NASA TM X-1512

ACTIVATION OF SODIUM, LITHIUM, AND POTASSIUM IN COMPACT FAST REACTORS AND ITS EFFECT ON SHIELDING

By Leonard Soffer

Lewis Research Center
Cleveland, Ohio

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION

For sale by the Clearinghouse for Federal Scientific and Technical Information
Springfield, Virginia 22151 – CFSTI price $3.00

ACTIVATION OF SODIUM, LITHIUM, AND POTASSIUM IN COMPACT FAST REACTORS AND ITS EFFECT ON SHIELDING

by Leonard Soffer

Lewis Research Center

SUMMARY

Neutron activation of sodium (Na), lithium (Li), and potassium (K) used as possible coolants in a "typical" compact fast reactor were analytically investigated. The natural elements, all individual isotopes, and one common eutectic (NaK 78) were examined.

Suitable energy dependent neutron fluxes were used in conjunction with activation cross sections to calculate saturated specific activities. Activation cross-section data for the elements studied were compiled and tabulated, and these data are judged to be adequate for engineering calculations.

It was also assumed that 1 percent of all the bromine (Br), krypton (Kr), xenon (Xe), and iodine (I) fission products were released into the coolant. The unshielded dose rates from both the fission products and the activated coolant were then calculated.

Results show that when zero holdup time is assumed, the best coolants, on the basis of unshielded activation dose rate, are K^{40}, K^{39}, Li^7, and natural Li. The worst are K^{41} and Na. Because of low natural abundance, K^{40} is impractical as a reactor coolant. Lithium dose rates are very sensitive to holdup time because of the short (0.8 sec) half-lives of the Li products. Holdup of even a few seconds greatly reduces the Li dose rates but not those for any other element. Lithium 7 held up for 8 seconds and gave the lowest activation dose rate of any coolant considered. The unshielded dose rate from the assumed fission product release was the highest of all. It was about two orders of magnitude greater than the zero-holdup Li^7 rate, and about five orders of magnitude greater than the dose rate for Li^7 held up for 8 seconds.

Gas evolution resulting from coolant irradiation was also investigated and found to be negligible except for Li^6.

INTRODUCTION

Alkali metals such as sodium (Na), lithium (Li), and potassium (K) in liquid form

are often considered as coolants for compact nuclear reactors intended for space power. Neutron activation occurs as the coolant circulates through the reactor. Subsequent transport of the radioactive coolant outside the reactor and adjacent biological shield may result in additional radiation exposure to personnel having access to unshielded loop components. The degree of activation clearly is a factor in selection of a reactor coolant for manned applications.

Activation of these materials in thermal spectrum reactors results primarily from radiative capture. The cross sections have been well known for a number of years. The different spectrum in a fast reactor greatly enhances the occurrence of other reactions as well, such as (n, p), (n, α), and (n, 2n). Until relatively recently, neutron cross sections for these reactions had been scarce. Since fast reactors often employ moderating reflectors, some thermal radiative capture is to be expected as well, especially if the reflector is actively cooled. Usually the degree of activation varies markedly in different isotopes of the same chemical element. Since isotope separation may sometimes be feasible, it is useful to examine the individual isotopes, as well as the naturally occurring elements.

The activations were calculated of Na, Li, and K used as coolants in compact fast reactors. A definitive calculation of actual activities or dose rates was not attempted. This calculation would require a detailed knowledge of the actual reactor, loop cycle times, and powerplant components layout. The emphasis in this study was to obtain a relative ranking of these coolants in terms of activation and unshielded dose rates. Rough estimates of these quantities could be useful for preliminary studies or screening purposes.

Actual reactor coolants may also contain small amounts of impurities with large activation cross sections. Quantitative knowledge of the impurity activation is essential in an actual application. This study assumed no impurities in the coolant, however, since the emphasis was to point out differences among the three alkali metals as the primary coolant constituents.

Since accurate and complete activation cross-section data are essential for an analysis of this kind, a literature survey and cross-section compilation were undertaken first. The results of such an effort indicated that, while gaps in cross-section data were present, fairly accurate and complete cross-section data were available for almost all reactions for each of the individual isotopes of Na, Li, and K to allow for a useful engineering assessment. The cross sections that were compiled as a result of this undertaking are listed in appendix A.

A typical fast reactor was then analyzed by using the one-dimensional neutron-transport code DTF (ref. 1) to obtain the magnitude of the neutron fluxes as a function of energy and spatial location throughout the reactor. A set of energy-dependent fluxes was obtained by performing a suitable spatial integration (see the section, Fast Reactor Model

and Spatial Integration of Fluxes). These fluxes can be used to calculate specific activities and specific dose rates (see the section, Saturated Specific Activities and Specific Dose Rates).

Problems other than that of activation may be present. Release of fission products into the coolant by some method of fuel-element failure could add substantial amounts of radioactivity. A hypothetical release was assumed and dose rates were calculated (see the section, Dose Rates from Release of Fission Products into Coolant, and appendix B). In addition to reactions leading to activated products, use of the coolants considered herein also results in evolution of gases in the coolant. This gas evolution was also calculated for each of the coolants considered.

METHOD OF CALCULATION

Activation in Reactor Coolant Loop

Circulating coolants for reactors are activated as they pass through the core, and they decay there and in the external loop. If N is the number of radioactive atoms of some species formed in a unit volume of coolant as a result of the first pass through the core, then, as the unit volume exits from the reactor, N is given as

$$N = \frac{\Sigma\varphi}{\lambda}\left(1 - e^{-\lambda t_1}\right) \tag{1}$$

where

Σ cross section for reaction

φ average neutron flux

λ decay constant of species

t_1 time spent in core, sec

Equation (1) has temporarily neglected the energy dependence of Σ and the fact that φ is a function of both energy and spatial location in the core. During a second pass through the core, the number of new radioactive atoms formed is the same as that given by equation (1), while the number of atoms created in the first cycle decays by the factor $e^{-\lambda(t_1+t_2)}$, where t_2 is the time required for the unit volume to travel the external loop. Thus, after two passes through the core, the total number of radioactive atoms present is

$$N = \frac{\Sigma\varphi}{\lambda}\left(1 - e^{-\lambda t_1}\right)\left[1 + e^{-\lambda(t_1+t_2)}\right] \tag{2}$$

Since the bracketed quantity is a geometric series, the total number of atoms after n passes is

$$N = \frac{\Sigma\varphi}{\lambda}\left(1 - e^{-\lambda t_1}\right)\frac{1 - e^{-n\lambda(t_1+t_2)}}{1 - e^{-\lambda(t_1+t_2)}} \tag{3}$$

As n approaches infinity, N approaches its saturated value and is given by

$$N_{sat} = \frac{\Sigma\varphi}{\lambda}\,\frac{1 - e^{-\lambda t_1}}{1 - e^{-\lambda(t_1+t_2)}} \tag{4}$$

If the cycle time $t_1 + t_2$ is small compared with the half-life, the products $\lambda(t_1 + t_2)$ and λt_1 are both small compared with 1. The exponentials can then be expanded and simplified, and equation (4) can be reduced to

$$N_{sat} = \frac{\Sigma\varphi}{\lambda}\,\frac{t_1}{t_1 + t_2} \tag{5}$$

Thus, the saturated specific activity A_{sat} becomes

$$\lambda N_{sat} = \Sigma\varphi\,\frac{t_1}{t_1 + t_2}$$

When it is taken into consideration that Σ is energy dependent and φ is spatial and energy dependent, this equation is written more correctly as

$$A_{sat} = \left(\frac{t_1}{t_1 + t_2}\right)\int \Sigma(E)\overline{\varphi}(E)dE \tag{6}$$

where the simple product $\Sigma\varphi$ is replaced by an integration over all energies, and where

$\overline{\varphi}(E)$ are fluxes that have been suitably averaged spatially and are thus dependent only on energy.

The ratio $t_1/(t_1 + t_2)$ might range from 0.10 to 0.25 depending somewhat on the conversion method employed. For the purposes of this study, a value of 0.2 was assumed for the fraction of time spent by the coolant in the reactor.

The cycle time was assumed to be small compared with the half-life in the derivation of equations (5) and (6). This is true for all the activation products studied herein except lithium 8 (Li^8) and helium 6 (He^6), both of which are produced when lithium is activated. Both products have half-lives of 0.8 second.

It was not known what loop cycle times might be in an actual fast reactor. Therefore, the saturated activities for Li^8 and He^6 were treated in the same manner as the others, that is, by use of equations (5) and (6) instead of the rigorously correct equation (4). It should be kept in mind, nevertheless, that saturated activities for Li^8 and He^6 are highly sensitive to loop cycle times since their half-lives are so short. This point is explained in the section, Saturated Specific Activities and Specific Dose Rates, where holdup time is discussed in connection with lithium.

Fast-Reactor Model and Spatial Integration of Fluxes

To obtain typical fast-reactor neutron fluxes, a model fast reactor was chosen and neutron fluxes as a function of energy were computed over the entire reactor by the one-dimensional neutron transport code, DTF. The model was a cylindrical reactor fueled with mixtures of uranium dioxide (UO_2) and tungsten (W) in two radial zones. The inner-zone fuel consisted of 50-percent W and 50-percent UO_2 by volume, while the outer-zone fuel was 90-percent W and 10-percent UO_2 by volume. The reactor, which employed a radial 3-inch- (7.62-cm-) thick Be reflector, was 20 percent void by volume. The transport code obtained the required dimensions to achieve a multiplication constant K_{eff} of 1.05. A core diameter of 76 centimeters and a core height of 30 centimeters resulted. The relative radial power distribution obtained is shown in figure 1. This reactor model is certainly not one that would be chosen by design. It has a length-diameter ratio of 0.4 and a poor power distribution in terms of flatness. However, for the purpose of providing a set of fast-reactor neutron fluxes for activation calculations, it is probably acceptable, since the uneven variation in power distribution was properly averaged.

For convenience, the reactor fluxes were scaled to a power level of 1 megawatt. Activation rates per unit weight of coolant at other reactor powers will then scale accordingly. The power distribution rises just before the core-reflector interface. This power increase is caused by the peaking of the thermal flux near the core-reflector interface. This fact is illustrated in the plot of flux against radius for two energy groups (fig. 2).

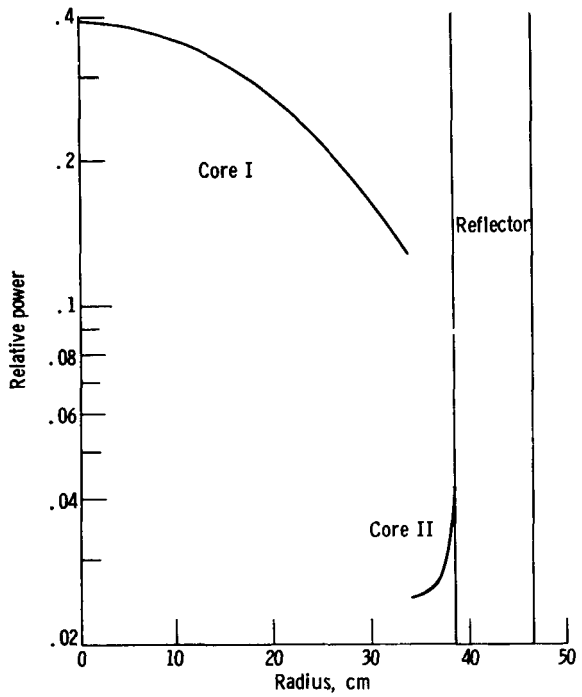

Figure 1. - Relative radial power generation.

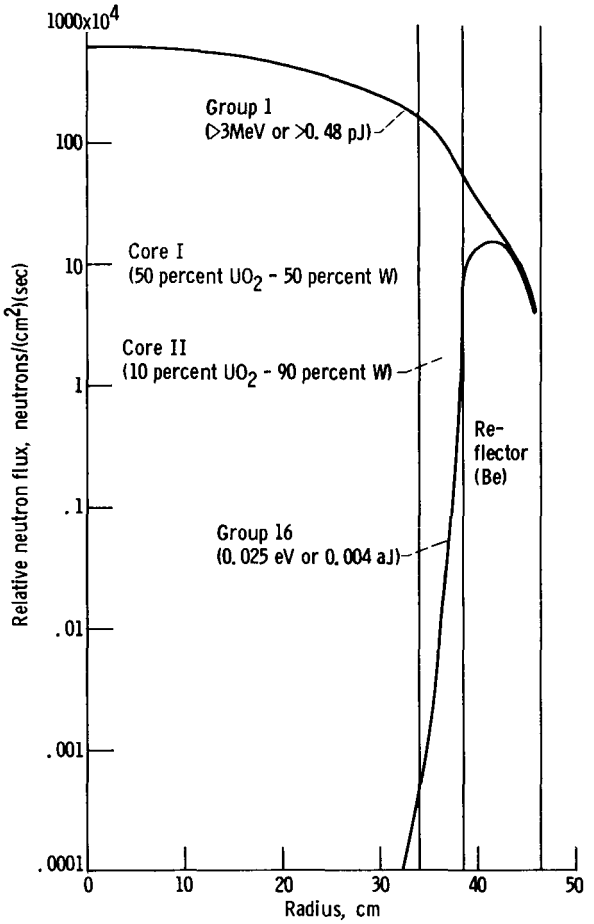

Figure 2. - Relative fast and thermal neutron fluxes as function of radius.

The first energy group (group 1) shows neutrons with an energy greater than 3 MeV (0.48 pJ), and the second group (group 16) shows neutrons at thermal energy (0.025 eV or 0.004 aJ). As shown, the fast flux decreases monotonically from a high value at the center of the core to the reflector edge. The thermal flux, however, has a negligible value except close to the reflector where it rises abruptly to large values, reaches a peak in the reflector, and then declines sharply near the outer boundary. Although the median fission energy for the reactor is between 0.100 and 0.400 MeV (0.016 and 0.064 pJ), the spectrum is fast throughout most of the core but is much more thermal in a relatively thin zone near the core-reflector interface. Thus, even in fast reactors, thermal neutron activation cannot be ignored unless heavy-metal, nonmoderating reflectors are employed. Such a case was not considered in this study.

Once the neutron fluxes as a function of energy for every spatial point in the core and reflector are known, these fluxes can be combined with the proper cross sections to determine the production rates of the radioactive species per unit volume of coolant. Such a double integration over space and energy for each reaction of every isotope inves-

tigated would be tedious and fortunately is unnecessary. A spatial integration can be performed just once to yield a set of neutron fluxes as a function of energy that have been properly averaged over the entire reactor. In the spatial averaging, the flux at a given location is weighted by the volume element at that location. In addition, since activation of a coolant is of interest here, the weighting should also include the relative amount of coolant flow through the volume at that location. Since coolant flow is required to remove heat generated from fissions, the coolant flow rate at a given point is proportional to the spatial power distribution. The equation for calculating spatially averaged fluxes suitable for reactor coolant activation is given as

$$\overline{\varphi}(E) = \frac{\int P(r)\varphi(E, r)dv}{\int P(r)dv} \tag{7}$$

where

$\varphi(E, r)$	flux of neutrons at energy E at space point r
$P(r)$	relative power distribution at space point r
$\overline{\varphi}(E)$	spatially averaged flux of neutrons of energy E
v	volume element of integration

In this investigation, the reflector was also assumed to require cooling because of gamma heating. Hence, the integration was carried out over the reflector as well, except that in the reflector a value of 5 percent of the total power in the core was substituted for the function P(r) in equation (7) to account for gamma-heat deposition in the reflector. Spatial averaging was performed only in the radial direction since the axial variation was not expected to be large.

The spatially averaged fluxes, as a function of energy, scaled for a reactor power of 1 megawatt, are listed in table I. It is these spatially averaged fluxes (together with the cross sections for a given reaction) that give a true indication of how important fast neutron activations are in relation to thermal neutron activations. The highest flux occurs in the group from 0.100 to 0.400 MeV (0.016 to 0.064 pJ). The median fission energy of the entire reactor also occurs in this group. Below 0.100 MeV (0.016 pJ), the fluxes decrease in magnitude as the energy is lowered, until at low energies the spatially weighted fluxes are about four orders of magnitude below those of the highest magnitude. Since many reactions leading to activation have large cross sections at low energies and the cross section often decreases monotonically as the energy increases, the total activation is calculated from the integral

$$\text{Total production rate} = \int \Sigma(E)\overline{\varphi}(E)dE$$

TABLE I. - SPATIALLY AVERAGED FLUXES

[Reactor power, 1 MW.]

Group	Energy limits of groups		Average fluxes, neutrons/(cm^2)(sec)
	U.S. customary units	SI units	
1	>3 MeV	>0.48 pJ	9.13×10^{11}
2	1.4 to 3 MeV	0.22 to 0.48 pJ	1.91×10^{12}
3	0.9 to 1.4 MeV	0.14 to 0.22 pJ	1.61
4	0.4 to 0.9 MeV	0.064 to 0.14 pJ	4.70
5	0.1 to 0.4 MeV	0.016 to 0.064 pJ	5.70
6	17 to 100 keV	2.72 to 16.0 fJ	2.20
7	3 to 17 keV	0.48 to 2.72 fJ	2.02×10^{11}
8	0.55 to 3 keV	0.088 to 0.48 fJ	2.12×10^{10}
9	100 to 550 eV	16 to 88 aJ	5.73×10^{9}
10	30 to 100 eV	4.8 to 16 aJ	2.80
11	10 to 30 eV	1.6 to 4.8 aJ	1.49
12	3 to 10 eV	0.48 to 1.6 aJ	1.26
13	1 to 3 eV	0.16 to 0.48 aJ	9.45×10^{8}
14	0.4 to 1 eV	0.064 to 0.16 aJ	5.30
15	0.1 to 0.4 eV	0.016 to 0.064 aJ	4.75
16	0.025 eV	0.004 aJ	8.50

and it is not immediately obvious from table I that thermal reactions can be neglected in primarily fast reactors.

Saturated Specific Activities and Specific Dose Rates

When the spatially averaged fluxes shown in table I and the cross-section data for the isotopes of sodium, lithium, and potassium given in appendix A are used, the saturated activities per gram of pure isotope per megawatt of reactor operating power can be calculated by use of equation (6). The results are shown in table II. From this table, the total saturated activities for various coolants operating in a reactor at some power level P (MW) can be obtained.

How important these activities are to unshielded personnel nearby in terms of absorbed dose rate is not indicated in table II. Hence, the absorbed dose rate to an unshielded observer located 1 meter away from 1 gram of coolant irradiated in a reactor operating at 1 megawatt was calculated. The results are summarized in table III.

From this table it can be seen that the lowest dose rate from activation is obtained with pure separated K^{40}. However, the natural abundance of K^{40} is about 0.01 percent, and it has both a lighter and a heavier neighbor isotope adjacent to it. Hence, at least

TABLE II. - SATURATED SPECIFIC ACTIVITIES

Target nuclide	Product nuclide	Half-life	Saturated specific activity, dis/sec (a)
Na^{23}	Na^{24}	15.0 hr	8.36×10^{7}
	Ne^{23}	38 sec	4.15×10^{7}
	Na^{22}	2.58 yr	6.22×10^{4}
	F^{20}	11 sec	2.81×10^{7}
Li^{6}	He^{6}	0.8 sec	1.39×10^{8}
Li^{7}	He^{6}	0.8 sec	4.0×10^{5}
	Li^{8}	0.8 sec	4.59×10^{6}
K^{39}	Ar^{39}	260 yr	1.52×10^{9}
	K^{38}	7.7 min	3.7×10^{3}
	Cl^{36}	3×10^{5} yr	2.3×10^{8}
K^{40}	Ar^{39}	260 yr	1.59×10^{7}
K^{41}	K^{42}	12.4 hr	2.15×10^{8}
	Ar^{41}	1.83 hr	5.2×10^{8}
	Cl^{38}	37.3 min	5.6×10^{7}

[a]Per g of target material; per MW of reactor power.

TABLE III. - SATURATED SPECIFIC DOSE RATES (UNSHIELDED)

Coolant	Absorbed dose rate, mrem/hr (a)
Na^{23}, pure (natural sodium)	4.77
Li^{6}, pure	[b]2.05
Li^{7}, pure	[b].26
K^{39}, pure	.155
K^{40}, pure	[c].004
K^{41}, pure	14.6
Li, natural	[b].394
Li, natural	[d].000394
K, natural	1.15
NaK (78 K) eutectic	1.95

[a]At distance of 1 m; per g of coolant; per MW of reactor power.

[b]No holdup time.

[c]Half of dose rate from unirradiated but naturally radioactive K^{40}.

[d]Holdup time, 8 sec.

two isotope separation processes would be required, and the cost of separation would probably be prohibitive.

Activation of Li^{6} and Li^{7} produces mainly He^{6} in the former and both He^{6} and Li^{8} in the latter. Both He^{6} and Li^{8} have half-lives of only 0.8 second and decay by emission of beta particles having maximum energies of 3.5 MeV (0.56 pJ) and 13 MeV (2.08 pJ), respectively. X-rays (known as Bremsstrahlung) are produced when these betas are stopped in the walls of the containment vessel or piping. The X-rays have a continuous energy spectrum extending up to the maximum beta energy; however, their average energy is approximately one-third of the maximum beta energy. In addition, when 3.5-MeV (0.56-pJ) and 13-MeV (2.08-pJ) betas are stopped in a relatively good material for production of X-rays, such as tantalum (Z = 73), on the average, 13 and 39 percent, respectively, of their kinetic energy is converted to X-ray energy. The specific dose rates shown in table III for these nuclides are for zero holdup time between saturated activity and dosage. Except for locations in or immediately adjacent to the core, zero holdup time is unrealistic. Typical loop-cycle times are of the order of seconds, and while these times have a negligible effect on the dose rate from Na or K isotopes, they profoundly affect the Li dose rates. A holdup of only 4 seconds, for example, whether from

normal loop cycling or deliberate holdup would reduce the lithium dose rates shown in table III by a factor of 32. An 8-second holdup would result in lowering the values shown in the table by a factor of 1000. Holdup time, therefore, complicates any evaluation of Li activation relative to that of Na or K.

The most undesirable isotopes are clearly K^{41} and Na. For K^{41}, about half the dose rate is from the 1.83-hour argon 41 (Ar^{41}), while the remainder is mainly from the 12.4-hour K^{42}. In Na, the 15.0-hour Na^{24} activity dominates the dose rate. Of the natural materials, natural Li has the lowest dose rate, with a potential for even lower dose rates if loop-cycle holdup or deliberate holdup is used.

Dose Rates from Release of Fission Products into Coolant

If fission products are released to the reactor coolant, these products may also result in high dose rates. Under such circumstances, the dose rates were calculated by assuming a hypothetical fission product release.

It was assumed that a saturated reactor operating at 1 megawatt of power releases 1 percent of the Br, Kr, Xe, and I isotopes into the coolant. Another way of saying this is that 1 percent of the fuel elements develop pinholes or minute cracks and all the volatile fission products (Br, Kr, Xe, and I) escape into the coolant. If the entire fission product inventory in the coolant can be regarded as a point source, the dose rate to an unshielded observer 1 meter away is approximately 2200 rem per hour (see appendix B for details).

TABLE IV. - DOSE RATES FROM FISSION PRODUCTS AND COOLANT ACTIVATION

Radiation source	Absorbed dose rate, rem/hr (a)
Na, pure	445
Li^6, pure	[b]188
Li^7, pure	[b]23
K^{39}, pure	13.9
K^{40}, pure	.315
K^{41}, pure	1310
Li, natural	[b]36
Li, natural	[c].036
K, natural	104
NaK (78 K) eutectic	176
Volatile fission products, 1 percent	2200

[a]At distance of 1 m; per MW of reactor power; 200 lb (90 kg) coolant assumed.
[b]No holdup time.
[c]Holdup time, 8 sec.

Shown in table IV are the dose rates to an unshielded observer 1 meter away, caused by the activation of various coolants with 200 pounds (90 kg) of coolant assumed to be in the primary loop of a 1-megawatt reactor.

While it is unrealistic to suppose that an observer could be subjected to the dose rate from the entire coolant loop, the relative values of these numbers should be valid for an observer in the vicinity of a major loop component. Thus, if the fission product

release model used herein is realistic, the selection of a coolant to minimize activation is of secondary importance. If the fission product release postulated herein is too severe, selection of a coolant becomes important. When appreciable holdup time between the reactor and the loop (of the order of a few seconds) can be arranged without undue shield weight penalty, pure Li^7 offers the potential of yielding the lowest dose rates. When little or no holdup is practical, pure K^{39} may be superior.

Even when no fission products are present, personnel are likely to require some shielding from activated loop components no matter what coolant is used, with the exception of Li that has been sufficiently held up for a number of seconds.

Gas Production from Irradiation of Sodium, Lithium, and Potassium Coolants

In addition to radioactive nuclei, gases (stable and radioactive) are produced from coolant activation. In table V, the number of moles of gas evolved in a 1-year operation of a reactor at 1 megawatt using 200 pounds (90 kg) of coolant are listed. It can be seen from this table that gas evolution is probably unimportant for all of the coolants except Li^6 when appreciable quantities of helium and tritium accumulate.

TABLE V. - GAS PRODUCTION RATES

Coolant material	Gas evolved	Gas production rate, g-moles/yr (a)
Na^{23}, pure	Ne^{20}	2.5×10^{-2}
	Ne^{22}	5.7×10^{-5}
Li^6, pure	He^4	6.6
	H^3	6.4
	H^2	2.15×10^{-1}
	H^1	3.8×10^{-3}
Li^7, pure	He^4	4.6×10^{-2}
	H^3	4.6×10^{-2}
	H^2	4.7×10^{-5}
K^{39}, pure	H^1	3.55×10^{-2}
	He^4	5.4×10^{-3}
	H^2	4.7×10^{-4}
	Ar^{38}	7.1×10^{-4}
K^{40}, pure	H^2	3.4×10^{-4}
	H^1	3.3×10^{-5}
K^{41}, pure	He^4	1.3×10^{-3}

[a] Per MW of reactor power.

CONCLUSIONS

An investigation of the neutron activation of sodium (Na), lithium (Li), and potassium (K) used as possible coolants in a typical fast reactor has resulted in the following conclusions:

1. Existing neutron activation cross-section data of the individual isotopes of Na, Li, and K are adequate for a useful engineering assessment of the problem.

2. On the basis of dose rate to an unshielded observer and when zero holdup time is assumed, the best coolants of those considered are K^{40}, K^{39}, Li^{7}, and natural Li. The worst coolants are K^{41} and Na.

3. Because of low natural abundance and the need for two isotopic separation processes, K^{40} is impractical as a useful reactor coolant.

4. Lithium is relatively difficult to rate on activation because of the effect of holdup between the reactor exit and the external loop. Because of the short half-lives (0.8 sec) of the Li products, holdup times of even a few seconds strongly affect Li dose rates. Lithium 7 held up for 8 seconds gives the lowest dose rates to an unshielded observer of any of the coolants considered. The dose rates of the other coolants are not affected by holdup.

5. Some shielding will be needed to protect personnel in the vicinity of the various loop components for all of the coolants considered except Li that has been held up for periods of the order of several seconds or more.

6. If the fission product release rate into the coolant assumed herein is realistic, coolant selection to minimize activation is of secondary importance. A fission product release rate at least two orders of magnitude smaller than that assumed would be required before coolant activation exerted a major influence on personnel dose rate.

7. Gas evolution as the result of irradiation of isotopes of Na, Li, and K is small except for Li^{6} (and possibly natural Li).

8. Use of a fast-reactor model different from the one chosen should not alter the major conclusions reached herein, although some numerical changes would be expected.

Lewis Research Center,
National Aeronautics and Space Administration,
Cleveland, Ohio, August 30, 1967,
120-27-06-07-22.

APPENDIX A

NEUTRON REACTIONS, PRODUCTS, AND CROSS SECTIONS FOR ISOTOPES OF SODIUM, LITHIUM, AND POTASSIUM

The basic nuclear data used in this report to calculate activations and dose rates are tabulated herein. The important reactions and products formed are listed in tables VI, VII, and VIII for the isotopes of sodium, lithium, and potassium, respectively. The cross sections are similarly given in tables IX, X, and XI. The references from which the data were obtained are also listed.

TABLE VI. - NEUTRON REACTIONS OF SODIUM AND PRODUCTS FORMED

Target nuclide	Natural abundance, percent	Reaction	Product formed	Half-life	Type of radiation emitted	Energy of radiation	
						MeV	pJ
Na^{23}	100	n, γ	Na^{24}	15.0 hr	γ	1.37	0.219
		n, γ	Na^{24}	15.0 hr	γ	2.75	.440
		n, p	Ne^{23}	38 sec	γ	.44	.0704
		n, p	Ne^{23}	38 sec	γ	1.65	.264
		n, 2n	Na^{22}	2.58 yr	γ	1.28	.205
		n, α	F^{20}	11 sec	γ	1.63	.261
		n, α	F^{20}	11 sec	β	5.4	.864

TABLE VII. - NEUTRON REACTIONS OF LITHIUM AND PRODUCTS FORMED

Target nuclide	Natural abundance, percent	Reaction	Product formed	Half-life	Type of radiation emitted	Energy of radiation	
						MeV	pJ
Li^{6}	7.42	n, γ	Li^{7}	Stable	---	------	-------
		n, n'd	He^{4}	Stable	---	------	-------
		n, p	He^{6}	0.8 sec	β	3.5	0.560
		n, α	H^{3}	12.2 yr	β	.018	.00288
		n, 2n	He^{4}	Stable	---	------	-------
Li^{7}	92.58	n, γ	Li^{8}	0.8 sec	β	13	2.080
		n, n't	H^{3}	12.2 yr	β	.018	.00288
		n, d	He^{6}	0.8 sec	β	3.5	.560
		n, 2n	Li^{6}	Stable	---	------	-------
		n, n'p	He^{6}	0.8 sec	β	3.5	.560
		n, 3n	He^{4}	Stable	---	------	-------

TABLE VIII. - NEUTRON REACTIONS OF POTASSIUM AND PRODUCTS FORMED

Target nuclide	Natural abundance, percent	Reaction	Product formed	Half-life	Type of radiation emitted	Energy of radiation	
						MeV	pJ
K^{39}	93.1	n, γ	K^{40}	1.3×10^{9} yr	β	1.32	0.211
		n, γ	K^{40}	1.3×10^{9} yr	γ	1.46	.2337
		n, p	Ar^{39}	260 yr	β	.57	.0912
		n, 2n	K^{38}	7.7 min	γ	2.2	.352
		n, α	Cl^{36}	3×10^{5} yr	β	.71	.1207
		n, d	Ar^{38}	Stable	---	----	------
		n, np	Ar^{38}	Stable	---	----	------
K^{40}	0.0118	n, γ	K^{41}	Stable	---	----	------
		n, p	Ar^{40}	Stable	---	----	------
		n, 2n	K^{39}	Stable	---	----	------
		n, α	Cl^{37}	Stable	---	----	------
		n, d	Ar^{39}	260 yr	β	0.57	0.0912
		n, np	Ar^{39}	260 yr	β	.57	.0912
K^{41}	6.88	n, γ	K^{42}	12.4 hr	γ	1.52	0.2431
		n, p	Ar^{41}	1.83 hr	γ	1.3	.208
		n, 2n	K^{40}	1.3×10^{9} yr	β	1.32	.211
		n, 2n	K^{40}	1.3×10^{9} yr	γ	1.46	.2337
		n, α	Cl^{38}	37.3 min	γ	1.6	.256
		n, α	Cl^{38}	37.3 min	γ	2.2	.352
		n, d	Ar^{40}	Stable	---	----	------
		n, 2p	Cl^{40}	1.4 min	γ	1.5	.240
		n, 2p	Cl^{40}	1.4 min	γ	2.75	.440
		n, He^{3}	Cl^{38}	37.3 min	γ	1.6	.256
		n, He^{3}	Cl^{38}	37.3 min	γ	2.2	.352

TABLE IX. - NEUTRON CROSS SECTIONS FOR ACTIVATION OR GAS-FORMING REACTIONS OF SODIUM

Reaction	Energy range		Average cross section, b (or 10^{-24} cm^2)	Reference
	MeV	pJ		
$Na^{23}(n,\gamma)Na^{24}$	(a)	(a)	0.53	2
	0.02 to 1	0.0032 to 0.16	.001	4
	2.7	0.432	<.00023	4
	4.0	.640	<.00012	4
	>1.0	>.16	[b].0005	
$Na^{23}(n,p)Ne^{23}$	<4.4	<0.704	0	4
	4.6 to 6.2	0.736 to 0.992	.0196	4
	6.2 to 8.0	.992 to 1.280	.030	4
	8.0 to 15	1.280 to 2.40	.050	4
$Na^{23}(n,2n)Na^{22}$	<12.5	<2.00	0	4
	13.5 to 18	2.16 to 2.88	.030	
$Na^{23}(n,\alpha)F^{20}$	<6	<0.96	0	4
	6 to 15	0.96 to 2.40	.050	4

[a]Thermal at 0.025 eV (0.004 aJ). Cross section varies inversely with neutron velocity to 300 eV (48.0 aJ).

[b]Average.

TABLE X. - NEUTRON CROSS SECTIONS FOR ACTIVATION OR GAS-FORMING REACTIONS OF LITHIUM

Reaction	Energy range		Average cross section, b (or 10^{-24} cm^2)	References
	MeV	pJ		
$Li^6(n, n'd)He^4$	<3	<0.48	0	4 and 5
	3 to 14	0.48 to 2.24	.50	4 and 5
$Li^6(n, p)He^6$	<4	<0.64	0	4 and 5
	4 to 14	0.64 to 2.24	.015	4 and 5
$Li^6(n, \alpha)H^3$	(a)	(a)	945	2
	0.01 to 20	0.0016 to 3.20	(see ref. 2)	2
$Li^6(n, 2n)He^4$ plus H^1	<8	<1.28	0	5
	8 to 15	1.28 to 2.40	.050	5
$Li^7(n, \gamma)Li^8$	(b)	(b)	0.036	6
	0.15 to 15	0.024 to 2.40	.00001	6
$Li^7(n, n't)He^4$	<4	<0.64	0	4 and 6
	4 to 6	0.64 to 0.96	.20	4 and 6
	6 to 15	.96 to 2.40	.40	4 and 6
$Li^7(n, d)He^6$	<10	<1.6	0	6
	10 to 15	1.6 to 2.40	.005	6
$Li^7(n, 2n)He^4$ plus H^2	<10	<1.6	0	6
	10 to 15	1.6 to 2.40	.020	6
$Li^7(n, n'p)He^6$	<11.5	<1.84	0	6
	>11.5	>1.84	No data	---
$Li^7(n, 3n)He^4$ plus H^1	<12	<1.92	0	6
	>12	>1.92	No data	---

[a]Thermal at 0.025 eV (0.004 aJ). Cross section varies inversely with neutron velocity to 10 keV (1.60 fJ).

[b]Thermal at 0.025 eV (0.004 aJ). Cross section varies inversely with neutron velocity to 150 keV (24.0 fJ).

TABLE XI. - NEUTRON CROSS SECTIONS FOR ACTIVATION OR GAS-FORMING REACTIONS OF POTASSIUM

Reaction	Energy range		Average cross section, b (or 10^{-24} cm^2)	References
	MeV	pJ		
$K^{39}(n, p)Ar^{39}$	0 to 1.0	0 to 0.16	0	4 and 7
	1 to 3	.16 to 0.48	.060	4 and 7
	3 to 8	.48 to 1.28	.30	4 and 7
	8 to 14	1.28 to 2.24	.35	4 and 7
$K^{39}(n, 2n)K^{38}$	<14	<2.24	0	4
	>14	>2.24	.007	4
$K^{39}(n, \alpha)Cl^{36}$	<2	<0.32	0	4 and 7
	2 to 4	0.32 to 0.64	.02	4 and 7
	4 to 6	.69 to 0.96	.13	4 and 7
	6 to 15	.96 to 2.40	.15	4 and 7
$K^{39}(n, d)Ar^{38}$	<6	<0.96	0	7
	6 to 8	0.96 to 1.28	.07	7
	8 to 15	1.28 to 2.40	.04	7
$K^{39}(n, np)Ar^{38}$	<7	<1.12	0	7
	7 to 10	1.12 to 1.60	.05	7
	10 to 15	1.60 to 2.40	.15	7
$K^{40}(n, p)Ar^{40}$	-------	----------	No data	---
$K^{40}(n, d)Ar^{39}$	-------	----------	No data	---
$K^{40}(n, np)Ar^{39}$	-------	----------	No data	---
$K^{41}(n, \gamma)K^{42}$	(a)	(a)	1.3	3
	0.020	0.0032	.022	4
	.024	.00384	[b]<.020	4
$K^{41}(n, p)Ar^{41}$	<1.75	<0.280	0	4
	14.5	2.320	[c].075	4
			(No other data)	
$K^{41}(n, \alpha)Cl^{38}$	<0.1	<0.016	0	4
	14.5	2.320	[d].030	4
			(No other data)	
$K^{41}(n, 2p)Cl^{40}$	<8	<1.28	0	4
	14.5	2.32	<.00013	4
			(No other data)	
$K^{41}(n, He^3)Cl^{38}$	12.9	<2.063	0	4
	14.5	2.320	<.0025	4
			(No other data)	

[a]Thermal at 0.025 eV (0.004 aJ).

[b]See fig. 3 (data from ref. 7). Cross-section value assumed to be 0.667 times (n, γ) cross section for natural K.

[c]See fig. 4 (data from ref. 7). Cross-section value assumed to be 0.25 times (n, p) cross section for natural K.

[d]See fig. 4 (data from ref. 7). Cross-section value assumed to be 0.28 times (n, α) cross section for natural K.

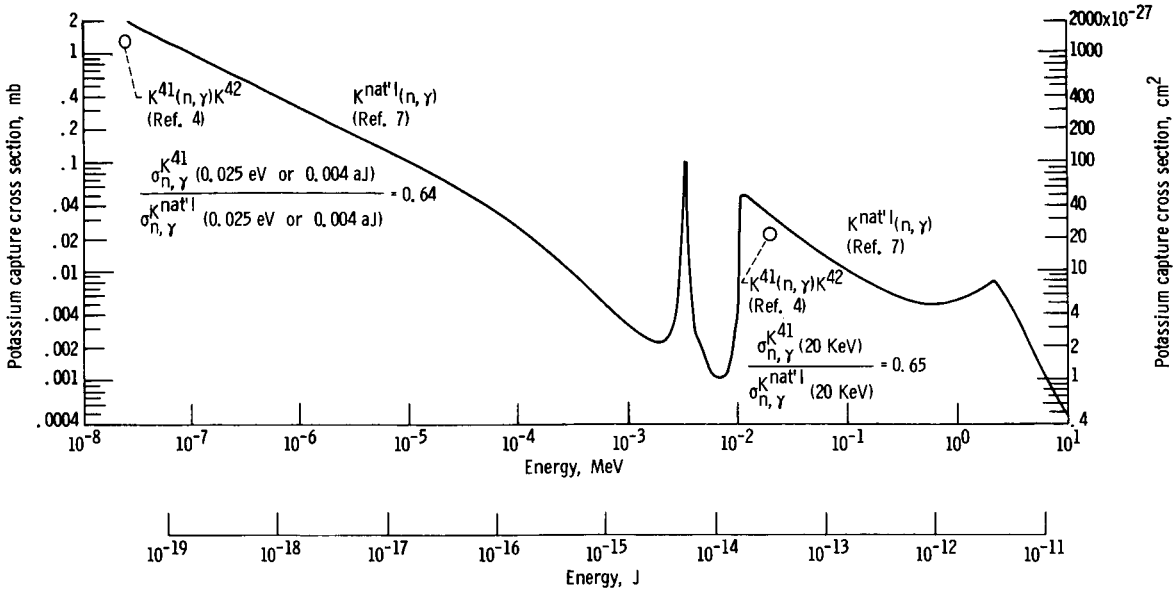

Figure 3. - Potassium capture cross sections as function of energy.

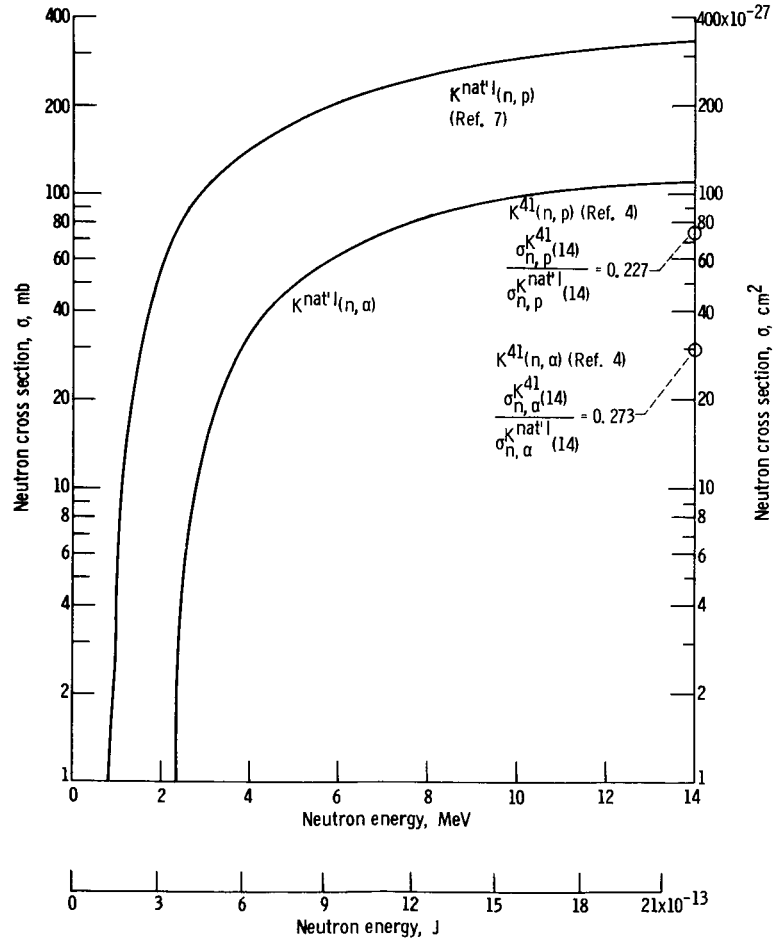

Figure 4. - Potassium (n, α) and (n, p) cross sections.

APPENDIX B

DOSE RATES FROM RELEASE OF FISSION PRODUCTS INTO COOLANT

Data on the volatile fission products (Br, Kr, Xe, and I) were obtained from reference 8. In table XII are listed those nuclides that are gamma emitters, have appreciable yields, and emit photons of 0.1 MeV (0.016 pJ) or greater, and, thus, have an appreciable effect on human dose rates.

TABLE XII. - GAMMA-EMITTING VOLATILE FISSION PRODUCTS

Nuclide	Half-life	Fission yield, percent	Photon energy		Frequency, percent
			MeV	pJ	
Br^{84}	30 min	1.0	1.9 .9	0.30 .14	30 50
Br^{87}	56 sec	2.7	5.4 3.0	0.86 .48	56 14
Kr^{85m}	4.36 hr	1.5	0.3 .15	0.048 .024	20 80
Kr^{87}	78 min	2.5	2.0	0.32	50
Kr^{88}	2.77 hr	3.7	2.0 1.5 1.0	0.32 .24 .16	70 15 25
Kr^{89}	3.18 min	2.7	2.5	0.4	50
I^{131}	8.05 days	2.9	0.5	0.08	100
I^{132}	2.4 hr	4.7	0.7	0.11	100
I^{133}	20.8 hr	6.5	0.5	0.08	100
I^{134}	52.5 min	6.7	1.0	0.16	100
I^{135}	6.68 hr	5.9	1.5	0.24	100
Xe^{133}	5.27 days	6.5	0.1	0.016	100
Xe^{135m}	15.6 min	1.8	0.5	0.08	100
Xe^{135}	9.13 hr	6.2	0.2	0.032	100

The activity from any one of the volatile fission products is given by

$$A = 3.1\times10^{10}\ Py(1 - e^{-\lambda\tau})e^{-\lambda t}$$

where

A activity, dis/sec

P reactor power, W

y fractional yield of fission product of interest

λ decay constant, sec^{-1}

τ reactor operating time

t decay time

If infinite reactor operating time ($\tau = \infty$), no shut down (t = 0), a reactor power of 1 megawatt ($P = 10^6$), and a release of 100 percent of the volatile fission products from 1 percent of the fuel elements are assumed, $A = 3.1\times10^{10}\times10^{6}\times10^{-2}$ y disintegrations per second (or 3.1×10^{14} y dis/sec) for each of the nuclides in question. The source strengths of the fission products can then be calculated and grouped, for convenience, by energy as shown in table XIII.

If it is assumed that all the fission products are collected in a point source, the absorbed dose rate to an unshielded observer can be calculated and is tabulated for various distances in table XIV.

TABLE XIII. - GAMMA SOURCE STRENGTHS FOR VOLATILE FISSION PRODUCTS IN COOLANT

[Grouped by energy. Saturated reactor at 1 MW; 1 percent release of volatiles.]

Average energy		Source strength	
MeV	pJ	MeV/sec	J/sec
0.2	0.032	6.70×10^{12}	1.07
.5	.08	3.73×10^{13}	5.98
1.0	.16	2.53	4.05
1.5	.24	3.01	4.81
2.0	.32	2.55	4.08
2.5	.40	1.40	2.24
5.4	.86	2.55	4.08

TABLE XIV. - UNSHIELDED DOSE RATE FROM VOLATILE FISSION PRODUCTS

[Saturated reactor at 1 MW; 1 percent release of volatiles.]

Separation distance, m	Dose rate, rem/hr
1	2200
2	550
5	88
10	22
20	5.5
50	.88
100	.22

REFERENCES

1. Lathrop, K. D.: DTF-IV - A Fortran-IV Program for Solving the Multigroup Transport Equation with Anisotropic Scattering. Rep. No. LA-3373, Los Alamos Scientific Lab., July 1965.

2. Hughes, Donald J.; and Schwartz, Robert B.: Neutron Cross Sections. Rep. No. BNL-325, second ed., Brookhaven National Lab., July 1958.

3. Hughes, D. J.; Magurno, B. A.; and Brussel, M. K.: Neutron Cross Sections. Rep. No. BNL-325, second ed., suppl. 1, Brookhaven National Lab., Jan. 1960.

4. Stehn, John R.; Goldberg, Murrey D.; Magurno, Benjamin A.; and Wiener-Chasman, Renate: Neutron Cross Sections. Volume I. Z = 1 to 20. Rep. No. BNL-325, second ed., suppl. 2, vol. I, Brookhaven National Lab., May 1964.

5. Pendlebury, E. D.: Neutron Cross Sections of Li-6 in the Energy Range 0.001 ev-15 Mev. Rep. No. AWRE-0-60/64, United Kingdom Atomic Energy Authority, July 1964.

6. Pendlebury, E. D.: Neutron Cross Section of Li-7 in the Energy Range 0.001 ev-15 Mev. Rep. No. AWRE-0-61/64, United Kingdom Atomic Energy Authority, July 1964.

7. Tralli, N.; Kalos, M. H.; Longano, A. J.; Ray, J. H.; Sullivan, R. P.; Troubetzkoy, E. S.; and Trupin, B. H.: Neutron Cross Sections for Titanium, Potassium, Magnesium, Nitrogen, Aluminum, Silicon, Sodium, Oxygen, and Manganese. Rep. No. UNC-5002, United Nuclear Corp., Jan. 1962.

8. Blomeke, J. O.; and Todd, Mary F.: Uranium-235 Fission-Product Production as a Function of Thermal Neutron Flux, Irradiation Time, and Decay Time. I. Atomic Concentrations and Gross Totals. Rep. No. ORNL-2127, pt. 1, vol. 1, Oak Ridge Nat. Lab., Aug. 1957.

"The aeronautical and space activities of the United States shall be conducted so as to contribute . . . to the expansion of human knowledge of phenomena in the atmosphere and space. The Administration shall provide for the widest practicable and appropriate dissemination of information concerning its activities and the results thereof."

—NATIONAL AERONAUTICS AND SPACE ACT OF 1958

NASA SCIENTIFIC AND TECHNICAL PUBLICATIONS

TECHNICAL REPORTS: Scientific and technical information considered important, complete, and a lasting contribution to existing knowledge.

TECHNICAL NOTES: Information less broad in scope but nevertheless of importance as a contribution to existing knowledge.

TECHNICAL MEMORANDUMS: Information receiving limited distribution because of preliminary data, security classification, or other reasons.

CONTRACTOR REPORTS: Scientific and technical information generated under a NASA contract or grant and considered an important contribution to existing knowledge.

TECHNICAL TRANSLATIONS: Information published in a foreign language considered to merit NASA distribution in English.

SPECIAL PUBLICATIONS: Information derived from or of value to NASA activities. Publications include conference proceedings, monographs, data compilations, handbooks, sourcebooks, and special bibliographies.

TECHNOLOGY UTILIZATION PUBLICATIONS: Information on technology used by NASA that may be of particular interest in commercial and other non-aerospace applications. Publications include Tech Briefs, Technology Utilization Reports and Notes, and Technology Surveys.

Details on the availability of these publications may be obtained from:

SCIENTIFIC AND TECHNICAL INFORMATION DIVISION

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION

Washington, D.C. 20546