown in Table 6 for 2 years' production). This is an enormous amount of ^{14}C, when compared to the world market, which is now only about 200 curies/year. At this rate these power reactors could, in 1 year, supply the market for the next 20 years.

Other Radioisotopes That Might Be Produced

With the exception of ^{60}Co and ^{170}Tm, the radioisotopes listed are produced by transmutation reactions; and radioisotopes can be produced at high specific activity, even at relatively low neutron fluxes, if transmutation reactions are used. In such cases, the product element is chemically different from the parent target material and therefore can be separated from it in pure form by chemical means. The best known case of this is, of course, the production of ^{239}Pu from uranium. Several other well-known radioisotopes produced by transmutation are ^{3}H, ^{32}P, ^{35}S, and ^{131}I. Since the neutron flux in power reactors is usually low as compared to research reactors, one would want to take advantage of transmutation reactions to get the high-specific-activity radioisotopes needed for research. However, even in the case of ^{131}I, the market is probably too low to justify competition with research reactors.

ESTIMATION OF RADIOISOTOPE PRODUCTION

The production calculations employed cross sections from Westcott,[3] when available, and other 2200-m/sec values were corrected by Westcott's method to give effective cross sections as follows:

$$\sigma = \sigma_{2200} (g + rs) ,$$

where g is a correction for deviation from $1/v$ behavior at thermal energies, and s includes resonance effects in the epithermal region. Temperature-dependent values of g and s are given by Westcott. The value of r is determined by the proportion of epithermal neutrons in the reactor spectrum, which is assumed to be Maxwellian in the thermal region with a dE/E distribution in the epithermal region. Based on cadmium ratio measurements in the MTR fuel elements, a value of $r = 0.09$ was obtained. This value was used in calculating cross sections by the referenced method. All 2200-m/sec cross sections were adjusted to apply to a Maxwellian distribution about a most-probable velocity corresponding to 300°C (572°F), a temperature chosen to be representative of most power reactor operations. The neutron flux in all cases is the product $n\bar{v}$, where n is the neutron density, and $\bar{v}$ is the average velocity of a Maxwellian distribution about a most-probable velocity corresponding to 300°C. Flux depression and self-shielding effects of the sample material were not included in these calculations but will be introduced later. Also, in all cases, radioisotopes having half-lives greater than 100 years were considered to be effectively stable. (The production of the selected radioisotopes was computed by the use of a suitable computer program – CRUNCH.)[4]

As pointed out earlier, an accurate appraisal of the potential for radioisotope production in any given power reactor *can only be obtained by a careful engineering study of the particular system under consideration.* Moreover, the results will depend upon certain initial assumptions or "ground rules" which describe the extent to which radioisotope production will be permitted to affect a power reactor's primary

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[3] C. H. Westcott, *Effective Cross Section Values for Well Moderated Thermal Reactor Spectra*, CRRP-960 (Nov. 1, 1960).

[4] M. P. Lietzke and H. C. Claiborne, *CRUNCH – An IBM-704 Code for Calculating n-Successive First Order Reactions*, ORNL-2958 (Oct. 24, 1960).

function – the production of electric power. In power producers, interest is, of course, primarily conducting an economically sound business operation to yield a profit; radioisotope production will be attractive only if it generates sufficient revenue to more than offset any additional operating costs incurred, including the value of any associated loss of electrical capacity. Obviously, this implies that marketing studies, as well as engineering studies, are necessary in order to make a clear determination of the desirability of radioisotope production in power reactors.

Examination of the design details of the power reactors revealed, in each case, a number of places in which target materials could be inserted (Appendix, pp. 36–42). Most of the information used was obtained from the hazards evaluation reports for the various reactors; and, in general, no attempt was made to seek further details. Thus, the information is valid only insofar as these reports reflect the actual final design and operating conditions of the reactor.

Although it may prove feasible to install special devices for the insertion and removal of target materials during operation or to develop other special arrangements for irradiation, only stationary target sites currently existing in the reactors were considered. It was assumed that the insertion and removal of the targets would coincide with the normal refueling cycle of the reactor, which ranges from 6 to 18 months in the various power reactors. The possible target sites include such locations as control rods, poison curtains, flow baffles, flux depressors, and other components which normally contain high-cross-section material. Other sites are unused fuel positions, particularly on the periphery of the reactor. Irradiation space can also be obtained by replacing a small amount of the fuel with target material.

The use of components such as control rods and flux depressors, whose normal function is to modify the shape of the neutron flux, is feasible only if the target material can be utilized in such a way that this function is not impaired. This is not always possible and, in any case, requires a careful examination of the physics involved. In the case of peripheral positions, the neutron flux is often considerably different from the average values listed in the documents used.

Insertion in shim-rod followers would require considerable redesign and probably would result in low flux due to the need for heavy structural parts and the fact that the followers would remain in the reactor only a portion of the total cycle time. Insertion of targets in the poison portion of the rods is not attractive because of the lower flux and because the removal of shim rods for insertion or removal of targets would add considerably to the operating problems.

Peripheral positions are available in some power reactors, and these offer attractive irradiation sites since they do not interfere with fuel and can be changed readily during any shutdown for reloading fuel. However, these positions often have markedly lower fluxes than the center portion of the core and are not available at all in the newest designs, in which the diameter of the pressure vessel is optimized.

While some radioisotopes could be produced in peripheral positions in those reactors which have them or by redesigning components such as control rods, it appears that because of limited volume and lower flux (thermal fluxes below 10^{13} neutrons cm^{-2} sec^{-1} were considered unsuitable) the amounts would be rather limited. It was decided that for a large program such positions would not contribute sufficiently to be considered at this time. Of course, for small production programs these positions, especially the peripheral ones, could be used.

For the reasons cited, the overall estimates of the production potential were obtained under the assumption that 3% of the fuel assemblies in each reactor are replaced by target assemblies and that 80% of the fuel-element volume is subject to a thermal-neutron flux represented by the average values given. Thus, 2.4% of the fuel volume was assumed to be available as target sites. (Eighty percent was chosen so that the targets would be positioned vertically in the central portion of the core.)

Some of the target material proposed has a high neutron-absorption cross section, and this would result in a reduction in the effective neutron flux in the target. In a more detailed study, one would compute the optimum target concentration which results in maximum production for each case. This will vary from place to place in the reactor, and the values of the reactor parameters involved are needed for the computation. Since these values were not available, it was assumed that the high-cross-section

Table 2. Radioisotope Production (megacuries) in Power Reactors

The amount of time shown beneath each isotope is the period of irradiation on which the calculated yields for that isotope are based

	Reactor Power [Mw (thermal)]	^{238}Pu (2 years)	^{210}Po (1 year)	^{242}Cm (1 year)	^{244}Cm (4 years)	^{60}Co (2 years)	^{170}Tm (1 year)
Big Rock Point	240	0.09	<0.01	4.4	0.5×10^{-3}	1.7	6.8
BONUS	50	0.01	<0.01	0.6	$<0.1 \times 10^{-3}$	0.2	0.9
CVTR	64.9	0.02	<0.01	1.0	0.1×10^{-3}	0.4	1.7
Conn. Yankee	1,473	0.50	0.02	23.8	1.8×10^{-3}	9.2	35.7
Dresden	630	0.25	0.02	11.3	3.5×10^{-3}	5.1	19.8
Elk River	58.2	0.02	<0.01	0.9	$<0.1 \times 10^{-3}$	0.3	1.2
Hallam	240	0.05	<0.01	2.3	$<0.1 \times 10^{-3}$	0.8	3.1
Humboldt Bay	165	0.08	<0.01	3.7	0.1×10^{-3}	1.5	5.3
Indian Point	585	0.18	<0.01	8.4	0.4×10^{-3}	3.2	12.4
La Crosse	165	0.07	<0.01	3.1	0.2×10^{-3}	1.2	4.8
Malibu	1,473	0.46	0.02	21.4	2.5×10^{-3}	8.7	33.8
9 Mile Point	1,538	0.39	0.03	23.0	4.4×10^{-3}	9.9	37.3
Oyster Creek	1,600	0.82	0.05	38.6	7.3×10^{-3}	15.7	62.7
Pathfinder	188.9	0.08	<0.01	3.8	1.2×10^{-3}	1.7	6.5
Saxton	20	<0.01	<0.01	0.2	$<0.1 \times 10^{-3}$	<0.1	<0.1
Shippingport	231	0.23	0.02	9.6	7.4×10^{-3}	4.8	19.1
So. Calif. Edison	1,210	0.43	0.02	19.9	1.5×10^{-3}	7.7	30.6
Yankee	392	0.32	0.01	13.1	4.1×10^{-3}	6.0	22.8
Total	10,324	4.01	0.22	189.1	0.035	78.1	304.5

Table 3. Radioisotope Production (kw) in Power Reactors

The amount of time shown beneath each isotope is the period of irradiation on which the calculated yields for that isotope are based

	Reactor Power [Mw (thermal)]	^{238}Pu (2 years)	^{210}Po (1 year)	^{242}Cm (1 year)	^{244}Cm (4 years)	^{60}Co (2 years)	^{170}Tm (1 year)
Big Rock Point	240	3.1	0.13	158	<0.02	24	18
BONUS	50	0.4	0.02	22	<0.01	3	2
CVTR	64.9	0.7	0.04	37	<0.01	6	4
Conn. Yankee	1,473	16.8	0.74	857	0.06	124	94
Dresden	630	8.2	0.59	408	0.12	68	52
Elk River	58.2	0.6	0.03	32	<0.01	4	3
Hallam	240	1.6	0.12	82	<0.01	11	8
Humboldt Bay	165	2.6	0.13	132	<0.01	20	14
Indian Point	585	6.0	0.19	301	0.01	43	32
La Crosse	165	2.2	0.10	113	<0.01	16	13
Malibu	1,473	15.3	0.76	773	0.08	117	89
9 Mile Point	1,538	13.0	1.05	828	0.15	134	98
Oyster Creek	1,600	27.4	1.58	1392	0.24	211	164
Pathfinder	188.9	2.7	0.14	136	0.04	23	17
Saxton	20	0.2	<0.01	8	<0.01	1	<1
Shippingport	231	7.7	0.48	346	0.25	65	50
So. Calif. Edison	1,210	14.3	0.68	720	0.05	104	80
Yankee	392	10.7	0.41	474	0.14	80	60
Total	10,324	133.7	7.21	6819	1.18	1053	799

target material (cobalt metal and Am, Tm, Np, and Pu in the form of oxides) would be loaded in that concentration that would produce a reduction in the neutron flux by a factor of 0.8; the concentrations of these target materials were adjusted accordingly.

Finally, *it was assumed in the production calculations that a sufficient quantity of target material is available*, although it will be shown later that this is not true in at least two cases.

The production potential for the seven radioisotopes ^{238}Pu, ^{210}Po, ^{242}Cm, ^{244}Cm, ^{60}Co, ^{170}Tm, and ^{14}C was estimated under the foregoing assumptions.

The results for the 18 reactors are given in Tables 2 and 3, with the radioisotope production shown both as megacuries and kilowatts of radiation output. The production of ^{14}C is not listed separately for each of the reactors, but the total is shown in Table 6. Tables A.26–A.33 in the Appendix present more details on production of the radioisotopes; Table A.29 is devoted exclusively to ^{14}C.

AN ESTIMATE OF COSTS

In estimating cost items, interest, overhead, insurance, and other miscellaneous charges were not included. Only direct costs were considered, except where possibly some indirect costs had been included in the original data and their separation would require detailed investigation.

The costs of a full-scale radioisotope program in the Malibu Reactor were estimated, based on the irradiation of targets inside the fuel region. This reactor was chosen as an example because it appears to be easily adaptable to an isotope-production program.

The target materials were divided into two groups, the first having high cross sections for producing ^{60}Co, ^{170}Tm, ^{238}Pu, ^{242}Cm, and ^{244}Cm, and the second having very low cross sections for producing ^{14}C and ^{210}Po. In the first case, an appreciable enrichment of the reactor fuel is required; none is required in the second. Details of the cost estimates are given in the Appendix, pp. 51 to 53.

In computing the additional fuel costs resulting from the insertion of radioisotope targets in the reactor, it was assumed that the fuel cycle would not be changed because of the radioisotope production. The fuel enrichment required to accomplish this was calculated on the premise that the thermal utilization remains constant, which would, as a first approximation, tend to keep the fuel cycle unchanged. However, again it is recognized that detailed calculations would be necessary to refine the results in actual practice. In the case of the Malibu Reactor, it is not necessary to remove fuel in order to insert target material equivalent to 2.4% of the fuel volume (see p. 51 in the Appendix). Thus, no additional enrichment is required in order to compensate for loss of fuel.

The initial enrichment of the Malibu Reactor is 3.8%. No additional enrichment is required where the product under consideration is ^{210}Po or ^{14}C because their target materials have very low cross sections. In the case of the other isotopes, the enrichment will be increased to 4.3%. This is discussed in the Appendix, pp. 51 to 52.

A cost summary, Table 4, shows that for most of the radioisotopes studied the reactor cost is a relatively small amount of the total cost, being less than 50% in all cases. The elements making up target

Table 4. Unit Costs for Isotopes Produced in the Malibu Reactor

	Price		Cost Elements (%)	
	($/curie)	($/watt)	Reactor	Target
^{238}Pu	200	60	<1	>99
^{210}Po	10	318	12	88
^{242}Cm	4	110	<1	>99
^{244}Cm	2000	58,200	27	73
^{60}Co	0.17	13	44	56
^{170}Tm	0.05	18	21	79
^{14}C	1200		6	94

Table 5. Target Cost Elements

	Target Compound	Price ($/kg of target compound)	Kilograms/Rod[a]	Canning ($/rod[a])	Total Target ($/rod[a])
^{238}Pu	NpO_2	440,500	0.245	300	108,000
^{210}Po	Bi metal	1	1.74	250	252
^{242}Cm	AmO_2	1,324,000	0.07	5000	98,000
^{244}Cm	PuO_2	39,700	0.09	300	4,280
^{60}Co	Co metal	15	0.34	1000	1,000
^{170}Tm	Tm_2O_3	2,750	0.425	300	1,460
^{14}C	Be_3N_2	400	0.283	1000	1,130

[a]Estimated for rods for insertion in the Malibu Reactor.

costs are shown in Table 5. It should be pointed out that the largest potential savings are in the materials costs for expensive targets, such as ^{241}Am and ^{237}Np, and in the costs of canning the materials for irradiation. It should be possible to lower both of these types of costs on a full-scale production basis with the application of further research and development.

TARGET MATERIAL AVAILABILITY

Table 6 lists the target materials required for the 18 power reactors, assuming in each case that all reactors are filled with the one target. For five of the seven product isotopes, the target material appears to be reasonably available, assuming that the plutonium target (for ^{244}Cm) might be supplied by the AEC. Two, however, ^{237}Np and ^{241}Am, are not available in the amounts needed. They would have to be produced largely in the power reactors considered in this study since these constitute the major potential source unless AEC production and research reactors were used.

The annual production of 189 megacuries of ^{242}Cm from the 18 power reactors requires that 460 kg of ^{241}Am be available annually. Actually, only about 1% of this would be available from these reactors initially. The availability of this material will increase with time since it results from the decay of ^{241}Pu. The annual production rate of ^{241}Pu from the 18 power reactors would be about 85 kg. This would result in the production of about 4.3 kg of ^{241}Am the first year (neglecting any current stockpiles of ^{241}Pu). Assuming that ^{241}Pu production remained constant, this would increase to 35 kg of ^{241}Am in 10 years and reach a peak of 77 kg of ^{241}Am in about 70 years.

The current production of ^{237}Np is about 31 kg/year as compared to the maximum of 775 kg which could be loaded for the purpose of producing, annually, 2 megacuries of ^{238}Pu (note that the ^{238}Pu figures in Table 6 apply to a two-year irradiation cycle). The ^{237}Np results from the decay of ^{237}U, which

Table 6. Target Material Required for the Power Reactors Studied

	Curies of Product per kg of Target (av)	Total Production (megacuries)	Target Required (kg)
^{238}Pu	2,600	4.01	1,550
^{210}Po	18	0.22	11,900
^{242}Cm	411,000	189.1	460
^{244}Cm	55	0.035	640
^{60}Co	31,240	78.1	2,500
^{170}Tm	113,000	304.5	2,700
^{14}C	7.2	7.9×10^{-3}	1,100

has a rather short half-life (~6.75 days). Because of this and because of the extremely long half-life of ^{237}Np, the production of the latter remains essentially proportional to the total reactor power available for production.

Since the supply of these target materials is limited, it would seem advisable to use them in the most efficient manner, both from the standpoint of cost and for the conservation of valuable natural resources. In the first place, it would appear advisable to separate the plutonium and neptunium from the spent fuel as soon as possible after removal from the reactor. This would permit disposal of the depleted uranium and thus give relief from the interest charges. The plutonium could be stored and the ^{241}Am separated from it at periodic intervals. The economic feasibility of this would, of course, depend upon the revenue available from the immediate sale of the plutonium vs that which could be realized from the production and sale of ^{242}Cm. The neptunium could also be stored until ready for use or sold immediately depending upon economic factors.

Because of the relatively small amount of target material available, it would seem sensible to collect all this material from the several power reactors and reirradiate it in those reactors which would yield the most economical and efficient use of the targets. For this purpose it might be better to use one or more of the high-flux test reactors, such as GETR, MTR, or ORR. Moreover, in the case of ^{237}Np, a circulating loop or some other scheme, such as a short irradiation cycle which permits frequent removal and separation of the product, would materially increase the amount of product obtained per unit quantity of initial target irradiated. The decision of whether to use the relatively low-flux power reactors or the high-flux test reactors would depend upon economic as well as physical considerations and also upon the availability of space in the reactors. For example, the amount of ^{242}Cm per gram of original ^{241}Am which can be produced is, in the flux ranges under consideration, rather insensitive to the flux but is nearly proportional to the *nvt*. Consequently, while it requires more time, essentially the same amount can be produced in a flux of 2×10^{13} as can be produced in a flux of 2×10^{14}. On the other hand, the amount of ^{241}Am destroyed per gram of ^{242}Cm produced is flux dependent.

The conditions for producing target materials might be optimized on a number of bases, but this cannot be done in a meaningful fashion until the economic ground rules have been established. These include such items as the potential market for the products, the revenue from the sale of ^{242}Cm or ^{244}Cm vs the revenue from the sale of plutonium, the way in which interest charges are applied, and the cost of chemical processing.

DISCUSSION

In making this study, it became evident that different conditions prevail at nearly every utility company and with each reactor manufacturer. This makes it difficult to generalize about the many problems of setting up a radioisotope production program. In spite of this risk, it seems worthwhile to discuss some of these problems, although our comments may not apply in all cases.

Compatibility of Isotope Production

An isotope-production program at a power reactor must conform to the refueling cycle because any appreciable loss of plant availability caused by the program would reduce its attractiveness. It is doubtful if any utility company would agree to change their refueling cycle or to have the reactor shut down for an appreciable time beyond the normal shutdown period. This will require some ingenuity in designing targets and in the insertion and removal procedures.

Some Effects of Isotope Target Insertion on Reactor Core

If the heat transfer per unit area limits the capacity of a power reactor in which it is necessary to remove fuel to insert targets, the power would have to be reduced, with a consequent loss of capacity. Some of the earlier reactors have appreciably more capacity than their turbines, and in these, of course, the loss of fuel would not limit the capacity of the plant since the heat flux could be increased on a smaller total heat-transfer area; however, it seems probable that future reactors will be closely matched to their turbines, and the fuel will be operated at the maximum permissible heat flux.

Most boiling-water reactors and some pressurized-water reactors studied have completely filled fuel assemblies or bundles (i.e., all positions are filled by fuel rods). The addition of target rods within such fuel bundles would require that an equal number of fuel rods be removed. This, naturally, would reduce the fuel in the reactor at the same time that poison in the form of the target is being added. In one of the latest pressurized-water reactor designs, 16 spaces within each fuel assembly are left vacant to accommodate a control-rod cluster. The control-rod clusters would be used in only about half the fuel assemblies, leaving the remaining assemblies with a number of open spaces. This design is especially attractive for a radioisotope program since fuel would not be displaced and the heat-producing surface in the reactor would not be reduced.

Changing Design of Fuel

If a large isotope production program were begun, it would be necessary to redesign the fuel bundles or assemblies for most power reactors so that target insertion and removal could be carried out quickly without prolonging shutdowns. The fuel enrichment would also have to be increased to compensate for the added poison and, in some cases, the overall loss of fuel.

Utility companies often look to the reactor designer or fuel manufacturer for approval of design changes, and any change which affected the fuel would usually have to be worked out with both the reactor manufacturer and the utility company. The reactor manufacturer should normally be better able to assess the effect of changes on reactivity and heat flux since the original calculations and core measurements on the reactor in question were made by the manufacturer's engineers. Some utility companies may, however, elect to handle such design changes themselves and, from an administrative standpoint, this should be simpler.

In most power reactors only one-third to one-fifth of the fuel will be changed each year, whereas most isotope targets should be changed yearly (or less often). The targets in those fuel bundles remaining in the reactor would, therefore, have to be arranged so that they could be changed within a reasonable time. The redesigned assemblies would have to be compatible with the assemblies already in the reactor since both would have to be handled by the same tools and equipment. Also, the changes should not increase the hazard of mechanical failure of the isotope targets themselves. Since targets may have to be changed three to five times during the life of the fuel assembly, such procedures would obviously have to be harmless to the fuel assembly itself.

Interest of Reactor Designer and/or Fuel Manufacturer in Isotope Target Irradiations

Some utilities have a very close relationship with the reactor designer-builder and/or fuel manufacturer, under which the latter warrants performance of the reactor for a number of years. This service includes designing the fuel, calculating the change in fuel enrichment necessary to maintain the desired length of cycle, limiting the heat transfer to the chosen value, etc. In cases where this relationship exists, it is believed that the preliminary and detailed design of the fuel to accommodate the radioisotope program should be done by the reactor designer or fuel manufacturer. If a third firm should attempt to redesign the fuel, a great deal of confusion might ensue. Questions would arise concerning proprietary information which might have to be divulged; and, in cases where a bonus-penalty contract existed on the performance of the reactor, any change affecting the core would affect the contractual rights of the manufacturer.

Beneficial Effect of Isotope Targets

Isotope targets could serve as beneficial neutron poisons to reduce flux peaking at strategic regions in the fuel. Fuel suppliers probably would take advantage of this if they were asked to design a fuel loading for radioisotope production. The fuel assemblies presently being used in power reactors are

sometimes poisoned by added material or by increased cladding thickness in certain regions of the assembly, and isotope targets might be used advantageously in such locations.

Planning for Radioisotope Production During Design

While some changes could be made to accommodate radioisotope production in existing reactors and in some cases the inclusion of targets as planned reactor poisons could be beneficial, it would be much easier to make such changes to reactors during the design stage. Additional space around the periphery of the core might be provided at little added cost. It might even be found feasible to include equipment for insertion and removal of miscellaneous targets during operation or to design, into the reactor, facilities for circulating loops (e.g., to produce ^{238}Pu), if these features are examined in the early stages.

Schedule for a Power Reactor Isotope Program

Briefly, the time required to start an irradiation program will depend upon the changes required in a reactor and the time required to fabricate suitable targets, as follows: No enrichment change, simple targets – one year; an enrichment change, complex targets – two to three years.

While a program could be started in one year, it would be desirable to carry out an investigative program for at least one year to study the design changes necessary and develop targets. In this way, obvious pitfalls, such as rupture of targets or decrease in fuel cycles, would be avoided. After the preliminary study, about two years would be required to start irradiation of the first production targets; however, small-scale irradiations could, of course, be started much sooner.

At least a year in advance of concluding firm agreements for production, a program of familiarizing the utility companies with the program and developing the general features of a radioisotope program in each reactor should be started. If begun on such a schedule, there would be time to develop sufficient details to satisfy the utilities that the program would be compatible with operations; and, when a time came to make firm contracts, both parties would have a good idea of the factors involved. If started on a shorter schedule, the benefits of preplanning would be lost, the cost would probably be much higher, and the utility companies would not have sufficient opportunity to study the effects on normal operation. This same lead time would be necessary in any case (perhaps even more time) to develop the proper forms of target material to avoid costly failures due to target cladding rupture. The details of the program could be optimized at each reactor by studies of target location, effect on heat flux, number and blackness of targets, specific activity of product, cost of enrichment of fuel, effect on operations, and other factors. In summary, the following tasks would be attempted during the first year.

1. Any necessary redesign of fuel elements would be made, information on enrichment of fuel would be developed, and the program would be optimized for each reactor.
2. For each reactor, a general plan should be developed which would permit targets to be loaded and unloaded during a normal outage without prolonging the shutdown.
3. The tools, casks, and other additional equipment required should be designed in a preliminary fashion but in sufficient detail to satisfy the utilities that the handling operations are safe and feasible.

4. Development of target preparation and testing procedures would be started. Some of the target materials are quite hazardous, and the use of present-day techniques might result in excessively high costs. Canning plutonium and neptunium, for example, would have to be done by improved remote techniques. The target should be sufficiently impervious to erosion and corrosion in the reactor so that only permissible quantities would be leached out if the cladding should fail.

Following the preliminary development of an isotope program at each reactor and the development of satisfactory target materials, a lead time of about two years would be necessary from the time a decision was made to proceed before a full load of targets could be installed in most reactors. This assumes that enough of the necessary development had been done and that the general details had been sufficiently established to make it possible to reach an agreement with the utility companies within a few months. Once the general framework of the program is established, the detailed design of the isotope targets and any changes required in the reactor, such as the fuel enrichment and changes in fuel assemblies, should be determined so that the target manufacturer could begin production in time to be ready for incorporation with the next loading of fuel. Assuming that the redesign of the fuel could be completed in six months to a year, an additional year would be required to manufacture the new fuel so that the targets could be accommodated. Further, most power reactors schedule their outages in the spring or fall, and as much as four to six months might be lost waiting for a scheduled outage.

Some targets could be loaded in special locations much sooner; and, in reactors using control-rod-cluster fuel, perhaps a year could be saved if no appreciable changes were required in fuel assemblies.

Development of Target Material

Utility companies might require a guarantee against loss due to any malfunction, such as a rupture of a target, which might contaminate the primary cooling system and complicate operations during shutdowns. For example, if an alpha-emitting target material were widely distributed throughout the reactor vessel, pumps, piping, and heat exchangers or the turbine of a power reactor, maintenance and fuel handling might be made considerably more difficult for years to come. Shutdowns would be prolonged, and other problems would develop during operations. For these reasons, great emphasis should be placed on target development, including the fabrication of target material in a form resistant to corrosion or erosion. The cost of a prolonged shutdown of a large power reactor due to loss of revenue and incremental charges would be considerable, and a testing program should be undertaken to ensure that good targets are available.

Some of the elements of a target-development program are listed below.

1. Target material should contain the proper concentration of active targets in an inactive, low-absorbing diluent to obtain the desired neutron absorption and heat-transfer characteristics. The unclad target pellet should be resistant to dissolution or erosion in the reactor coolant if the cladding fails. It would be desirable to irradiate the targets in a test reactor to demonstrate that the rate of release of radioactive contamination is satisfactorily low.
2. Cladding for the target material should be developed to ensure a low failure rate. A method of testing the cladding should be developed, and tests of the cladding should be carried out in the reactor environment.

3. Reactor tests would be necessary on the complete target assemblies to show that the target material and cladding suffer no deleterious effects due to irradiation such as brittle fracture or buildup of gas pressure.
4. Different methods of preparing the target should be tried to reduce the cost. This is especially necessary for americium, neptunium, plutonium, and similar materials.
5. The effects of cladding rupture should be measured by irradiating capsules with known defects under reactor conditions. To avoid release of radioactive material, double-canning techniques may be necessary.
6. Methods of detecting failure in target cladding should be investigated where necessary.
7. Methods of decontaminating a reactor system should be available in case a target should rupture and cause serious contamination.

Efficient Use of Target Materials

The efficiency of radioisotope production in power reactors – as compared to research reactors and possibly production reactors – will be an important factor in their possible use for this purpose. The low flux in power reactors is a point against their use for radioisotope production. Larger amounts of target materials are required than in high-flux reactors for equivalent production; and, in the case of ^{60}Co and ^{170}Tm, the lower power density of the product is also an important factor. However, usable specific activities of ^{60}Co can be made, and little is known yet about whether ^{170}Tm will have any importance as an isotopic power source.

Comparative Power of Isotopes Studied

Table 7 has been included to illustrate the potential of power reactors in the field of special radioisotope production. The amount of power that can be packed into ^{242}Cm is particularly interesting, as is the amount of energy that can be put into ^{238}Pu and ^{60}Co. Further, it is notable that the power and

Table 7. Comparative Power of Isotopes from the Power Reactors Studied

Isotope	Half-Life	Irradiation Time (years)	Heat Power (kw)	Energy, One Half-Life (kwh)
^{242}Cm	163d	1	6819	19.2×10^6
^{60}Co	5.3y	2	1053	37.5×10^6
^{170}Tm	127d	1	799	1.7×10^6
^{238}Pu	89y	2	134	74×10^6
^{210}Po	138d	1	7.2	0.024×10^6
^{244}Cm	17.6y	4	1.2	0.18×10^6
^{90}Sr[a]	28y	1	50	9.6×10^6

[a]Formed as fission product in the reactors; listed for comparison only.

energy available from fission product ^{90}Sr formed in the power reactor fuels is not as great as that of some of the neutron products.

CONCLUSIONS

1. It is technically feasible to produce certain radioisotopes in power reactors. For example, in one particularly well-suited reactor, it is estimated that approximately 5,000,000 curies of ^{60}Co could be produced per year.

2. It appears that certain radioisotopes can be produced economically on a very large scale in some existing power reactors if adequate markets are developed and indirect costs (e.g., target materials, target preparation, reprocessing, etc.) are not controlling factors.

3. Certain present-day power reactor designs are particularly adaptable to auxiliary radioisotope production (e.g., the control-rod-cluster design). The first of the reactors incorporating this design will be ready for operation in approximately three years.

4. In less suitably designed reactors, it probably would require several years to make reactor structural changes or changes in fuel-element design to accommodate radioisotope-production targets.

5. If a large-scale radioisotope production program is envisioned for power reactors, changes in future power reactor designs to accommodate radioisotope production should be considered. It may be possible to incorporate these changes at relatively small additional capital cost.

6. Several of the power demonstration reactors, which will require additional financial support in the future, may be particularly advantageous places to initiate small production or development programs.

7. A strong target-development program would be an essential prerequisite for radioisotope production in power reactors.

ACKNOWLEDGMENTS

We wish to acknowledge the assistance of E. E. Beauchamp, E. N. Cramer, T. P. Hamrick, W. A. Hartman, E. Lamb, J. P. Little, M. W. Rosenthal, R. W. Schaich, and J. H. Swanks in the preparation of this report.

APPENDIX

COBALT-60

The nuclear processes considered for production of ^{60}Co from ^{59}Co are shown in Fig. A.1. Some metastable 10.4-min ^{60m}Co is formed which is lost by beta decay to ^{60}Ni, neutron capture to form ^{61}Co, or isomeric transition to ^{60}Co. The thermal-neutron capture cross section of 10.4-min ^{60m}Co is 100 barns; but, in a flux as high as 10^{15} neutrons cm^{-2} sec^{-1}, the transformation rate by neutron capture is only 0.01% of the beta decay rate. The ratio of isomeric transition to beta decay is >99 : 1 (ref. 1); therefore, isomeric transition is predominant, and very little error is introduced by neglecting the formation of ^{60m}Co and assuming that all ^{59}Co is converted directly into ^{60}Co. CRUNCH input data are shown in Table A.1, and the results are presented in Fig. A.2. References 2 and 3 to 9 were used to compute the input data.

ORNL-DWG 65-2238

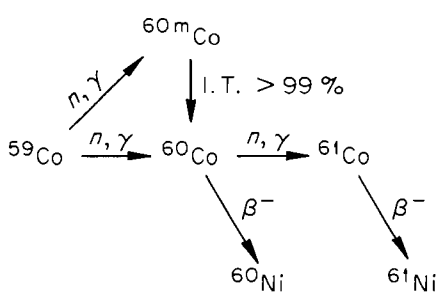

Fig. A.1. Production of ^{60}Co from ^{59}Co.

Table A.1. CRUNCH Input Data for ^{60}Co

Isotope	Half-Life	λ (sec^{-1})	$(\sigma_c)_{2200}$ (barns)	$\bar{\sigma}_c$ (barns)
^{59}Co	Stable	0	45.5	28.7
$^{60m}60$	10.4m			
^{60}Co	5.24y	4.17×10^{-9}	6	3.8
^{61}Co	99m			
^{60}Ni	Stable	0		
^{61}Ni	Stable	0		

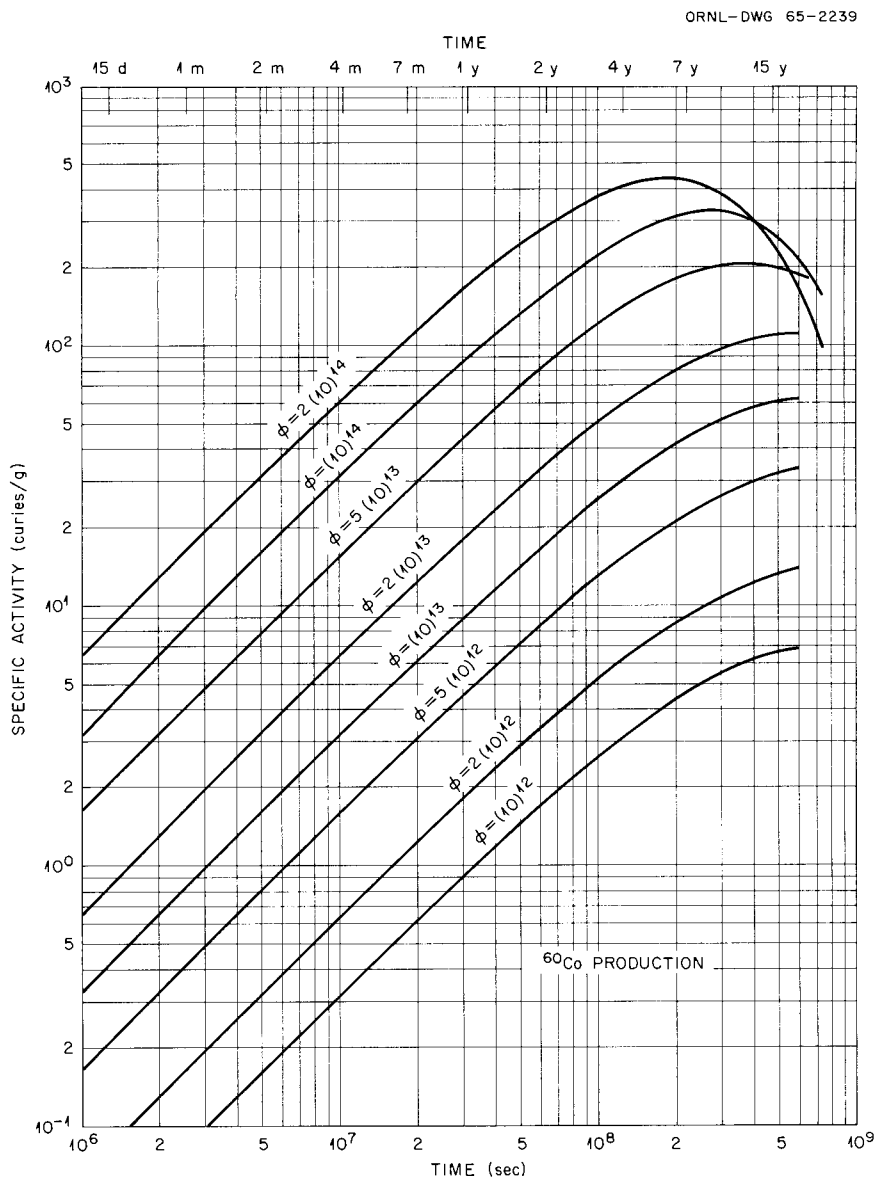

Fig. A.2. Production of ^{60}Co.

THULIUM-170

Natural thulium is composed of 100% ^{169}Tm, and the calculation of the production of ^{170}Tm was performed according to the scheme shown in Fig. A.3. Unfortunately, Westcott's compilation of cross sections does not include ^{169}Tm; and the 2200-m/sec value of 125 barns was used. This neglects the resonance peak of 1700 barns at 3.9 ev and several smaller peaks in the range 10 to 100 ev. The production of ^{170}Tm shown in Fig. A.4 is probably lower than can be achieved. Table A.2 includes the input data and sources.

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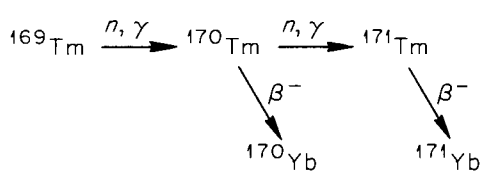

Fig. A.3. Production of ^{170}Tm.

Table A.2. CRUNCH Input Data for ^{170}Tm

Isotope	Half-Life	λ (sec^{-1})	$(\sigma_c)_{2200}$ (barns)	$\bar{\sigma}_c$ (barns)
^{169}Tm	Stable	0	125	79.5
^{170}Tm	127d	6.32×10^{-8}	150	95.4
^{171}Tm	680d			
^{170}Yb	Stable	0		
^{171}Yb	Stable	0		

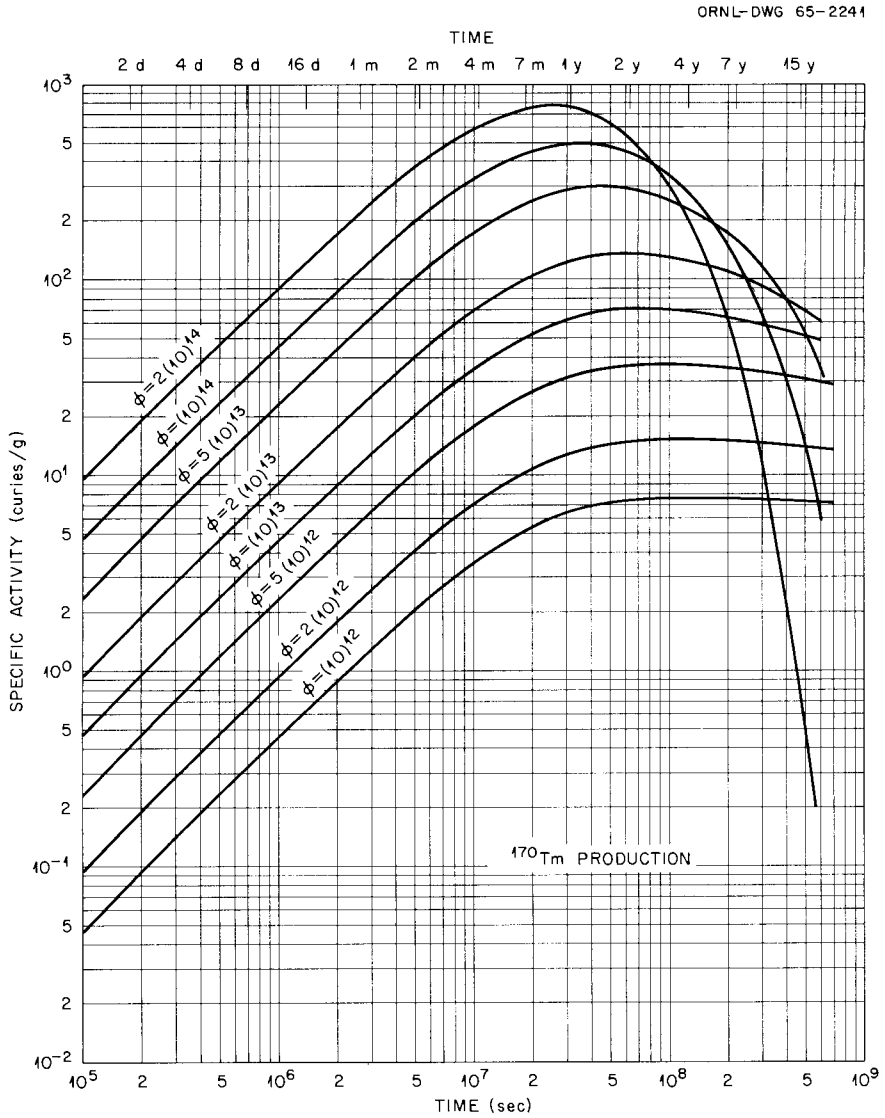

Fig. A.4. Production of ^{170}Tm.

CARBON-14

Carbon-14 can be produced from ^{14}N by an (n,p) reaction, which competes with the (n,γ) reactions, as shown in Fig. A.5. The effect of ^{14}C decay to ^{14}N by beta decay and the subsequent reproduction of ^{14}C is negligible. Table A.3 gives the input data, and the results are shown in Fig. A.6.

ORNL-DWG 65-2242

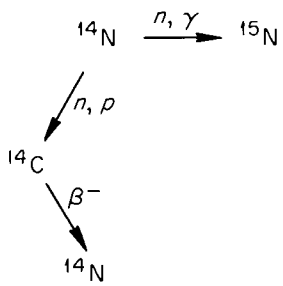

Fig. A.5. Production of ^{14}C from ^{14}N.

Table A.3. CRUNCH Input Data for ^{14}C

Isotope	Half-Life	λ (sec^{-1})	$(\sigma_c)_{2200}$ (barns)	$\bar{\sigma}_c$ (barns)
^{14}N	Stable	0	(n,p) 1.75	(n,p) 1.10
^{14}N			(n,γ) 0.08	(n,γ) 0.05
^{15}N	Stable	0		
^{14}C	5680y	3.866×10^{-12}		

ORNL-DWG 65-2243

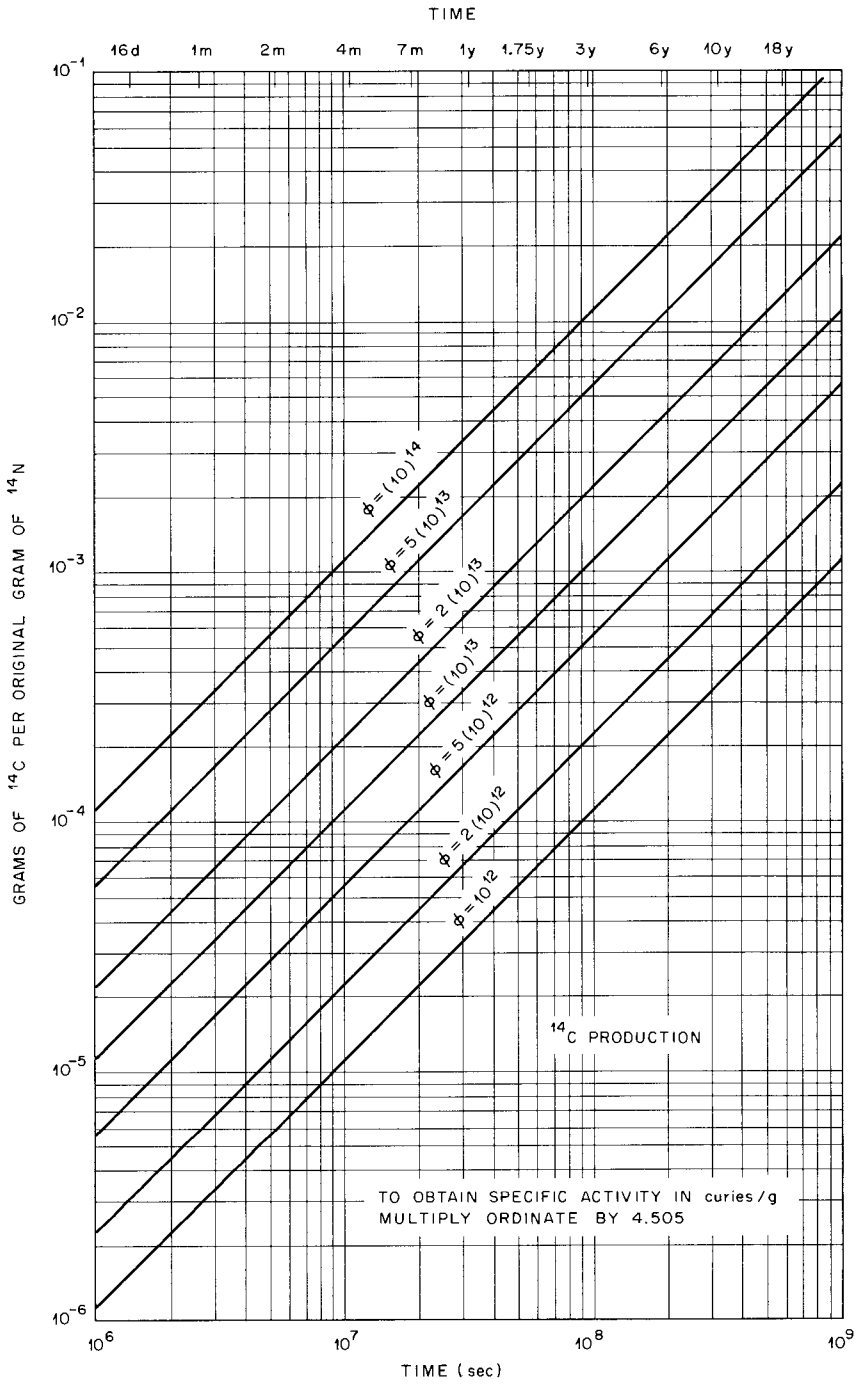

Fig. A.6. Production of ^{14}C from ^{14}N.

POLONIUM-210

The production of ^{210}Po from ^{209}Bi was calculated by considering the reactions shown in Fig. A.7. Contributions from alpha decay of ^{210}Po and ^{211}Bi and subsequent reactivation were found to be negligible. In fact, the buildup of lead isotopes does not increase the sample cross section significantly even after 20 years' exposure in a flux of 10^{14} neutrons cm^{-2} sec^{-1}. The input data are listed in Table A.4, and the results are shown in Fig. A.8.

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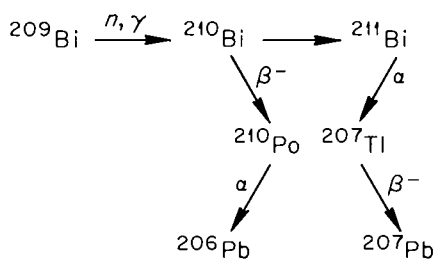

Fig. A.7. Production of ^{210}Po from ^{209}Bi.

Table A.4. CRUNCH Input Data for ^{210}Po

Isotope	Half-Life	λ (sec^{-1})	$(\sigma_{act})_{2200}$ (barns)	$(\sigma_a)_{2200}$ (barns)	$\bar{\sigma}$ (barns)
^{209}Bi	Stable		0.019		0.012
^{210}Bi	5.0d	1.6×10^{-6}	0.063		0.04
^{211}Bi	2.15m	5.37×10^{-3}			
^{210}Po	138.4d	5.77×10^{-8}		<0.03	
^{206}Pb	Stable			0.03	0.019
^{207}Tl	4.78m	2.42×10^{-3}			
^{207}Pb	Stable			0.73	0.47

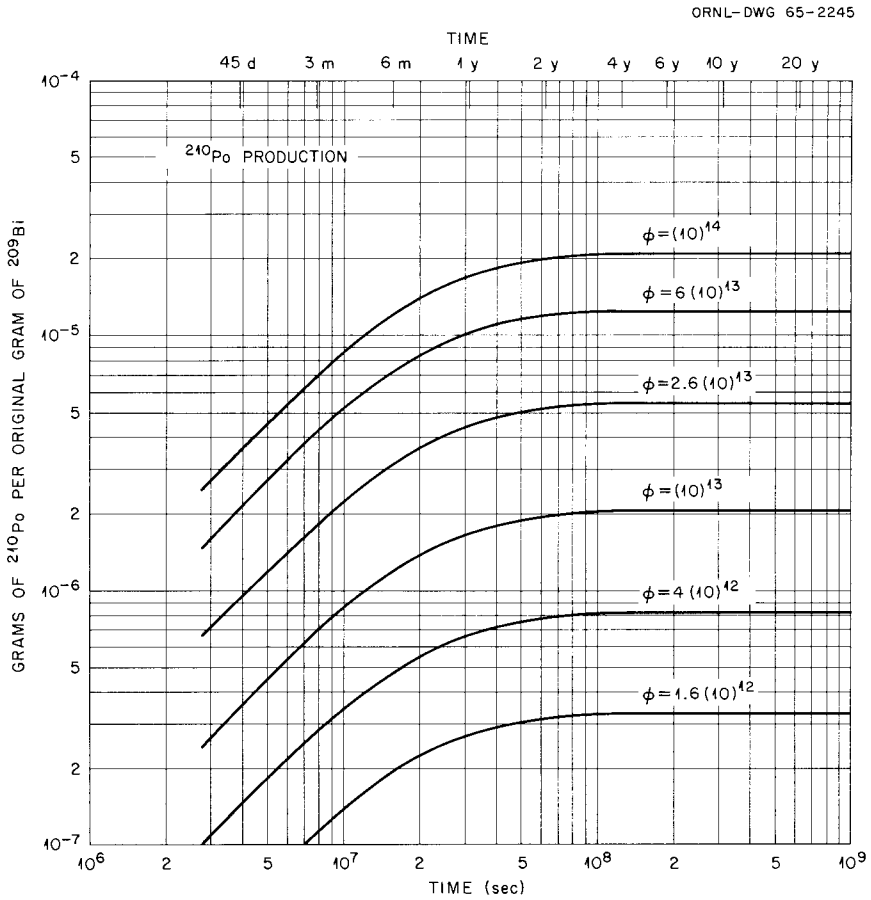

Fig. A.8. Production of ^{210}Po from ^{209}Bi.

PLUTONIUM-238

Figure A.9 gives the reactions employed in calculating the production of ^{238}Pu from ^{237}Np. The results are shown in Fig. A.10, and the input data are listed in Table A.5.

ORNL-DWG 65-2246

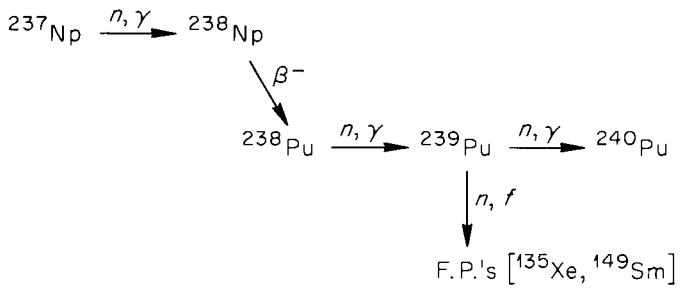

Fig. A.9. Production of ^{238}Pu from ^{237}Np.

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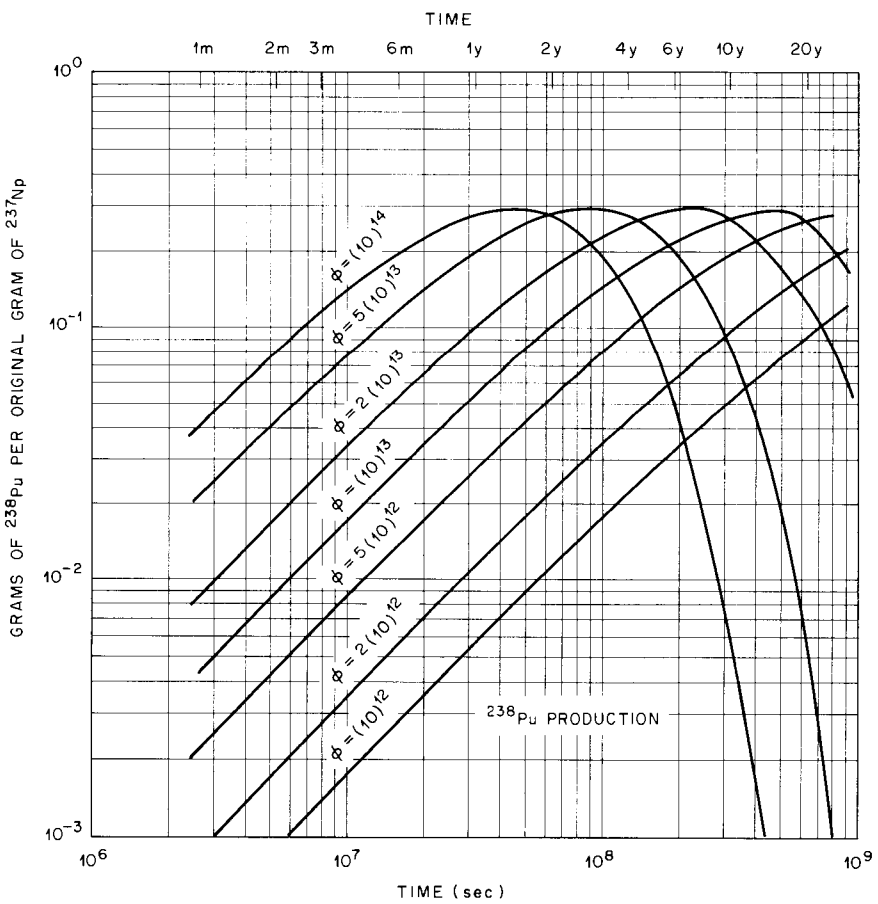

Fig. A.10. Production of ^{238}Pu from ^{237}Np.

Table A.5. CRUNCH Input Data for ^{238}Pu

Isotope	Half-Life	λ (sec^{-1})	$(\sigma_{act})_{2200}$ (barns)	$(\sigma_a)_{2200}$ (barns)	$(\sigma_f)_{2200}$ (barns)	$\bar{\sigma}_a$ (barns)	$\bar{\sigma}_f$ (barns)	$\bar{\sigma}_c$ (barns)	$\bar{\eta}$ ($_0n$/fission)
^{237}Np	2.14×10^6y	~0	170	172	0.016	181	0.010	181	2.96
^{238}Np	2.1d	3.83×10^{-6}			1600		1024	0	3.0
^{238}Pu	89y	2.47×10^{-10}	403		16.6		11	258	3.0
^{239}Pu	24,360y	~0	315	1,029	742	1,174	761	413	2.9
^{240}Pu	6,760y	~0	250	278		994		994	
^{135}Xe	9.2h	2.09×10^{-5}		2.7×10^6		1.8×10^6		1.8×10^6	
^{149}Sm	Stable			41,400		11,480		11,480	
F.P.				80		51		51	

CURIUM-242

The production scheme for calculating ^{242}Cm production from ^{241}Am is shown in Fig. A.11. The existence of metastable 152-year ^{242}Am was taken into consideration in the following manner. The total absorption cross section of ^{241}Am is calculated by Westcott's method, so the fission cross section must be subtracted; and then fractions thereof are used for the production of the two possible ^{242}Am isomers. The activation cross section for 16-hour ^{242}Am has been found to be 750 barns; and for 152-year ^{242m}Am it has been measured as 50 barns, using reactor neutrons.[3] Therefore, the fraction of captures resulting in 16-hr ^{242}Am was assumed to be 750/800 = 0.94. By multiplying the Westcott capture cross section by 0.94, a cross section for use in the ^{241}Am(n,γ) 16-hr ^{242}Am reaction was obtained. A factor of 0.06 was used to obtain the cross section for the formation of 152-year ^{242}Am.

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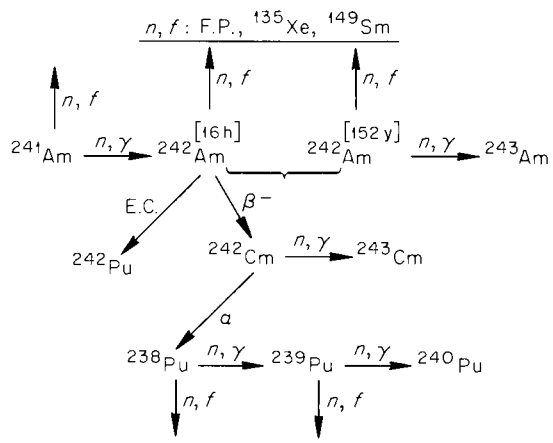

Fig. A.11. Production of ^{242}Cm from ^{241}Am.

Table A.6. CRUNCH Input Data for ^{242}Cm

Isotope	Half-Life	λ (sec^{-1})	$(\sigma_{act})_{2200}$ (barns)	$(\sigma_a)_{2200}$ (barns)	$(\sigma_f)_{2200}$ (barns)	$\bar{\sigma}_a$ (barns)	$\bar{\sigma}_f$ (barns)	$\bar{\sigma}_c$ (barns)	$\bar{\eta}$ ($_0n$/fission)
^{241}Am	458y	~0	16h, 750 152y, 50	584	3.6	710	4.04	706[a]	3.1
^{242}Am	152y	~0		8,000	6400	15,120	4,096	1,024	3.1
^{242}Am	16h	1.20×10^{-5}			3000		1,920		3.1
^{243}Am	7,650y	~0		169		186		186	
^{242}Cm	163d	4.92×10^{-8}	20					12.8	
^{238}Pu	89y	2.47×10^{-10}	403		16.8		11	258	3.0
^{239}Pu	24,360y	~0	315	1,029	742	1,174	761	413	2.9
^{135}Xe	9.2h	2.09×10^{-5}		2.7×10^6		1.8×10^6		1.8×10^6	
^{149}Sm	Stable	0		41,400		11,480		11,480	
F.P.				80		51		51	

[a]Total: for 16h ^{242}Am, $\bar{\sigma}_c$ = 664; for 152y ^{242}Am, $\bar{\sigma}_c$ = 42.4.

The desired product, ^{242}Cm, is formed by beta decay of 16-hr ^{242}Am; but only 80% of the decays are by beta emission – the other 20% are electron-capture events resulting in ^{242}Pu (ref. 1). A loss of 16-hr ^{242}Am nuclei equivalent to 0.2 λ was obtained in the computer calculation by introducing a fictitious capture cross section such that $0.2\ \lambda = \bar{\sigma}_c \phi$. The metastable 152-year ^{242m}Am exhibits alpha decay and until 1960 was thought to be the ground state.[1,10] The isomeric transition was previously thought to make up less than 6% of the loss of 152-year ^{242m}Am. Because of these uncertainties, no isomeric transitions were included in the calculations.

The input data and sources are listed in Table A.6, and the results are presented in Fig. A.12.

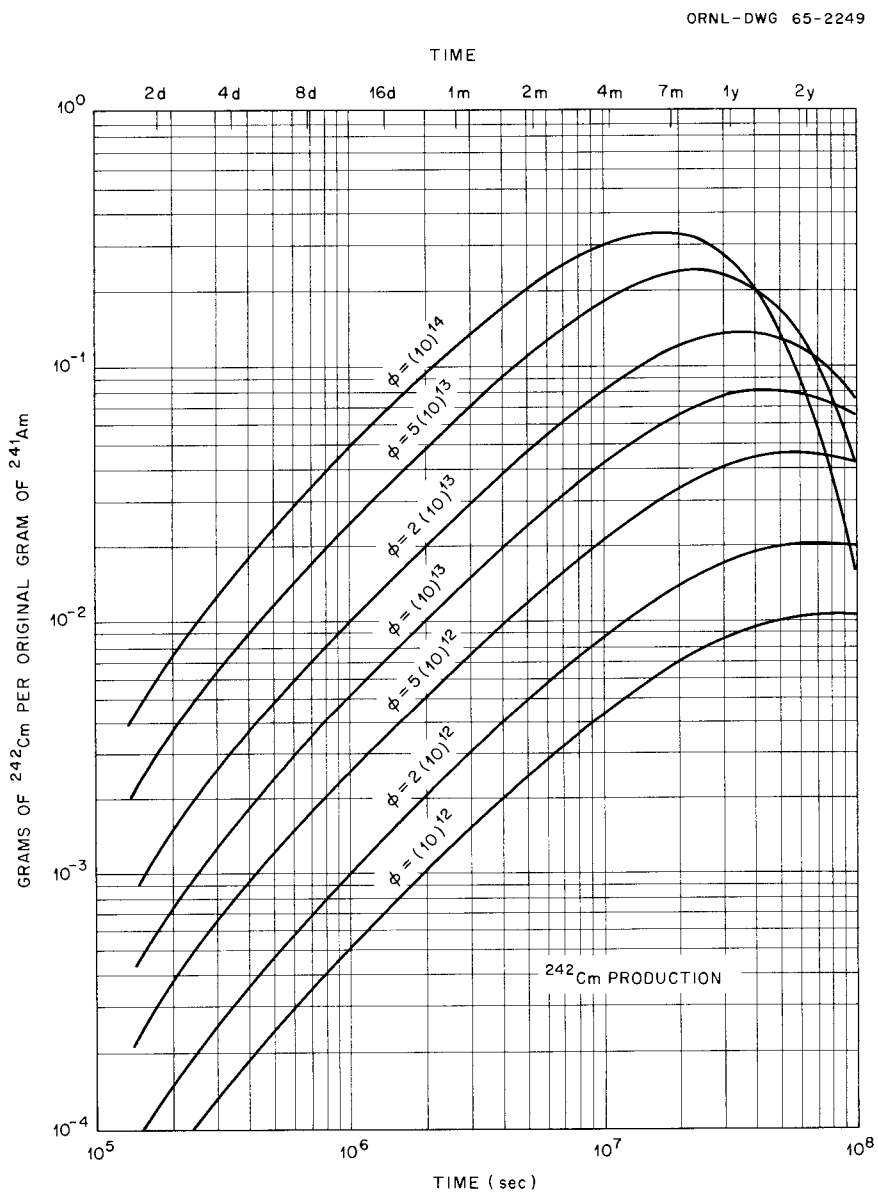

Fig. A.12. Production of ^{242}Cm from ^{241}Am.

CURIUM-244

The reactions considered in calculating ^{244}Cm production from ^{239}Pu are shown in Fig. A.13. The formation of 152-year ^{242m}Am was calculated as explained in the preceding paragraph. It was assumed that the isotope is stable and that no isomeric transitions to 16-hr ^{242}Am occur. Two isomers are also possible in the production of ^{244}Am. Since both have such short half-lives, it might be permissible to assume that all ^{243}Am goes directly to ^{244}Cm, with a cross section equal to that for ^{243}Am (n,γ). This would have been done if it were not for the large fission cross section of the 10.1-hr isomer of ^{244}Am. For values of flux used in this study, the error in neglecting this fission loss would have been ~1%. However, at a flux of 10^{15} neutrons cm^{-2} sec^{-1}, the error would be 10%. The formation of both isomers was considered in this study and was treated in a manner similar to that employed in the case of the ^{242}Am isomers.

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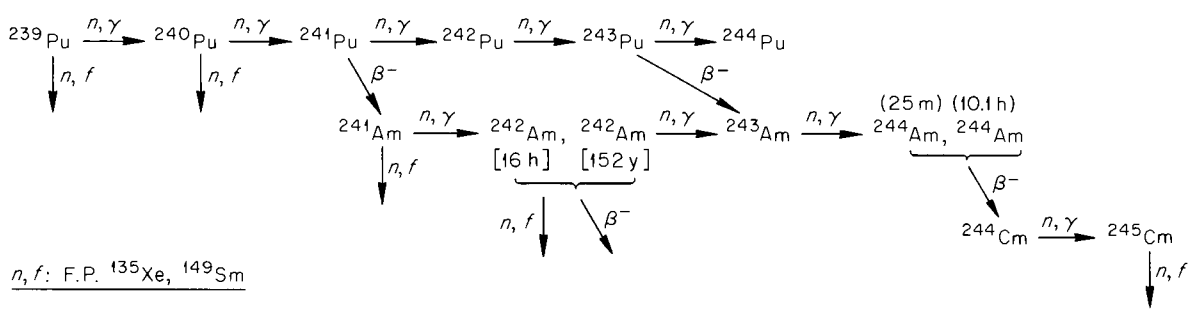


Fig. A.13. Production of ^{244}Cm from ^{239}Pu.

Table A.7. CRUNCH Input Data for ^{244}Cm

Isotope	Half-Life	λ (sec^{-1})	$(\sigma_{act})_{2200}$ (barns)	$(\sigma_a)_{2200}$ (barns)	$(\sigma_f)_{2200}$ (barns)	$\bar{\sigma}_a$ (barns)	$\bar{\sigma}_f$ (barns)	$\bar{\sigma}_c$ (barns)	$\bar{\eta}$ ($_0n$/fission)
^{239}Pu	2.44×10^4y	~0	315	1,029	742	1,174	761	413	2.9
^{240}Pu	6760y	~0	250	278		994		994	
^{241}Pu	13y	1.69×10^{-9}	390	1,397	1015	1,171	851	320	3.06
^{242}Pu	3.79×10^5y	~0		30.9		112		112	
^{243}Pu	4.98h	3.87×10^{-5}	170			107		107	
^{241}Am	458y	~0	16h, 750[a]	584	3.6	710	4.04	706	3.1
^{242}Am	152y	~0		8,000	6400	5,120	4096	1,024	3.1
^{243}Am	7650y	~0	25m, 74[b]	169		186		186	
^{244}Am	25m	4.62×10^{-4}							
^{244}Am	10.1h	1.91×10^{-5}			2300		1449		?
^{244}Cm	17.6y	1.25×10^{-9}	15(2×10^4y)			9.5		9.5	
^{245}Cm	9320y	~0	200(6.6×10^3y)		1900		1197	126	?
^{135}Xe	9.2h	2.09×10^{-5}		2.7×10^6		1.8×10^6		1.8×10^6	
^{149}Sm	Stable	0		41,400		11,480		11,480	
F.P.				80		51		51	

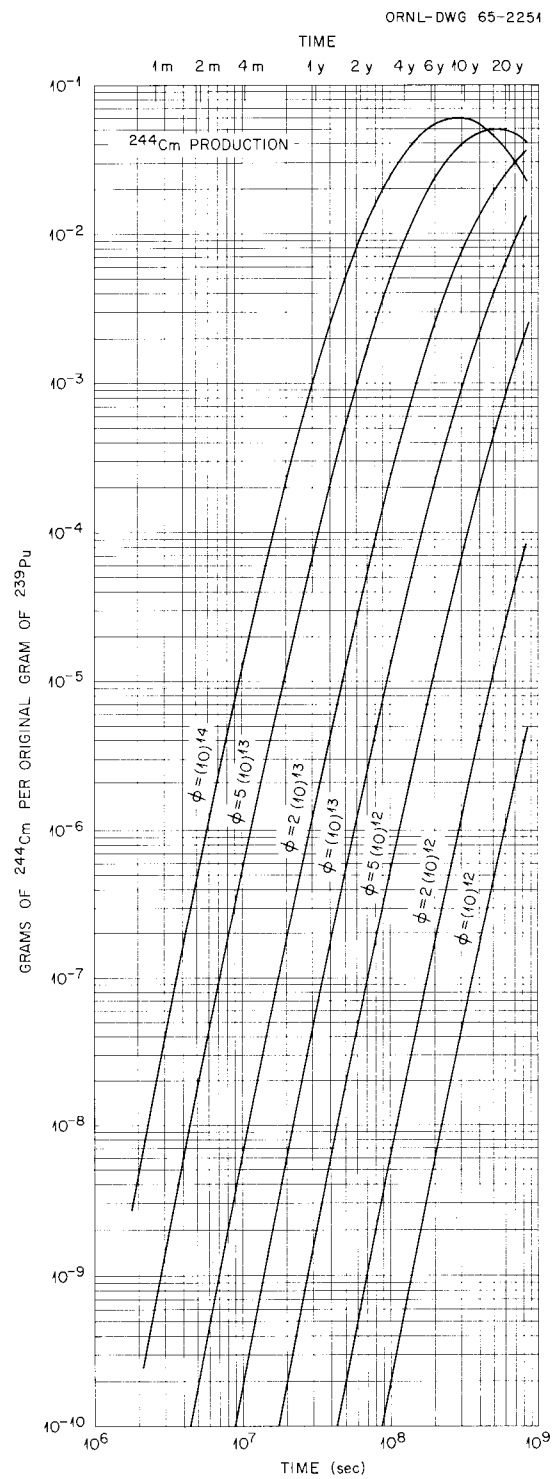
[a] 152y, 50.

[b] σ_{act} 25m/σ_{act} 10.1h = 18.6.

The ratio of the thermal-neutron activation cross section for producing 25-min ^{244}Am to the activation cross section for producing 10.1-hr ^{244}Am has been measured as 18.6 (±1.9):1 (ref. 4). The capture cross section used for the formation of 10.1-hr ^{244}Am is then 5% of the total and for formation of 25-min ^{244}Am, it is 95% of the total.

Input data for the calculation are listed in Table A.7 and the results are presented in Fig. A.14.

Fig. A.14. Production of ^{244}Cm from ^{239}Pu.



TARGET SITES

An examination of the design details of the 18 reactors considered has revealed in each case a number of places in which target materials could be inserted. These are listed in Tables A.8 to A.25. The information in these tables was obtained from the hazards evaluation reports for the various reactors; and, in general, no attempt was made to seek further details. The documents used are listed in refs. 11 to 30.

Table A.8. Possible Target Sites for the BONUS Reactor[a]

Location	Number	Geometry	Volume	Original Material Occupying Site
Boiler fuel assembly	64	32 cylindrical rods/assembly 0.450 in. ID 54 in. long	8.6 in.^{3}/rod	4 assemblies contain natural uranium oxide; 60 assemblies contain UO_2 enriched to 2.4 wt % ^{235}U
Superheater fuel assembly	32	32 cylindrical rods/assembly 0.500 in. ID 54 in. long	10.6 in.^{3}/rod	UO_2 enriched to 3.25 wt % ^{235}U
Shim assemblies				Boron-SS and/or SS

[a]Average thermal-neutron flux in the fuel: 1.5×10^{13} neutrons cm^{-2} sec^{-1}.

Table A.9. Possible Target Sites for the La Crosse Boiling-Water Reactor[a]

Location	Number	Geometry	Volume	Original Material Occupying Site
Fuel elements	288	25 rods/element 0.356 in. ID 83 in. long	8.26 in.^{3}/rod	UO_2 pellets
Structural spacer holes	32	0.310 in. in diameter 83 in. long	6.26 in.^{3}/hole	H_2O
Central control rod followers	9	Cruciform 0.312 in. thick 9.75-in. blade width 83 in. long	497 in.^{3}/follower	Zircaloy-2
Core periphery		Arbitrary		H_2O

[a]Average thermal-neutron flux in the core: 1.9×10^{13} neutrons cm^{-2} sec^{-1}.

Table A.10. Possible Target Sites for the Pathfinder Atomic Power Plant[a]

Location	Number	Geometry	Volume	Original Material Occupying Site
Superheater fuel assemblies	415 2 fuel tubes/assembly	Outer tube fuel OD: 0.814 in. Inner tube fuel OD: 0.605 in. Fuel thickness: 0.020 in. Active fuel length: 71.5 in.	3.144 in.^{3}/ assembly	UO_2 in 316L SS (cermet mixture)
Boiler fuel assemblies	96 81 fuel rods/assembly	4 rod sections/rod 0.352 in. ID (lower sections) 0.315 in. ID (upper sections) 18 in. long/section	1.75 in.^{3}/ lower section 1.40 in.^{3}/ upper section	UO_2 pellets
Poison shims (boiler region)	56	0.10 in. thick 2.78 in. wide 74 in. long	20.57 in.^{3}/ plate	0.20% boron in 304 SS
Superheater burnable poison rods	415	0.4025 in. in diameter 72.25 in. long	9.19 in.^{3}/rod	0.35% boron carbide in alumina

[a]Average thermal-neutron flux in the fuel: 3.1×10^{13} neutrons cm^{-2} sec^{-1}.

Table A.11. Possible Target Sites for the Humboldt Bay Power Plant[a]

Location	Number	Geometry	Volume	Original Material Occupying Site
Fuel locations	172 assemblies 49 rods/assembly	Cylindrical 0.425 in. ID 79 in. long	11.2 in.^{3}/rod	UO_2 enriched to 2.6% ^{235}U
Control rod blades	40	Cruciform made of small SS tubes		B_4C powder packed to 65–75% theoretical density
Control rod follower	32	Cruciform 4.25 in. span 0.25-in.-thick blades 67 in. long	71.2 in.^{3}/follower	Zircaloy-2
Poison curtains	96	Plates 1/16 in. thick 9 in. wide 80 in. long	45.0 in.^{3}/curtain	0.09% boron in SS
Irregularly shaped voids on periphery of core		Arbitrary	Arbitrary	H_2O

[a]Average thermal-neutron flux in the fuel: 1.5×10^{13} neutrons cm^{-2} sec^{-1}.

Table A.12. Possible Target Sites for the Dresden Nuclear Power Station[a]

Location	Number	Geometry	Volume	Original Material Occupying Site
Fuel assemblies	288 assemblies 10,351 rods	4 elements/rod 0.502 in. element ID 26 in. long	5.15 in.^{3}/element	UO_2 pellets
Channel spacers	~160	~0.5 in. thick ~2.0 in. wide ~100 in. long	100 in.^{3}/spacer	Zircaloy-2
Core periphery		Arbitrary		H_2O

[a]Average thermal-neutron flux in the fuel: 3.05×10^{13} neutrons cm^{-2} sec^{-1}.

Table A.13. Possible Target Sites for the Big Rock Point Plant[a]

Location	Number	Geometry	Volume	Original Material Occupying Site
Fuel assemblies	84	132 "standard" rods/assembly 0.350 in. ID 70 in. active length	6.7 in.^{3}/rod	UO_2 pellets
		8 "regular corner" rods/assembly 0.288 in. ID 70 in. active length	4.6 in.^{3}/rod	
		4 "segmented corner" rods/assembly 0.288 in. ID 59 in. active length	3.8 in.^{3}/rod	
Between control rod blade tips	29	~58 in. square 70 in. long	~27 in.^{3}/position	H_2O
Core periphery		Arbitrary		H_2O

[a]Average thermal-neutron flux in the fuel: 2.1×10^{13} neutrons cm^{-2} sec^{-1}.

Table A.14. Possible Target Sites for the RCPA Elk River Reactor[a]

Location	Number	Geometry	Volume	Original Material Occupying Site
Fuel assemblies	148	25 tubes/assembly 0.410 in. ID 60 in. long	7.9 in.^{3}/tube	ThO_2-UO_2
Fuel assembly center pins	40	1 pin/assembly 0.410 in. ID 60 in. long	7.9 in.^{3}/tube	1.5 wt % boron in SS or ThO_2-UO_2
"Extra" fuel assembly positions	16	25 tubes/assembly 0.410 in. ID 60 in. long	7.9 in.^{3}/tube	Unknown
Control rod followers	13	Cruciform 14.5-in. blade width 0.25 in. thick 45 in. long	323.4 in.^{3}/follower	Zircaloy-2
Core periphery		Arbitrary	Arbitrary	H_2O

[a]Average thermal-neutron flux in the fuel: 1.1×10^{13} neutrons cm^{-2} sec^{-1}.

Table A.15. Possible Target Sites for the 9 Mile Point Nuclear Station[a]

Location	Number	Geometry	Volume	Original Material Occupying Site
Fuel assemblies	500	36 rods/assembly 0.464 in. ID	24.35 in.^{3}/rod	UO_2
Control curtains	216	9 in. wide 144 in. long		0.3% boron in 216 SS
Core periphery		Arbitrary		H_2O

[a]Average thermal-neutron flux in the fuel: 2.6×10^{13} neutrons cm^{-2} sec^{-1}.

Table A.16. Possible Target Sites for the Shippingport Reactor[a]

Location	Number	Geometry	Volume	Original Material Occupying Site
Seed fuel assemblies	32	Plate-type element design; 15 plates/subassembly; 4 subassemblies/assembly 0.069 in. thick 2.50 in. wide 72.4 in. long (includes Zr clad)	12.489 in.^{3}/plate	Zr-clad uranium enriched to 93% ^{235}U
Blanket fuel assemblies	113	7 fuel bundles per assembly; 121 cylindrical tubes per bundle 0.411 in. ID 9.5 in. long	1.26 in.^{3}/tube	UO_2 composed of natural uranium
Positions around core barrel	Arbitrary	Arbitrary cross section; ~72 in. long	Arbitrary	H_2O

[a]Average thermal-neutron flux in the fuel: 4.4×10^{13} neutrons cm^{-2} sec^{-1}.

Table A.17. Possible Target Sites for the Oyster Creek Reactor[a]

Location	Number	Geometry	Volume	Original Material Occupying Site
Fuel locations	532 assemblies 64 rods/assembly	Cylindrical 0.434 in. ID 144 in. long	21.3 in.^{3}/rod	UO_2 enriched to 2.1% ^{235}U
Peripheral core positions	Arbitrary	Arbitrary cross section 144 in. length	Arbitrary	H_2O
Control rods	129	Cruciform made of stainless steel tubes		B_4C packed to 65–75% theoretical density
Poison curtains	232	Plates of stainless steel 10 in. wide 140 in. long		0.3% boron in SS

[a]Average thermal-neutron flux in the fuel: 3.6×10^{13} neutrons cm^{-2} sec^{-1}.

Table A.18. Possible Target Sites for the Malibu Nuclear Plant[a]

Location	Number	Geometry	Volume	Original Material Occupying Site
Fuel assemblies in central core region	52	180 cylindrical rods/assembly 0.379 in. ID 120 in. long	13.5 in.^{3}/rod	UO_2 (10.5 g/cm^3) enriched to 3.4 wt % ^{235}U
Fuel assemblies in intermediate core region	52	180 cylindrical rods/assembly 0.379 in. ID 120 in. long	13.5 in.^{3}/rod	UO_2 (10.5 g/cm^3) enriched to 3.8 wt % ^{235}U
Fuel assemblies in outer core region	52	180 cylindrical rods/assembly 0.379 in. ID 120 in. long	13.5 in.^{3}/rod	UO_2 (10.5 g/cm^3) enriched to 4.2 wt % ^{235}U
Positions around outside of core barrel	Arbitrary	Arbitrary cross section 120 in. long	Arbitrary	H_2O

[a]Average thermal-neutron flux in the fuel: 2.2×10^{13} neutrons cm^{-2} sec^{-1}.

Table A.19. Possible Target Sites for the Consolidated Edison Thorium Reactor[a]

Location	Number	Geometry	Volume	Original Material Occupying Site
Fuel elements	120	195 rods/element 0.263 in. ID 98.5 in. long	5.35 in.^{3}/rod	UO_2-ThO_2 pellets
Fixed shim rods	24 (max)	Cruciform 10.00-in. blade width 0.125 in. thickness 98.5 in. long	244.7 in.^{3}/rod	1.0% boron in SS
Fixed filler rods	24 (max)	(Same as shim rods)	244.7 in.^{3}/rod	Zircaloy-2
Flux depressors	12 (T-shaped) 8 (L-shaped) 4 (cruciform)	(Dimensions are the same as shim rods)	183.5 in.^{3} (T) 122.3 in.^{3}(L) 244.7 in.^{3} (+)	304 SS
Core periphery		Arbitrary		H_2O

[a]Average thermal-neutron flux in the fuel: 1.7×10^{13} neutrons cm^{-2} sec^{-1}.

Table A.20. Possible Target Sites for the Yankee Nuclear Power Station[a]

Location	Number	Geometry	Volume	Original Material Occupying Site
Fuel assemblies	76	23,142 cylindrical rods 0.298 in. ID 90 in. long	6.28 in.^{3}/rod	UO_2 packed to a density of 10.16 g/cm^3 U enriched to 3.4% ^{235}U
Reflector	Arbitrary	Arbitrary	Arbitrary	H_2O
Permanent shims	8	Cruciform 7.865-in. span 0.265-in.-thick blades	362.5 in.^{3}/shim	1.2 wt % natural boron in SS
Ferrules		Cylindrical; $\frac{1}{2}$ in. long		348 SS
Instrument positions	76	1 per fuel assembly 0.298 in. ID 90 in. long	6.28 in.^{3}/rod	Instruments

[a]Average thermal-neutron flux in the fuel: 3.1×10^{13} neutrons cm^{-2} sec^{-1}.

Table A.21. Possible Target Sites for the Saxton Reactor[a]

Location	Number	Geometry	Volume	Original Material Occupying Site
Special removable fuel rods	2/fuel assembly (21 fuel assemblies)	Cylindrical 0.361 in. ID 36.6 in. long	3.74 in.^{3}/rod	UO_2 enriched to 5.7 wt % ^{235}U
Removable fuel subassemblies	1/special assembly (8 special assemblies) 9 rods/subassembly	Cylindrical rods 0.361 in. ID 36.6 in. long	3.74 in.^{3}/rod	UO_2 enriched to 5.7 wt % ^{235}U
L-shaped fuel subassemblies	7 subassemblies 9 rods/subassembly	Cylindrical rods 0.361 in. ID 36.6 in. long	3.74 in.^{3}/rod	UO_2 enriched to 5.7 wt % ^{235}U
Dummy fuel assemblies	11	Parallelepiped $5.39 \times 5.39 \times 50.25$ in.	1460 in.^{3}/ assembly	304 SS
Control rod followers	6	Cruciform, constructed of 18 rods/follower 0.361 in. ID		UO_2 enriched to 5.7 wt % ^{235}U

[a]Average thermal-neutron flux in the fuel: 1.2×10^{13} neutrons cm^{-2} sec^{-1}.

Table A.22. Possible Target Sites for the Connecticut Yankee Reactor[a]

Location	Number	Geometry	Volume	Original Material Occupying Site
Fuel assemblies	157	34,069 cylindrical rods 0.3795 in. ID 120 in. long	13.6 in.^{3}/rod	UO_2 with equal amounts of 3.4, 3.8, and 4.2 wt % ^{235}U
Control rod guide tube assemblies	77	16 cylindrical rods/ assembly 120 in. long		H_2O when no fuel or control material occupies position; fuel may be of the three enrichments listed above
Irregularly shaped positions around core periphery		Arbitrary cross section 120 in. long	Arbitrary	H_2O

[a]Average thermal-flux in the fuel: 1.9×10^{13} neutrons cm^{-2} sec^{-1}.

Table A.23. Possible Target Sites for the Southern California Edison Reactor[a]

Location	Number	Geometry	Volume	Original Material Occupying Site
Fuel assemblies	156	189 cylindrical rods/ assembly, 0.3755 in. ID 120 in. long	13.3 in.^{3}/rod	UO_2 with equal parts of 3.2, 3.6, 4.0 wt % ^{235}U
Positions between barrel and core		Arbitrary cross section 120 in. long	Arbitrary	H_2O
Control rod followers	37	Offset cruciform		Zircaloy

[a]Average thermal-neutron flux in the fuel: 1.9×10^{13} neutrons cm^{-2} sec^{-1}.

Table A.24. Possible Target Sites for the Carolinas-Virginia Tube Reactor[a]

Location	Number	Geometry	Volume	Original Material Occupying Site
Fuel locations	72 assemblies 19 rods/assembly	Cylindrical 0.4445 in. ID 95.4 in. long	14.8 in.^{3}/rod	UO_2 enriched to 2.0 wt % ^{235}U or 1.5 wt % ^{235}U
Unused pressure tubes	6 tubes	Cylindrical 3.66 in. ID 95.4 in. long	1004 in.^{3}/tube	D_2O
Baffle strips in fuel assemblies				Zircaloy-4
Flow, baffles	42 pressure tubes 2 baffles/pressure tube	Inner baffle 3.232 in. ID, 0.012 in. thick, 95.4 in. long	11.7 in.^{3}/baffle	Zircaloy-4
		Outer baffle 3.338 in. ID, 0.012 in. thick, 95.4 in. long	12.0 in.^{3}/baffle	Zircaloy-4
Control rods	28	Rectangular parallepiped $1 \times 4 \times 95$ in.		1.5 wt % boron in SS

[a]Average thermal-neutron flux in the fuel: 2.4×10^{13} neutrons cm^{-2} sec^{-1}.

Table A.25. Possible Target Sites for the Hallam Nuclear Power Facility[a]

Location	Number	Geometry	Volume	Original Material Occupying Site
Fuel assemblies	137	18 rods/assembly 0.640 in. ID 13.5 ft long	52.1 in.^{3}/rod	U-Mo alloy
Central tubes of fuel assembly	137	1 tube/assembly 0.640 in. ID 13.5 ft long	52.1 in.^{3}/rod	Sodium
Dummy assemblies	45	19 tubes/assembly 0.640 in. ID 13.5 ft long	52.1 in.^{3}/rod	Unknown
Reflector filler assemblies	75	19 tubes/assembly 0.640 in. ID 13.5 ft long	52.1 in.^{3}/rod	Unknown
Holes drilled in graphite "logs"		Arbitrary		Graphite
Core periphery		Arbitrary		Sodium

[a]Average thermal-neutron flux in the fuel: 1.0×10^{13} neutrons cm^{-2} sec^{-1}.

Table A.26. Data Used in Production Calculations

	Reactor Thermal Power (Mw)	Weight of ^{235}U (kg)	Enrichment (wt %)	Coolant/ Moderator	Average Thermal-Neutron Flux in the Fuel (n cm^{-2} sec^{-1})	Effective Thermal-Neutron Flux in the Target (n cm^{-2} sec^{-1})	Number of Available Fuel Rods	Number of Target Rods	Volume per Rod (cm^3)	Available Target Volume per Rod (cm^3)	Total Target Volume (cm^3)	Rod ID (cm)
					$\times 10^{13}$	$\times 10^{13}$						
Big Rock Point	240	374.8	3.2	H_2O	2.1	1.69	11,088	332	109.8	87.8	29,160	0.889
BONUS	50	111.3	nat, 2.4, 3.3	H_2O	1.5	1.2	2,048	61	140.9	112.7	6,875	1.143
CVTR	64.9	56.9	1.5, 2.0	D_2O	2.4	1.92	1,368	41	242.5	194.0	7,954	1.129
Conn. Yankee	1473	2579	3.4 to 4.2	H_2O	1.9	1.52	34,069	1,022	222.9	178.3	182,200	0.964
Dresden	630	782.5	1.5	H_2O	3.0	2.44	10,351	310	337.6	270.1	83,720	1.275
Elk River	58.2	168.9	93.13	H_2O	1.1	0.885	3,700	111	129.5	103.6	11,500	1.041
Hallam	240	994.7	3.6	Na/C	1.0	0.80	2,466	74	853.8	683.0	50,500	1.626
Humboldt	165	359.6	2.6	H_2O	1.5	1.2	8,428	252	183.5	146.8	38,620	1.079
Indian Point	585	1100	90	H_2O	1.7	1.35	23,400	702	87.7	70.1	49,240	0.668
La Crosse	165	292.4	3.4	H_2O	1.9	1.49	7,200	216	135.4	108.3	23,390	0.904
Malibu	1473	2169.8	3.4 to 4.2	H_2O	2.2	1.76	28,080	842	221.2	177.0	149,000	0.963
9-Mile Point	1538	2003	2.1	H_2O	2.6	2.06	18,000	540	399.0	319.2	172,400	1.179
Oyster Creek	1600	2171	2.1	H_2O	2.4	1.92	34,048	1,020	349.0	279.2	285,100	1.102
Pathfinder	188.9	194.3	2.2, 93	H_2O	3.1	2.48	7,776	233	103.3	82.6	19,260	0.894
Saxton	20	54.1	5.7	H_2O	1.2	0.96	1,512	45	61.3	49.0	2,224	0.917
Shippingport	231	168.4	93	H_2O	4.4	3.52	94,920	2,847	20.6	16.5	46,980	1.044
So. Calif. Edison	1210	2084.4	3.2 to 4.0	H_2O	1.9	1.52	29,484	884	217.9	174.3	154,200	0.954
Yankee	392	708.3	3.4	H_2O	3.1	2.48	23,142	694	102.9	82.3	57,140	0.757

Table A.27. Production of ^{60}Co

	Weight of Original Target Element (kg)	Specific Activity[a] (curies/g)		^{60}Co Production				Production Ratio[b] (mg/g)		Specific Heat Generation[c] (w/g)		Heat Generation (kw)	
		One Year	Two Years	One Years (megacuries)	Two Years (megacuries)	One Year (g)	Two Years (g)	One Year	Two Years	One Year	Two Years	One Year	Two Years
Big Rock Point	60.3	15.0	29.0	0.903	1.747	798	1,545	13	25	0.200	0.385	12.1	23.5
BONUS	11.1	11.0	21.0	0.123	0.234	109	207	9.6	18	0.148	0.277	1.7	3.1
CVTR	12.9	18.0	33.0	0.234	0.425	207	376	16	29	0.246	0.447	3.1	5.8
Conn. Yankee	348.1	13.6	26.5	4.738	9.222	4,189	8,154	12	23	0.185	0.354	63.7	124.0
Dresden	120.8	21.6	42.0	2.611	5.075	2,309	4,487	19	37	0.293	0.570	35.0	68.2
Elk River	20.4	8.0	15.3	0.163	0.311	144	275	7	13	0.108	0.200	2.3	4.2
Hallam	57.2	7.2	14.0	0.412	0.801	364	708	6.3	12	0.097	0.185	5.5	10.8
Humboldt	66.1	10.8	22.0	0.714	1.455	631	1,287	9.4	19	0.145	0.293	9.6	19.6
Indian Point	135.8	12.0	23.5	1.630	3.193	1,441	2,823	10	21	0.154	0.323	21.9	42.9
La Crosse	47.7	13.3	25.6	0.634	1.220	561	1,079	12	22	0.185	0.339	8.6	16.5
Malibu	284.8	15.7	30.5	4.469	8.683	3,951	7,678	14	27	0.216	0.416	60.1	116.8
9 Mile Point	268.4	19.0	37.0	5.104	9.938	4,513	8,787	16.6	32	0.256	0.493	68.5	133.7
Oyster Creek	476.0	17.0	33.0	8.093	15.709	7,156	13,890	14.8	29	0.228	0.447	108.8	211.2
Pathfinder	39.8	22.0	43.0	0.875	1.711	774	1,513	19	38	0.293	0.585	11.8	22.9
Saxton	4.5	8.6	17.0	0.038	0.076	34	67	0.7	1.6	0.011	0.025	0.6	1.0
Shippingport	83.2	30.0	58.0	2.493	4.819	2,204	4,261	26	51	0.400	0.785	33.5	64.7
So. Calif. Edison	299.1	13.5	26.0	4.007	7.716	3,543	6,822	12	23	0.185	0.354	53.9	103.7
Yankee	139.1	22.0	43.0	3.069	5.978	2,714	5,286	19	38	0.293	0.585	41.2	80.3
Total	2475.3			40.310	78.313	35,642	69,245					541.9	1052.9

[a]Curies of ^{60}Co per gram of original target element.
[b]Milligrams of ^{60}Co per gram of original target element.
[c]Watts produced per gram of original target element.

Table A.28. Production of ^{170}Tm

	Weight of Original Target Element (kg)	Specific Activity[a] (curies/g)		^{170}Tm Production				Production Ratio[b] (mg/g)		Specific Heat Generation[c] (w/g)		Heat Generation (kw)	
		One Year	Two Years	One Year (megacuries)	Two Years (megacuries)	One Year (g)	Two Years (g)	One Year	Two Years	One Year	Two Years	One Year	Two Years
Big Rock Point	66.2	103	116	6.82	7.68	1,146	1,291	17	19	0.265	0.296	17.87	20.14
BONUS	12.2	74	85	0.90	1.03	152	173	12	14	0.187	0.218	2.36	2.70
CVTR	14.2	118	132	1.67	1.87	281	314	20	22	0.312	0.343	4.38	4.90
Conn. Yankee	380	94	106	35.73	40.19	6,002	6,753	16	18	0.250	0.281	93.63	105.34
Dresden	133	149	165	19.78	21.91	3,323	3,680	25	28	0.390	0.437	51.85	57.40
Elk River	22.3	55	63	1.23	1.41	206	236	9.2	11	0.144	0.172	3.22	3.69
Hallam	62.3	50	57	3.11	3.55	523	596	8.4	9.6	0.131	0.150	8.15	9.29
Humboldt	71.8	74	85	5.31	6.10	892	1,024	12	14	0.187	0.218	13.92	15.98
Indian Point	149	83	95	12.37	14.15	2,077	2,378	14	16	0.218	0.250	32.41	37.09
La Crosse	52.2	92	103	4.80	5.38	807	904	15	17	0.234	0.265	12.59	14.10
Malibu	313	108	121	33.77	37.84	5,675	6,360	18	20	0.281	0.312	88.55	99.25
9 Mile Point	293	127	143	37.27	41.96	6,260	7,012	21.3	24	0.332	0.374	97.64	110.11
Oyster Creek	519	120	134	62.74	70.06	10,541	11,773	20.2	26	0.315	0.406	164.47	183.64
Pathfinder	43.5	150	168	6.53	7.31	1,096	1,228	25	28	0.390	0.437	17.10	19.16
Saxton	4.9	6	7	0.03	0.03	5	5	1.0	1.1	0.016	0.017	0.08	0.08
Shippingport	90.9	210	220	19.10	20.00	3,211	3,361	35	37	0.546	0.577	50.09	52.67
So. Calif. Edison	325	94	105	30.65	34.23	5,151	5,752	16	18	0.250	0.281	80.39	89.70
Yankee	152	150	170	22.79	25.80	3,829	4,334	25	29	0.390	0.452	59.74	67.61
Total	2704.5			304.60	340.50	51,177	57,174					798.44	892.85

[a]Curies of ^{170}Tm per gram of original target element.
[b]Milligrams of ^{170}Tm produced per gram of original target element.
[c]Watts produced per gram of original target element.

Table A.29. Production of ^{14}C

	Weight of Original Target Element (kg)	Product Concentration[a] (mc/g)		^{14}C Production				Production Ratio[b] (mg/g)	
		One Year	Two Years	One Year (curies)	Two Years (curies)	One Year (g)	Two Years (g)	One Year	Two Years
Big Rock Point	22.9	3.88	5.93	88.8	136	19.5	29.8	0.85	1.30
BONUS	5.6	2.36	4.77	13.3	27	2.9	5.9	0.52	1.00
CVTR	6.5	3.72	7.56	24.1	49	5.3	10.8	0.82	1.70
Conn. Yankee	148.3	2.95	6.06	438	899	96.4	198	0.65	1.30
Dresden	65.8	4.79	9.58	315	630	69.1	138	1.05	2.10
Elk River	9.0	1.76	3.51	15.9	32	3.5	7.0	0.38	0.77
Hallam	41.2	1.59	3.18	65.4	131	14.4	28.8	0.35	0.70
Humboldt	31.4	2.35	4.77	74.0	150	16.3	33.0	0.52	1.00
Indian Point	38.7	2.69	5.38	104.1	208	22.8	45.6	0.59	1.20
La Crosse	18.4	3.92	6.02	72.1	111	15.8	24.3	0.86	1.30
Malibu	117.1	3.47	6.93	405.8	812	89.0	178	0.76	1.50
9 Mile Point	140.4	4.07	8.08	572	1149	126	253	0.90	1.80
Oyster Creek	232.2	3.83	7.75	858	1736	188	380.8	0.84	1.70
Pathfinder	15.1	4.83	9.76	73.1	148	16.0	32.4	1.10	2.10
Saxton	1.8	1.90	3.81	3.4	7	0.8	1.5	0.42	0.84
Shippingport	36.9	7.27	13.6	268	502	58.7	110	1.60	3.00
So. Calif. Edison	125.6	3.01	6.02	365	729	80.0	160	0.66	1.30
Yankee	46.5	4.99	9.76	232	454	51.1	100	1.10	2.10
Total	1103.4			3988.0	7910	875.6	1612.7		

[a]Millicuries of ^{14}C per gram of original target element.

[b]Milligrams of ^{14}C produced per gram of original target element.

Table A.30. Production of ^{210}Po

	Weight Of Original Target Element (kg)	Product Concentration[a] (mc/g)		^{210}Po Production				Production Ratio[b] (μg/g)		Specific Heat Generation[c] (mw/g)		Heat Generation (w)	
		One Year	Two Years	One Year (kilo-curies)	Two Years (kilo-curies)	One Year (g)	Two Years (g)	One Year	Two Years	One Year	Two Years	One Year	Two Years
Big Rock Point	285.8	14.1	16.7	4.02	4.77	0.89	1.06	3.1	3.7	43.7	52.2	125.5	149.5
BONUS	67.9	11.2	13.5	0.76	0.91	0.17	0.20	2.5	3.0	35.2	42.3	24.0	28.8
CVTR	77.9	18.0	21.5	1.40	1.68	0.31	0.37	4.0	4.7	56.4	66.3	43.7	52.2
Conn. Yankee	1786	13.2	17.0	23.60	30.40	5.27	6.79	3.0	3.8	42.3	53.6	743.1	957.4
Dresden	820.4	22.9	27.8	18.79	22.85	4.18	5.09	5.1	6.2	71.9	87.4	589.4	717.7
Elk River	112.7	8.4	9.9	0.95	1.11	0.21	0.25	1.9	2.2	26.8	31.0	29.7	35.0
Hallam	495.3	7.6	9.0	3.77	4.44	0.84	0.99	1.7	2.0	24.0	28.2	118.7	139.7
Humboldt	378.5	11.2	13.5	4.23	5.11	0.95	1.14	2.5	3.0	35.2	42.3	133.4	160.7
Indian Point	482.5	12.9	15.3	6.22	7.37	1.38	1.64	2.9	3.4	40.9	47.9	194.6	231.2
La Crosse	229.2	14.4	17.1	3.29	3.91	0.73	0.87	3.2	3.8	45.1	53.6	103.5	122.8
Malibu	1460	16.6	19.8	24.25	28.88	5.40	6.43	3.7	4.4	52.0	62.0	761	907
9 Mile Point	168.9	19.7	22.9	33.30	38.60	7.43	8.61	4.4	5.1	62.0	71.9	1048	1214
Oyster Creek	2794	18.0	21.6	50.20	60.22	11.20	13.40	4.0	4.8	56.0	68.0	1579	1889
Pathfinder	188.7	23.4	28.3	4.41	5.34	0.98	1.19	5.2	6.3	73.3	88.8	138.3	167.6
Saxton	21.8	8.9	10.7	0.19	0.23	0.04	0.05	2.0	2.4	28.2	33.8	6.2	7.3
Shippingport	460.3	33.0	39.9	15.19	18.37	3.37	4.08	7.4	8.7	104.3	122.7	475.2	575.3
So. Calif. Edison	1510	14.4	17.1	21.74	25.78	4.84	5.74	3.2	3.8	52.0	54.0	682	809
Yankee	560	23.2	27.7	13.00	15.50	2.91	3.47	5.2	6.2	73.3	87.4	410.3	489.3
Total	11,899.9			229.31	275.47	51.10	61.37					7205.6	8653.5

[a]Millicuries of ^{210}Po per gram of original target element.
[b]Micrograms of ^{210}Po per gram of original target element.
[c]Milliwatts produced per gram of original target element.

Table A.31. Production of ^{238}Pu

	Weight of Original Target Element (kg)	Product Concentration[a] (curies/g)		^{238}Pu Production				Production Ratio[b] (g/g)		Specific Heat Generation[c] (mw/g)		Heat Generation (kw)	
		One Year	Two Years	One Year (kilo-curies)	Two Years (kilo-curies)	One Year (kg)	Two Years (kg)	One Year	Two Years	One Year	Two Years	One Year	Two Years
Big Rock Point	38.65	1.39	2.44	53.9	94.1	3.21	5.60	83	145	46.5	81.2	1.80	3.14
BONUS	7.13	1.01	1.85	7.2	13.1	0.43	0.78	60	110	33.6	61.6	0.24	0.44
CVTR	8.31	1.58	2.69	13.1	22.3	0.78	1.33	94	160	52.6	89.6	0.44	0.74
Conn. Yankee	222.55	1.28	2.25	284.4	504.5	16.93	30.03	76	134	42.6	75.0	9.48	16.82
Dresden	77.47	1.92	3.18	148.7	245.8	8.85	14.63	114	189	63.8	105.8	4.96	8.20
Elk River	13.03	0.78	1.46	10.1	19.0	0.60	1.13	46	87	25.8	48.7	0.34	0.64
Hallam	36.64	0.71	1.34	25.9	49.2	1.54	2.93	42	80	23.5	44.8	0.86	1.64
Humboldt	42.16	1.34	1.85	56.6	78.0	3.37	4.64	80	110	44.8	61.6	1.89	2.60
Indian Point	86.79	1.14	2.07	99.1	179.4	5.90	10.68	68	123	38.1	68.9	3.30	5.98
La Crosse	30.51	1.25	2.22	38.0	67.7	2.26	4.03	74	132	41.4	73.9	1.26	2.25
Malibu	182.07	1.45	2.52	263.1	458.8	15.66	27.31	86	150	48.0	84.0	8.73	15.28
9 Mile Point	136.47	1.68	2.86	229.0	389.6	13.63	23.19	100	170	56.0	95.2	7.63	12.98
Oyster Creek	305.27	1.55	2.69	471.7	820.7	28.08	48.85	92	160	52.0	90.0	15.75	27.38
Pathfinder	25.32	1.93	3.19	48.9	80.6	2.91	4.80	115	190	64.4	106.4	1.63	2.70
Saxton	2.86	0.84	1.58	2.4	4.5	0.14	0.27	49	94	27.4	52.6	0.08	0.15
Shippingport	52.98	2.54	4.37	134.7	231.5	8.02	13.78	152	260	85.1	145.6	4.50	7.72
So. Calif. Edison	190.57	1.26	2.27	239.4	430.9	14.25	25.65	75	135	42.0	76.0	7.96	14.34
Yankee	89.09	2.22	3.61	198.2	322.2	11.80	19.18	132	215	73.9	120.4	6.61	10.74
Total	1547.87			2324.4	4011.9	138.36	238.81					77.46	133.74

[a]Curies of ^{238}Pu per gram of original target element.
[b]Grams of ^{238}Pu per gram of original target element.
[c]Milliwatts produced per gram of original target element.

Table A.32. Production of ^{242}Cm

	Weight of Original Target Element (kg)	Product Concentration[a] (curies/g)		^{242}Cm Production				Production Ratio[b] (mg/g)		Specific Heat Generation[c] (w/g)		Heat Generation (kw)	
		Six Months	One Year	Six Months (mega-curies)	One Year (mega-curies)	Six Months (g)	One Year (g)	Six Months	One Year	Six Months	One Year	Six Months	One Year
Big Rock Point	11.25	309.7	389.6	3.477	4.382	1,044	1,316	93	117	11.2	14.0	125	158
BONUS	2.08	233.1	299.7	0.486	0.626	146	188	70	90	8.4	10.8	18	22
CVTR	2.42	243.0	426.2	0.832	1.032	250	310	103	128	12.4	15.4	30	37
Conn. Yankee	65.02	296.4	366.3	19.194	23.773	5,764	7,139	89	110	10.7	13.2	691	857
Dresden	22.70	416.2	499.5	9.451	11.335	2,838	3,404	125	150	15.0	18.0	340	408
Elk River	3.78	176.5	233.1	0.666	0.879	200	264	53	70	16.4	8.4	24	32
Hallam	10.69	159.8	213.1	1.708	2.278	513	684	48	64	5.8	7.7	61	82
Humboldt	12.31	233.1	299.7	2.867	3.673	861	1,103	70	90	8.4	10.8	103	132
Indian Point	25.41	253.1	329.7	6.424	8.368	1,929	2,513	76	99	9.1	11.9	231	301
La Crosse	8.90	279.7	353.0	2.494	3.144	749	944	84	106	10.1	12.7	90	113
Malibu	53.57	323.0	399.6	17.309	21.422	5,198	6,433	97	120	11.6	14.4	625	773
9 Mile Point	50.29	366.3	456.2	18.428	22.987	5,534	6,903	110	137	13.2	16.4	664	828
Oyster Creek	88.97	343.0	432.9	30.589	38.608	9,186	11,594	103	130	12.4	15.6	1103	1392
Pathfinder	7.43	426.2	506.2	3.167	3.760	951	1,129	128	152	15.4	18.2	114	136
Saxton	0.83	119.9	253.1	0.100	0.213	30	64	36	76	4.3	9.1	4	8
Shippingport	15.46	559.1	622.7	8.585	9.607	2,578	2,885	167	187	20.0	22.4	309	346
So. Calif. Edison	55.58	283.0	359.6	15.698	19.940	4,714	5,988	85	108	10.2	13.0	566	720
Yankee	25.96	422.9	506.2	10.982	13.144	3,298	3,947	127	152	15.2	18.2	396	474
Total	462.65			152.457	189.171	45,783	56,808					5494	6819

[a]Curies of ^{242}Cm per gram of original target element.
[b]Milligrams of ^{242}Cm per gram of original target element.
[c]Watts produced per gram of original target element.

Table A.33. Production of ^{244}Cm

	Weight of Original Target Element (kg)	Product Concentration[a] (curies/kg)		^{244}Cm Production				Production Ratio[b] (mg/g)		Specific Heat Generation[c] (mw/g)		Heat Generation (w)	
		Four Years	Eight Years	Four Years (curies)	Eight Years (curies)	Four Years (g)	Eight Years (g)	Four Years	Eight Years	Four Years	Eight Years	Four Years	Eight Years
Big Rock Point	15.7	29.1	241.3	457	3,789	5.48	45.43	0.35	2.9	0.98	8.12	15.35	127.2
BONUS	2.9	7.2	83.1	21	241	0.25	2.89	0.09	1.0	0.24	2.80	0.70	8.1
CVTR	3.4	46.2	354.7	157	1,206	1.88	14.46	0.56	4.3	1.57	12.04	5.26	40.5
Conn. Yankee	90.3	20.0	183.6	1,806	16,582	21.65	198.83	0.24	2.2	0.67	6.16	60.63	556.7
Dresden	31.4	112.7	659.7	3,538	20,716	42.42	248.39	1.35	7.9	3.78	22.12	118.78	695.6
Elk River	5.3	1.5	27.0	8	143	0.10	1.72	0.02	3.3	0.05	9.24	0.28	4.8
Hallam	14.9	1.4	19.1	21	285	0.25	3.42	0.02	2.3	0.05	6.44	0.71	9.6
Humboldt	17.1	7.1	83.4	121	1,426	1.45	17.10	0.08	1.0	0.24	2.80	4.07	47.9
Indian Point	35.4	11.7	123.6	414	4,374	4.96	52.45	0.14	1.5	0.39	42.00	13.89	146.9
La Crosse	12.4	17.8	169.0	221	2,096	2.65	25.13	0.21	2.0	0.59	5.60	7.42	70.4
Malibu	74.6	33.8	272.5	2,525	20,327	30.27	243.73	0.41	3.3	1.15	9.24	84.72	682.6
9 Mile Point	69.9	63.4	433.6	4,434	30,311	53.16	363.44	0.76	5.2	2.13	14.60	148.86	1017.8
Oyster Creek	123.7	58.9	387.4	7,282	47,922	87.32	574.60	0.70	4.6	1.96	12.90	244.50	1608.9
Pathfinder	10.3	117.3	687.0	1,208	7,076	14.48	84.84	1.40	8.2	3.92	23.00	40.56	237.5
Saxton	1.2	2.5	35.8	3	43	0.04	0.52	0.04	0.4	0.10	1.26	0.12	1.5
Shippingport	21.5	343.3	1421.3	7,381	30,557	88.50	366.39	4.11	17.0	11.51	47.6	247.80	1026.0
So. Calif. Edison	77.3	20.0	183.3	1,545	14,171	18.53	169.92	0.24	2.2	0.67	6.20	51.86	475.8
Yankee	36.0	115.1	667.0	4,143	24,013	49.68	287.92	1.38	8.0	3.86	22.40	139.12	805.9
Total	643.3			35,285	225,278	423.07	2701.18					1184.63	7563.7

[a]Curies of ^{244}Cm per kilogram of original target element.
[b]Milligrams of ^{244}Cm per gram of original target element.
[c]Milliwatts produced per gram of original target element.

AN ESTIMATE OF COSTS

Items Included in Cost Estimates

Costs do not include interest, overhead, insurance, or other miscellaneous charges; only direct costs were considered, except where some possible indirect costs had been included in our original data and could not be fully investigated in the time available.

Reactor Costs

The costs of a full-scale radioisotope program in the Malibu Reactor have been estimated and compared with the major cost differences at a similar boiling-water reactor. These costs are based upon irradiating targets inside the fuel region since this results in the greatest production; and, while some production could be obtained in the peripheral regions, instrument channels, and miscellaneous space available in various reactors, it would be relatively minor. The Malibu Reactor was chosen as an example because it will use a control-rod-cluster fuel arrangement in which those fuel assemblies not containing a control rod have a number of empty spaces into which targets could be inserted. This appears to be the only present-day power reactor design easily adapted to an isotope production program without removal of fuel and, consequently, a reduction of power. For comparison of reactor costs, the Oyster Creek Reactor was chosen as typical of reactors where a considerable amount of redesign would be required together with a reduction in power due to the necessity for removing 3% of the fuel rods and, consequently, 3% of the heat-transfer area.

Sufficient core data were not available to estimate enrichment costs for the Oyster Creek Reactor and, accordingly, these costs were assumed to be the same as those for the Malibu design. Since the power-loss costs overshadow the enrichment charges, however, the small error in enrichment costs is not significant for the purposes of an overall cost comparison.

Fuel Enrichment Costs

In computing the additional fuel costs resulting from the insertion of radioisotope targets in the reactor, it has been assumed that the fuel cycle will not be changed. The fuel enrichment required to accomplish this is based on the further assumption that the fuel cycle will not change significantly, provided that the thermal utilization remains constant. In the case of the Malibu Reactor, it is not necessary to remove fuel in order to insert target material equivalent to 2.4% of the fuel volume. Thus, no additional enrichment is required in order to compensate for loss of fuel volume.

The original initial enrichment of the Malibu Reactor is 3.8%. No additional enrichment is required where the product under consideration is ^{210}Po or ^{14}C because their respective targets have very low cross sections. In the case of the other isotopes, the enrichment will be 4.3%. The fuel costs associated with the original and modified enrichments are listed in Table A.34. Because in both cases the total amount of fuel is the same, the conversion and fabrication costs are also essentially the same for both cases. Moreover, it turns out that because of the high burnup (24,000 Mwd/ton) the reprocessing costs are the same for both cases. Cost data are given in Tables A.35 to A.39.

Table A.34. Enrichment Costs at the Malibu Reactor

Item	Costs/Calendar Year ($\$ \times 10^3$)		
	Original	Modified	Difference
U depletion	3850	4060	210
Pu credit[a]	(1370)	(1450)	(80)
Fuel use charge	1010	1180	170
Total	3490	3790	300

[a]Includes conversion to metal and $4\frac{3}{4}$% inventory charge.

Table A.35. Cost of Irradiating Isotope Targets in the Malibu Reactor

	Nonrecurring	Recurring ($/year)
Core calculations and heat-transfer studies	$50,000	
Enrichment (for high-absorber targets)		300,000
Handling-tool design and fabrication	35,000[a]	
Incremental costs for labor and material		20,000
Shipment of irradiated targets		10,000
Total	$85,000	330,000

[a]This estimate is for reactors of the control-rod-cluster design in which 16 empty spaces are available in each fuel assembly and only small changes need be made in the mechanical design of the fuel.

Table A.36. Costs of Irradiating Isotope Targets in the Oyster Creek Reactor

	Nonrecurring	Recurring ($/year)
Core calculations and heat-transfer studies	$ 50,000	
Enrichment (for high absorber)[a]		300,000
Loss of power-generating capacity[b]		525,000
Redesign of fuel assemblies	50,000	
Incremental cost of fuel		25,000
Handling-tool design and fabrication	35,000	
Incremental labor and material		20,000
Shipment costs		10,000
Total	$135,000	880,000

[a]Enrichment costs are assumed to be the same as at Malibu.

[b]At 620 Mw (electrical) and 80% plant factor, a 3% loss of generating capacity could amount to as much as $620 \times 0.80 \times 365 \times 0.03 = 5430$ Mwd or 131,000,000 kwh; at 4 mils this would equal $525,000. Since this boiling-water reactor as presently designed cannot accommodate many targets without removal of fuel simply from a space standpoint, the removal of fuel results in a loss of heat-transfer surface, and the ultimate capacity of the reactor is correspondingly reduced. Some reactors have a considerably greater capacity than their turbines; however, it is assumed that future reactors will have no such excess capacity.

Table A.37. Cost of Targets for the Malibu Reactor

Product	Target	Compound	Target Price ($/kg)	Weight of Target Compound (kg)	Weight of Target Element (kg)	Cost ($)	Canning Cost[b] ($/rod)	Number of Target Rods	Cost of Canning ($)	Cost of Canning & Target ($)
^{14}C	^{14}N	Be_3N_2	400[a]	238.5	117.1	95,500	1,000	842	842,000	937,500
^{60}Co	^{59}Co	Co metal	15	284.8	284.8	4,270	1,000	842	842,000	846,270
^{170}Tm	^{169}Tm	Tm_2O_3	2,750[a]	357.5	313.0	983,000	300	842	252,000	1,235,000
^{210}Po	^{209}Bi	Bi metal	1	1460.0	1460.0	1,460	250	842	210,000	211,000
^{238}Pu	^{237}Np	NpO_2	500,000	206.7	182.1	91,050,000	300	842	252,000	91,302,000
^{242}Cm	^{241}Am	AmO_2	1,500,000	60.7	53.6	80,400,000	5,000[c]	842	4,220,000	84,620,000
^{244}Cm	^{239}Pu	PuO_2	45,000	84.6	74.6	3,357,000	300	842	252,000	3,609,000

[a]Prices are for element, except where noted. Prices currently quoted are used here; however, it seems probable that lower prices could be obtained for the quantities mentioned in some cases.

[b]Most of these costs are based upon canning techniques which have been used. For a large program it would be well worthwhile to develop cheaper methods.

[c]Only limited experience is available for ^{241}Am, so that it is difficult to estimate costs accurately for a large-scale program.

Table A.38. Unit Costs of Isotopes for the Malibu Reactor

	^{14}C	^{60}Co	^{170}Tm	^{210}Po	^{238}Pu	^{242}Cm	^{244}Cm
Reactor charge, $	30,000	330,000	330,000	30,000	330,000	330,000	330,000
Target costs, $	937,500	846,270	1,235,000	211,000	91,302,000	84,620,000	3,609,000
Total yearly costs, $	967,500	1,176,270	1,565,000	241,000	91,632,000	84,950,000	3,939,000
		1,506,270[a]					4,920,000[b]
Production	405 curies	8.68×10^6 curies[a]	5675 g	5.4 g	15,660 g	6,433 g	30.3 g[b]
Unit cost	$2,380/curie	$0.17/curie	$276/g	$44,600/g	$5,850/g	$13,200/g	$162,000/g

[a]Two years' irradiation is assumed because of the favorable effects on specific activity and costs.

[b]Four years' irradiation is assumed. Eight years would produce much more, but it is not yet clear that it would be possible to obtain such an irradiation time.

Table A.39. General Nonrecurring Costs Involved in Establishing an Irradiation Program

Development of targets to achieve lower fabrication costs and to demonstrate safety and reliability	$500,000
Design and fabrication of shipping casks	75,000
Contingency	200,000
Nonrecurring costs associated with beginning a full-scale program in a large power reactor	85,000
Total	$860,000

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