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ELEMENTS OF HIGH-TEMPERATURE GAS-COOLED REACTOR FUEL REPROCESSING*

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INTRODUCTION

The purpose of this essay is to present the salient features of High-Temperature Gas-Cooled (HTGR) Reactor fuel reprocessing. It must be observed at the outset that the reprocessing technology for HTGR fuel is in an early stage of development, with much difficult work yet to be done before practical, acceptable processes exist. Thus, it may well be ten or more years before the beginning of commercial HTGR fuel reprocessing in the U. S. Of course, if HTGR's do not penetrate the market as significant energy producers, there will be no commercial reprocessing.

Table 1 is one of many projections which have been made about the increase in number of electric power producing plants with time. It shows in general how building one type of reactor affects the competitive position, and hence the rate of construction, of another. If we accept the numbers in the table at face value, it follows that in the year 2000 there may be as many as 410 HTGR's of 1150 MW(e) capacity. However, such projections are to be taken only as indicative of the nature of the interactions of the factors affecting reactor construction. The relative positions of the reactors are sensitively dependent upon the assumptions in the calculations. It is quite likely that important factors are completely omitted in some calculations, either by choice or because they were not thought of.

TABLE 1. OPTIMIZED CALCULATIONS OF PERCENT OF TOTAL ELECTRICAL POWER
FROM PLANTS IN VARIOUS COMBINATIONS^a

PLANTS INCLUDED ^b	FOSSIL	FOSSIL LWR	FOSSIL LWR LMFBR	FOSSIL LWR LMFBR HTGR	FOSSIL LWR LMFBR HTGR GCFR SCFR MSR
TOTAL NUCLEAR CAPACITY, GW(e)					
1980		124	158	143	162
2000		484	1,056	1,062	1,122
% OF TOTAL CAPACITY IN 2000					
FOSSIL	100	64.2	21.9	22.7	17.0
LWR		35.8	20.9	8.4	8.2
LMFBR			57.2	24.5	-
HTGR				44.4	5.9
GCFR					32.8
SCFR					2.4
MSR					25.7

^a TAKEN FROM POTENTIAL NUCLEAR POWER GROWTH PATTERNS, WASH-1098
(DECEMBER 1970).

^b LWR: LIGHT WATER REACTOR; LMFBR: LIQUID METAL FAST BREEDER REACTOR;
HTGR: HIGH-TEMPERATURE GAS-COOLED REACTOR; GCFR: GAS-COOLED FAST
REACTOR (HELIUM IS UNDERSTOOD TO BE THE COOLANT GAS); SCFR: STEAM-
COOLED FAST REACTOR; MSR: MOLTEN SALT REACTOR.

Since about 23 kg of uranium-plus-thorium must be processed each day to satisfy the refueling requirements for each 1150 MW(e) HTGR, it may be inferred from Table 1 that there will exist a reprocessing capacity requirement of about 9.5 metric tons of uranium-plus-thorium per day in the year 2000. This would truly be more than adequate justification for developing HTGR fuel recycle.

GENERAL ASPECTS OF REPROCESSING

The purpose of reactor fuel reprocessing is to recover and purify fuel values from irradiated fuel. In commercial power-producing reactors, the fuel has generally been in the reactor long enough that fission products and neutron capture products have accumulated to the point of significantly affecting the economic optimization of fuel recycle. A wide variety of factors interplay and affect this optimization. Thus, the high cost of fuel fabrication makes it imperative that fuel be left in the reactor as long as practical. On the other hand, the state of the art in fuel fabrication limits fuel life to four to six years, at best. If fuel is left in a reactor until a significant number of fuel elements fail, the overall costs of the failures may easily become a controlling economic factor. Failures may range from such things as element warpage leading to sticking in the core, to outright loss of fuel integrity, leading to fission product release, reactor shutdown, and the necessity for costly decontamination.

An important economic penalty is that associated with the reactor down-time required for re-fueling, or for moving fuel around in the core to produce a more uniform burnup in the core (fuel shuffling). Even when a reactor is not operating, there is a continuing cost associated with

fuel inventory and plant depreciation. Therefore, it is often the case that part of the fuel in a core is irradiated well beyond the point of optimum burnup with regard to buildup of fission products and neutron capture products; by so doing, the number of times the reactor is shut down is decreased. This mode of operation leads to overall optimization of fuel cycle economics. It is these highly exposed fuel elements which are most likely to control fuel reprocessing operations, since they have the highest radiation levels, the most fission products, and the highest incidence of failure in ways important in reprocessing, such as fission product leakage and fuel element distortion.

In general, reprocessing HTGR fuels consists of shipping irradiated fuel elements from the reactor to a reprocessing plant, burning the fuel to remove the bulk of the carbon, preparing the fuel for dissolution and dissolving it in nitric acid solutions, separating and purifying the uranium and thorium (plutonium must be recovered pure in some proposed fuel cycles) by solvent extraction, disposing of the radioactive wastes, and shipping the purified fuel values to the fuel element fabricator. Treatment and disposal of radioactive gases is an important problem which occurs during several of the preceding steps, e.g., the burning operation, preparation for dissolution and the actual dissolution of the fuel in nitric acid solutions, and which requires special attention.

There has been for some time considerable discussion and study of the importance of locations of the reactors and of the reprocessing plants. These important considerations center about the interrelations of the following: power transmission costs; irradiated fuel shipping costs and the probability of shipping accidents; and radioactive emissions, both from

the reactors and from the reprocessing plants. The hazard from radioactive emissions is dependent on the relative locations of the population and the sources of the emissions. Recognition of this fact is contained in the proposed changes to Title 10, Chapter 1, Code of Federal Regulations, Part 50. A summary of part of those changes and some comments on them are contained in Appendix A.

While the considerations in the preceding paragraph are of great importance in decisions on the future course of fuel reprocessing technology development, they are still being discussed and studied, and will not be considered further in this fairly specific discussion of fuel reprocessing.

REPROCESSING

Before beginning a discussion of HTGR fuel reprocessing per se, a few general comments on the national and international status of HTGR's and on their fuels are in order. Table 2 is a listing of the HTGR's in the world, and of some of their features. The fuels are described in a general way to show the similarities and differences in them, and thus to suggest the problems each presents to the fuel reprocessor.

With regard to the future of HTGR's, the purchase of two 1150 MW(e) HTGR's of the Fort St. Vrain Nuclear Generating Station type by Philadelphia Electric Company which was announced in August 1971, heralded what is probably the commercial breakthrough for HTGR's. Subsequently, an additional 5400 MW(e) in the form of 6 reactors have been sold, and one utility has an option to buy 2320 MW(e) more. Also, the Federal Republic of Germany, along with several German companies, will build a 300 MW(e) pebble-bed reactor of AVR type. Its projected completion date is about 1976. It is to be expected that this show of confidence in the HTGR by power utilities and other commercial firms will act as a catalyst to the sale of still others in the near

Table 2. HTGR's and Their Fuels

Dragon Reactor Experiment (near Bourenmouth, England)

Date of criticality: Summer 1964

Power: 20 MW(th)

Number of elements: 37

Fuel description: The 37 fuel elements each consists of seven rods comprising almost solid graphite end pieces and a graphite tube in the core region. Annular fuel compacts are stacked around a central spine inside the graphite tube. The graphite tubes are hexagonal in outer section, held apart by ribs down the middle of each flat surface. The entire seven-rod cluster assembly is 100 inches long; individual fuel rods are just under 90 inches long. The annular fuel compacts contain carbides of thorium and uranium in the form of about 400 micron diameter spheres coated with two layers of pyrocarbon separated by a silicon carbide layer. The coated spheres are held in a carbon matrix. On the average, each rod contains about 54 grams of uranium and 162 grams of thorium, although the thorium was concentrated in a small number of rods (about 70) for experimental purposes.

Peach Bottom HTGR (63 miles southwest of Philadelphia, Penn.)

Date of criticality: March 3, 1966

Power: 115 MW(th); 40 MW(e)

Number of elements: 804

Fuel description: The fuel elements are essentially cylinders 144 inches long and 3.5 inches in diameter. They are made up of graphite sleeve, annular fuel compacts that contain pyrocarbon-coated thorium-uranium carbide spheres in a graphite matrix, a central graphite spine, and upper and lower graphite reflectors. Each element contains about 275 grams of uranium and about 2100 grams of thorium.

AVR Experimental Station (near Julich, West Germany)

Date of criticality: 1967

Power: 50 MW(th); 15.6 MW(e)

Number of elements: Approximately 100,000

Fuel description: The AVR is what is called a "pebble-bed" reactor. The "pebbles" are the fuel elements. They are graphite spheres of 6 cm diameter which contain pyrocarbon-coated particles of the carbides of thorium and uranium. Each pebble contains uranium and thorium as the carbides in the ratio of about 1 to 5, with the total weight of metal in each sphere being somewhere between 5 and 12 grams. This reactor is no longer active, but is to be supplanted by the THTR, a 300 MW(e) advanced version of the pebble-bed reactor.

Fort St. Vrain Nuclear Generating Station (near Platteville, Colorado)

Date of criticality: Scheduled for fall 1972

Power: 842 MW(th); 330 MW(e)

Number of elements: 1482

Fuel description: The standard fuel elements are right parallelopipeds of hexagonal cross section. They are 31.2 inches high and 14.2 inches from one side of the hexagon to the opposing one. There are 212 fuel holes, six "poison" holes, and six coolant holes all of 0.5 inches in diameter, and 102 coolant holes 0.625 inches in diameter, all of which run parallel to the long dimension of the elements. The fuel holes contain fuel spheres of fissile and fertile materials coated with two pyrocarbon layers separated by a silicon carbide layer. The coated spheres are bonded together in a carbon matrix. Each element contains about 13 kg of thorium and 600 grams of uranium.

future. It is also not unreasonable to expect other reactor manufacturers than Gulf General Atomic (GGA) to consider marketing HTGR's.

The feature which sets HTGR fuel apart from all other reactor fuels is the very large amount of carbon which is directly associated with the fertile and fissile materials. Thus, the weight ratio of carbon-to-thorium in a fuel element of the Fort St. Vrain Nuclear Generating Station type is about 225. The basic problem in HTGR fuel reprocessing is removal of the carbon from the actual fuel material, i.e., uranium and thorium. The principal method being studied to remove this large amount of carbon from the relatively small amount of fuel material is burning.

It is apparent from Table 2 that HTGR fuels come in an extraordinary variety of forms and sizes. For definiteness, this discussion will be based on the type of fuel in the Fort St. Vrain Reactor. It is closely similar to the fuel that will be used in the large HTGR's to be built by GGA in the U.S., and is the type being used in the USAEC's and GGA's HTGR fuel recycle development program.¹ The fuel elements themselves are shown in Fig. 1. As indicated in Table 2, the Fort St. Vrain Reactor fuel elements have fuel holes which contain tiny coated particles of fissile and fertile materials.

To gain an appreciation of the reasons for the principal steps in reprocessing HTGR fuels it is necessary to go a little more into detail in describing the fuel particles and what takes place when they are irradiated in the reactor. It is also important to recognize that the HTGR is not a breeder reactor, and so must have a continuing source of fissile material (^{235}U — or perhaps ^{239}Pu) external to itself. Thus, fuel for large HTGR's will likely contain the following three types of coated particles:

ORNL DWG 71-9528

CONTROL FUEL ELEMENT

STANDARD FUEL ELEMENT

FIG. 1. FORT ST. VRAIN NUCLEAR GENERATING STATION FUEL ELEMENTS

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0

END TYPING

1. Fissile particles containing only ^{235}U compounds,
2. Fertile particles containing only ^{232}Th compounds,
3. Recycle particles containing both ^{233}U and ^{232}Th compounds.

Figure 2 is a diagrammatic representation of these particles. The purpose of the fissile particles of ^{235}U compounds is to make up for the fact that HTGR's do not breed. Approximately half of the weight of fissile material in large HTGR fuels will be ^{233}U which has been produced by operating that HTGR or a similar one. The ^{233}U comes from the fertile ^{232}Th according to the reaction

The other half will be essentially ^{235}U which has come from a uranium isotope enrichment plant. When ^{235}U is irradiated in a reactor, the bulk of it undergoes fission. However, several other nuclear reactions occur also, as shown in the following equation:

The important feature to note is the formation of ^{236}U . The cross-section it presents to neutron capture, while seemingly rather small, becomes very significant at high burnups of the fissile particle of ^{235}U compound. As a matter of fact, after several cycles of a given initial charge of ^{235}U through the reactor, the ^{236}U content results in an economically unacceptable parasitic loss of neutrons. The approximate composition of the ^{235}U particle

Fig. 2 DIAGRAMMATIC REPRESENTATION OF FISSILE, FERTILE,
AND RECYCLE PARTICLES

after one reactor cycle is: ^{235}U - 30%; ^{236}U - 50%; ^{238}U - 20%. The most profitable uses neutrons in a power reactor can be put to are producing fissions and producing fissile atoms from fertile atoms. Neutrons used in producing ^{237}U are wasted neutrons. Therefore, the uranium from particles containing ^{235}U should not be cycled more than once, or twice at the most. This is not true, however, of uranium from particles containing ^{233}U . That uranium may be recycled virtually indefinitely without incurring a significant cost penalty from parasitic loss of neutrons.

We thus arrive at a point where it becomes evident why the separation of ^{235}U from ^{233}U particles is one of the principle steps in HTGR fuel reprocessing. If the particles are permitted to mix so that the uranium isotopes are mixed, there is no economical way to separate them again. Thus, the ^{233}U would become "contaminated" with ^{236}U , and incur the concomitant cost penalty.

One device used to reduce the magnitude of the fuel particle separation problem is to fabricate two types of fuel elements. The first type contains only fuel particles of types 1 and 2 from the list above. The second type contains only fuel particles of types 2 and 3, i.e., it contains ^{235}U . In this approach, the separation problem only exists for fuel elements of the first type. The ^{233}U produced by neutron captures in particles of type 2 must be maintained separate from the burned-up type 1 particle. The fuel fabricator cooperates to make the necessary separation possible by making the two types of particles of different sizes, and different coatings. It is important that particle type 1 be enough larger or smaller than particle type 2 that it can be separated by a simple physical process; such as sieving. In practice, particle type 1 is made smaller,

as was shown in Fig. 2. It may be observed at this time, for reasons that will be obvious later, that not only are the outside dimensions of particles types 1 and 2 different, but also the diameter of the silicon carbide-coated $^{235}\text{UC}_2$ kernel is different from the diameters of the bare kernels of the other two particles.

A generalized diagram of the HTGR fuel reprocessing operations is shown in Fig. 3. This diagram is elaborated in Fig. 4, which emphasizes head-end operations, and also shows the solvent extraction step. Not shown are the fuel shipping step and the methods of handling wastes. In general, HTGR fuels will be stored ("cooled") for up to a year at the reactor before shipping them to the reprocessing plant. Figure 5 shows what the shipping cask may look like. Because of their long cooling times, the fuels pose no severe problems of fission product decay heat removal during shipment, in sharp contrast to fast reactor fuels, where fission product decay heat removal is a major problem in shipping. Table 3 shows how total heat generation and radioactivity from some of the fission products which are important in fuel reprocessing change with time. It has been calculated that the shipping cask outer surface temperature will not exceed about 65°C for 100-day cooled, fully-irradiated Fort St. Vrain Reactor fuel.

BURNER FEED PREPARATION

The very large amount of carbon present in the HTGR fuel imposes the problem of dealing with a very large amount of carbon dioxide gas when that carbon is burned. As was mentioned earlier, burning is the method that will be used to separate the uranium- and thorium-containing fuel particles from the carbon. In the reference process this separation is accomplished by crushing the fuel elements first in a large jaw crusher, which reduces the

FIG. 3 SCHEMATIC REPRESENTATION OF MAJOR REACTOR FUEL REPROCESSING OPERATIONS

Fig. 4. A GENERALIZED DIAGRAM OF ACTIVITIES IN THE HEAD-END STEPS OF THE REPROCESSING DEVELOPMENT

NOTE: LINES
OF TYPE

10
9
8
7
6
5
4
3
2
1
0

MARGIN

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CASK AND EXTENSIBLE TRAILER

15

Fig. 5 A SCHEMATIC REPRESENTATION OF AN HTGR
SPENT FUEL SHIPPING CASK

TABLE 3. VOLATILE FISSION PRODUCT CONTENT AND TOTAL DECAY
HEAT OF SPENT HTGR FUELS^a

DECAY TIME (DAYS)	NOBLE GASES (CURIES/TON)	IODINE (CURIES/TON)	CESIUM (CURIES/TON)	RUTHENIUM (CURIES/TON)	TOTAL DECAY HEAT (kW/TON)
(100,000 MWd/TON, 36 MW/TON)					
30	109,400	71,000	1,143,000	—	65.7
90	63,740	785	1,064,000	—	37.6
150	63,000	18	1,021,000	130,600	28.2
365	60,600	—	889,000	56,100	16.4
3,652	34,000	—	278,000	—	3.4

CURIES OF TRITIUM PER TON: 1,247 (COULD BE TRIPLED BY 7 ppm Li IMPURITY)

CURIES OF CARBON-14 PER TON: 2.6 (COULD BE TRIPLED BY 75 ppm N₂ IMPURITY)

^aTHESE VALUES ARE CALCULATED FOR FORT ST. VRAIN NUCLEAR GENERATING STATION FUEL IRRADIATED FOR 6 YEARS AT THE HEAT RATING AND TO THE BURNUP SHOWN. A DECAY TIME OF ABOUT A YEAR IS EXPECTED BEFORE REPROCESSING.

fuel elements to chunks smaller than about 1/2 inch. Figure 6 shows one kind of equipment used to carry out studies on crushing unirradiated fuel. Two subsequent crushing steps reduce the fuel to a size suitable for burning in a fluidized bed. The key consideration in this approach to carbon removal is, "What fraction of the coated fuel particles is broken?" This number must be small if the ^{233}U and ^{235}U are to be kept separate.

FUEL ELEMENT BURNING

The silicon carbide coating on the fuel particles serves two important functions. It prevents excessive fission product release from the fuel during irradiation in the reactor, and it preserves the integrity of fuel particles during burning to remove excess carbon. From Fig. 2 it is apparent that the $^{235}\text{UC}_2$ particle has a diameter of 280 microns out to and including the silicon carbide coating. (The carbon outside the SiC layer will be removed during burning.) This size allows the $^{235}\text{UC}_2$ particle to be separated from the other two particles by a sieving operation. The carbon coatings on both of the other particles may be expected to be completely burned off. As a matter of fact, the ThC_2 particle will surely be completely destroyed, and reduced to a powder during burning, because ThC_2 is highly reactive with oxygen. Thus, it will become a fine powder of ThO_2 . The $(\text{Th}, ^{233}\text{U})\text{O}_2$ particle may be expected to retain its integrity during the burning operation. It is already a refractory oxide, and even with a substantial fission product buildup should be strong enough to survive the burning step. So it is expected that, after burning, the $^{235}\text{UC}_2$ -silicon carbide-coated particle can be separated by sieving from the $(\text{Th}, ^{233}\text{U})\text{O}_2$ particle, which is substantially larger; and the " ThC_2 particle,"

FIGURE 6

CLASSIFICATION
OF MATERIALS

which is in reality a fine powder of ThO_2 . Both the ThC_2 and $(\text{Th}, {}^{233}\text{U})\text{O}_2$ particles will contain substantial amounts of fission products and newly-made ${}^{233}\text{U}$ (for example, the ThC particle will contain about 97.2% Th, 1.8% ${}^{233}\text{U}$, and the remainder largely fission products plus higher uranium isotopes).

The type of burner that will be used is not yet decided. Figure 3 shows that burning in a fluidized bed is the principal approach being studied. In this case, the material which is fluidized is the crushed fuel itself. It acts as an efficient heat transfer medium to carry the heat of combustion from the burning carbon to the wall of the burner, where it may be removed by use of a fluid coolant. If the heat is not removed efficiently, local overheating in the burner may lead to rupture of fuel particles or to holes in the burner wall.

PARTICLE CLASSIFICATION

After the carbon has been removed by burning, the sieving, or classifying, operation is carried out. The ${}^{235}\text{UC}_2$ -silicon carbide coated kernels are packaged and sent to solid waste disposal. The high burnup of the ${}^{235}\text{U}$ in these kernels (>90% burnup) makes heat production by fission product decay a consideration in the design of the waste disposal container. The ThO_2 powder and the $(\text{Th}, {}^{233}\text{U})\text{O}_2$ kernels with their attendant neutron capture and fission products are combined after classification and sent to the dissolver in preparation for separation and purification by solvent extraction.

OFF-GAS DECONTAMINATION

The present trend toward decreasing the permitted release of radioactivity into the environment, as reflected by the proposed amendments in Appendix A, is having an impact on nuclear reactor fuel reprocessing studies,

even though as yet no new restrictions have actually been imposed on reprocessing plants. The very large volume of carbon dioxide from the burner, with its associated relatively very small volumes of radioactive gases, creates a challenging problem for fuel reprocessing development. Whatever the process chosen to decontaminate the burner off-gas, it should not only meet the present gaseous effluent cleanup requirements, but also had better have the inherent capability of meeting future, more stringent requirements, of the type proposed for water-cooled power reactors. A large variety of gas cleaning processes are being studied.² A process which appears to hold promise of meeting the requirements for CO₂ decontamination is being studied at ORNL.³ It is based upon the fact that carbon dioxide may be liquefied at easily obtained temperatures and pressures, and thus, impurities may be removed by conventional chemical engineering processes such as distillation and vapor-liquid distribution. The radioactive gases to be removed from the carbon dioxide, primarily ⁸⁵Kr, have different vapor pressures from carbon dioxide, and so distribute themselves between the liquid and gaseous carbon dioxide in a way that makes separation possible. Analysis of this process shows that, under reasonable conditions, only about 0.1% of the krypton should leave the system with the decontaminated gaseous carbon dioxide waste. Although a large fraction of the xenon may be released to the environment by this process, it is of little consequence, since it is virtually all stable xenon one year after discharge of the fuel from the reactor. Thus, of the many radioactive xenon isotopes, the longest-lived one has a half-life of only 34 days. The bulk of the xenon present is stable xenon formed by beta decay of iodine isotopes. In contrast, the half-life of ⁸⁵Kr is 10.6 years.

From Table 3 it may be seen that about 60,000 curies of noble gases (Kr + Xe) per ton of uranium plus thorium are present in fuel decayed one year. Thus, if 0.1% is released to the environment, as in the above process, then 60 curies are released per day by a one ton a day reprocessing plant.

URANIUM AND THORIUM SEPARATION AND PURIFICATION

The final step in fuel reprocessing takes the highly contaminated ^{233}U and thorium and separates them cleanly, both from each other and from fission products. These separations are performed by solvent extraction. The uranium and thorium oxides from the burner are dissolved in a nitric acid-aluminum nitrate-hydrofluoric acid solution. After adjustment of concentrations, the resultant aqueous solution is equilibrated with immiscible solutions of tributyl phosphate (TBP) in n-dodecane. The TBP solutions preferentially extract uranium and thorium nitrates, leaving fission product and other nitrates in the aqueous phase. Subsequently, concentrations are adjusted so that the thorium leaves the TBP phase to enter an aqueous phase, leaving only uranium in the TBP phase. Finally, the uranium is stripped into an aqueous phase for use in fuel particle fabrication operations.

APPENDIX A

Proposed Amendments to Title 10, Chapter 1
Code of Federal Regulations, Part 50
Licensing of Production and Utilization Facilities
Light-Water-Cooled Nuclear Power Reactors

The AEC regulations specify design and operating requirements for nuclear power reactors to keep levels of radioactivity in effluents to unrestricted areas as low as practicable. In the spring of 1971 specific numerical criteria were proposed as an amendment to the earlier regulation, which was qualitative in nature. Although the regulation was written to apply solely to nuclear power reactors, there would appear to be little basis for having one set of radioactive release criteria for reactors and another set for fuel reprocessing plants, since the intent in all cases is to protect people. Therefore, the proposed amendments are at the least indicative of the type of regulation likely to be written for reprocessing plants, and may in fact be very close to that regulation. Following is a summary of some of the important numerical criteria in the proposed amendment. General goals of the numerical design objectives are to:

1. Assure that annual exposures to individuals at site boundaries of reactors do not exceed 5% of annual average exposures from natural background radiation.*
2. Assure that annual exposures to sizeable population groups from radioactivity released in reactor effluents on all sites in the U.S. for the foreseeable future, e.g., the year 2000, will generally be less than 1% of exposures from natural background radiation.

*Average exposures due to natural background radiation in the U.S. are in the range of 100 to 125 millirems per year.

A useful measure of the exposure to a large number of persons is the man-rem. Man-rem's are calculated by multiplying the number of people exposed by the average exposure, in rem's, of the individuals. The proposed man-rem standard for each 1000-MW(e) unit is 400 man-rem's whole body dose to the total population exposed. Average exposures to large population groups would be less than 1 millirem per year. Dose limits remain at 0.5 rem per year for individuals and 0.17 rem averaged over the entire population.

Liquid and gaseous effluents individually or together from a nuclear power reactor are not to produce exposures exceeding 5 millirems per year to all or any part of an individual in offsite areas. Total gaseous effluents released from a site should not be likely to result in average individual exposures exceeding 10 millirems annually from noble gases anywhere offsite. The exposure rates of 10 millirems annually from noble gases and of 5 millirems integrated whole body doses from all pathways or sources are not necessarily mutually exclusive because of the nature of the noble gas radiation (i.e., it is weak radiation, and largely localized on the skin).

For radioactive materials above background in liquid effluents to be released to unrestricted areas by each light-water-cooled nuclear power reactor at a site, the following criteria must be met:

1. The estimated annual total, except tritium, should not exceed 5 curies.
2. The estimated annual average concentration before dilution in a natural body of water should not exceed 2×10^{-8} curies/ml, except tritium, which should not exceed 5×10^{-6} curies/ml.

For radioactive materials above background in gaseous effluents, the estimated total quantities of radioactive materials to be released to unrestricted areas by all light-water-cooled nuclear power reactors at a site should not result in:

1. An annual average exposure due to noble gases anywhere off the reactor site in excess of 10 millirems.
2. Annual average concentrations anywhere off the reactor site of particulate radioactive material with a half-life greater than 8 days, or radioactive iodine in excess of the concentrations in air specified in Appendix B, Table II, Column I. of 10 CFR Part 20, divided by 10^5 .

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