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ABSTRACT

This evaluation is part of a project that has the objective to select a new method for radioactive waste disposal at the IEA The method used at present is the dilution in 10 000 liters of water We discuss the storage for decay before the dilution

We try to establish a production grown pattern for radioactive waste

We do a estimate for the next ten years about the growth of radioactive wastes produced routinely at the IEA

I – INTRODUCTION

The modifications to the IEAR 1 reactor to increase its power and the expansion plans of the I E A require a study of the problem Radioactive Waste Disposal This problem was studied previously by Rodrigues in the years 1963 and 1964 His two reports on the subject were published in 1964[8] and 1965[9] but on that occasion more emphasis was given to the sanitary engeneering part that specifically to the disposal part Also at that time the radioactive wastes quantities were a lot lower

From 1964 until now the IEA did not grow in size but the present expansion will continue in a accelerated rythm at least for the next few years

The recent alterations in the IEAR 1 reactor installations and the modifications that are at present in progress as well as the start up soon of the new Radioactive Materials Processing Service installation will cause a big increase in radioactive waste production In view of this it became necessary study a new method appropriate for radioactive waste disposal This method must take into account the increasing rythm of waste production that becomes effective each year in present and future production considering the power increase of our reactor

It is obvious that this future review must be done with certain flexibility to avoid that the selected waste disposal method became obsolete in the next years

We did a meticulous survey in the archives of the several divisions and services to collect dates that we can carry to an estimation of the radioactive materials produced or adquired that could result in immediate or subsequent waste We limitated ourselves only to those division and services that actually have waste problems leaving out those that have relativaly little quantities or those for which the work does not give radioactive wastes We considered only the waste produced in a routine manner and not those originated in a special research of short duration because with these dates we will make provision for the future

The principal dificulty in the quantitative evaluation of the radioactive waste lies in the indirect manner of its determination We found records about radioactive material production and about the production processes yield but not about what was left over as waste From the yield sometimes

theorethical of the irradiated substance as well as its mass it was possible to evaluate within the precision that such a type of method gave us the quantities of radioactive wastes

II – RADIOACTIVE WASTES SURVEY COMING FROM THE RADIOACTIVE MATERIALS PROCESSING SERVICE (S P M R)

In this service are produced in a routine manner several radiosotopes that need a chemical processing after the irradiation and that thus give us a large quantity of waste

The first radioisotope produced in a routine manner was ^{131}I which was started in the second semester of 1959 During the first three months the target material was telluric acid irradiated in the reactor operating in 50 KW during the last three months in 500 KW

By the comparison between the production of July and of December (Table I) we see a clear increase in the activity produced as waste in view of the increase of the reactor power The cause has been not only that but also the increase in the mass of irradiated material and the time of irradiation The increase in reactor power assuming the others factors to be constant should only be responsible for a factor of approximately 10

Table I

Month	Produced Activity (mCi)	I 131 wasted Activity (mCi)	Reactor Power
July	$37 \quad 10^{-2}$	$0\,84 \quad 10^{-2}$	50 KW
August	$372 \quad 10^{-2}$	$0\,28 \quad 10^{-2}$	50 KW
September	$9\,4 \quad 10^{-2}$	$0\,34 \quad 10^{-2}$	50 KW
October	10 409	1 598	500 KW
November	7 616	1 066	500 KW
December	5 58	0 282	500 KW
Total	23 773	2 961	
	10 0	2 5	

As already explained this table has been compiled from the S P M R archives, but in the last row we put the total production of I 131 in 1959 as noted in the Gonçalves da Silva report[11] The difference between the values derived by us and those of the report showld be due to the fact that some processing are not been considered as effective production

For us from the point of view of wastes the result of our investigation is more significant than the result of Gonçalves da Silva

It is called produced activity that part that results after the separation of the desired radioisotope that is made in an irradiated material solution and wasted activity that remains in the solution and has not been separated In other words the produced activity is a yield function and the wasted activity is its complement

In the third column of Table 1 the wasted activities are listed Since 1960 both produced activity and yields have been recorded From these data (yield for I 131 is 80%[2 5,7]) we have the wasted activity During that year the power of the reactor was 500 KW but in November and December the power was increased respectively to 1 MW and 1 5 MW

Since 1961 the reactor power has remained at 2 MW

In table 2 we show the I 131 wasted activities for the period 1959 to 1971 inclusive

Table II

Years	Wasted Activity (mCi)
1959	1 3
1960	10 8
1961	73 5
1962	1584 2
1963	2575 8
1964	2874 0
1965	3091 0
1966	4142 0
1967	4868 0
1968	4740 0
1969	5167 0
1970	6146 5
1971	7567 0

II 1 – INCREASE OF THE WASTED ACTIVITY IN I 131

We can see that there is a continuous increase with only a fluctuation between 1967 and 1968 which may have been caused by the change of the target from telluric acid to elemental tellurium There must have been in the first chemical processing a decrease in the yield

Figure 1 shows the same increase normalized to 1 in the first year We can see that the increase is exponential

The second radioisotope to be produced in a routine manner was the P 32 and the first datas founded were from July 1961

Until the end of 1966 the target material was magnesium sulphate when then is been changed for elemental sulfur

In this case also the yield is 90% therefore the wasted activity 10%

During the years (table III) there were fluctuations but we note that in general there was an increasing production

Table III

Increasing of the Wasted Activity in ^{32}P

Year	Wasted Activity (mCi)
1961	43 1
1962	130 0
1963	128 0
1964	118 0
1965	210 0
1966	104 0
1967	262 0
1968	213 0
1969	297 0
1970	237 0
1971	427 0

Again by the figure 2 we see an exponential increase

The third radioisotope to be produced was ^{198}Au and this production was been started in 1964

The target material is metallic gold

In the case of the ^{198}Au theoretically there are no wastes because the chemical processing yield is 100% but is has been considered that 1% of the total activity is lost in the flask washing wastes etc Therefore this last value is considered as wasted activity

By the table IV we see that there are fluctuations in the ^{198}Au wasted production but in a general manner the increase is also exponential (figure 3)

Table IV

Increasing of the Wasted Activity in the ^{198}Au

Years	Wasted Activity (mCi)
1964	53 1
1965	6 8
1966	170 3
1967	198 2
1968	310 8
1969	306 5
1970	420 5
1971	564 0

In that same year of 1964 ^{24}Na was produced also but it is not interesting for our analysis, because the waste is insignificant Also ^{35}S of which production is very irregular can be comproved in the table 5 or figure 4 For this radioisotope the yield is 80%

Table V

Increasing of the Wasted Activity in ^{35}S

Year	Wasted Activity (mCi)
1964	1 7
1965	2 5
1966	9 2
1967	18 5
1968	6 0
1969	7 2
1970	17 6
1971	60 0

In 1965 it was started to produce ^{51}Cr (table VI) and figure 5 The production reached a maximum in 1968 The yield is between 5% and 8 5% because the process used is Szilard Chalmers

Table VI

Wasted Activity in the ^{51}Cr

Year	Wasted Activity (mCi)
1965	17 1
1966	3249 0
1967	18316 0
1968	25313 7
1969	11479 8
1970	7467 0
1971	9120 0

Beyond these radioisotopes the S P M R produce ^{82}Br and ^{42}K that do not produce appreciable radioactive wastes

II 2 – CALCULATION OF THE TELLURIUM RADIOISOTOPE QUANTITIES FORMED DURING THE IRRADIATION

Of all radioactive effluents belonging to S P M R the most problematic of the point of view of disposal is the affluent of ^{131}I not because of wasted activity but by the several tellurium radioisotopes that are form during the irradiation

The natural tellurium contains the following isotopes

$$^{120}Te,\ ^{122}Te,\ ^{124}Te,\ ^{126}Te\ \ ^{128}Te\ e\ ^{130}Te$$

These isotopes during the irradiation form 10 important radioisotopes of the point of view of this evaluation according to the following reactions[10]

$^{120}Te\ (n,\gamma)\ ^{121m}Te \xrightarrow[154d]{I.T} {}^{121}Te \xrightarrow[17d]{E.C} {}^{121}Sb$ (stable)

$^{122}Te\ (n,\gamma)\ ^{123m}Te \xrightarrow[104d]{I.T} {}^{123}Te$

$^{124}Te\ (n,\gamma)\ ^{125m}Te \xrightarrow[58d]{I.T} {}^{125}Te$

$^{126}Te\ (n,\gamma)\ ^{127m}Te \xrightarrow[105d]{I.T} {}^{127}Te \xrightarrow[9.3h]{\beta^-} {}^{127}I$

$^{128}Te\ (n,\gamma)\ ^{129m}Te \xrightarrow[33d]{I.T} {}^{129}Te \xrightarrow[74m]{\beta^-} {}^{129}I \xrightarrow[1.7\ 10^7a]{\beta^-} {}^{129}Xe$

$^{130}Te\ (n,\gamma)\ ^{131m}Te \xrightarrow[30h]{I.T} {}^{131}Te \xrightarrow[25m]{\beta^-} {}^{131}I \xrightarrow[8.05d]{\beta^-} {}^{131}Xe$

The quantitative results of these reactions for a gram of irradiated elemental tellurium during a week (5 days or 4 days) in the reactor in the normal power are shown in table 7

Table VII

Isotopic	Isotopic Abundance %	Capture Cross Section (barn)	Activade Nuclide	Half Life	Activity week 5 days (mCi)	Activity week 4 days (mCi)
^{120}Te	0.089	0.3	^{121}Te	17d	0.023	0.019
		2.0	^{121m}Te	154d	0.018	0.015
^{122}Te	2.46	1,0	^{123m}Te	104d	0.368	0.285
		2.0				
^{124}Te	4.61	5.0	^{125m}Te	58d	5.898	4.746
		2.0				
^{126}Te	18.71	0.1	^{127m}Te	105d	0.268	0,215
		0.809	^{127}Te	9.3h	120.100	120.022
^{128}Te	31.79	0.017	^{129m}Te	33d	0235	0.190
		0.14	^{129}Te	74min	57.314	57.314
^{130}Te	34.48	0.008	^{131m}Te	30h	1.315	1.250
		0.22	^{131}Te	25min	97.218	97.218

In this table we show the isotopic abundances capture cross section and half life of the radioisotope formed

In this manner knowing the irradiated tellurium mass the reactor power that is constant and the irradiation time we could evaluate the radioisotopes quantities formed (table 8 9 and 10)

Table VIII

Year	Mass (g)	Irradiation time (h)
1967	1300	72
1968	2900	72
1969	3850	72
1970	5200	72
1971	3550 e 1850	72 e 144

Table IX

Radioisotope	Activity after 72h of irradiation (mCi)	Activity after 144h of irradiation (mCi)
^{121m}Te	0 032	0 062
^{121}Te	0 036	0 051
^{123m}Te	0 645	1 23
^{125m}Te	10 17	18 8
^{127m}Te	0 47	0 90
^{129m}Te	0 39	0 68
^{131m}Te	1 28	1 28
^{131}Te	97 3	97 3

Table X

Year	^{121m}Te	^{121}Te	^{123m}Te	^{125m}Te	^{127m}Te	^{129m}Te	^{131m}Te	^{131}Te
1967	41 6	46,8	838 5	13221 0	611 0	507 0	1664 0	126490 0
1968	92,8	104 4	1870 5	29493 0	1363 0	1131 0	3712 0	282170 0
1969	123,2	138,6	2483,2	39154 5	1809 5	1501 5	4928 0	374605 0
1970	166 4	187 2	3354 0	52884 0	2444 0	2028 0	6656 0	505960 0
1971	228 3	222 15	4565 2	70883 5	3333 5	2842 5	6912 0	525420 0

We have done these calculations only since the second semester of 1967 because it is only since that date that the reactor is mantaining the given time of operation and therefore the table 9 is valid More interesting to us is the waste produced at the present and that which will be produced in the future It is obvious that the wastes of the years previous to 1967 can be disregarded without damage to this work

Since the second semester of 1971 the irradiation time of the elemental tellurium increased from 72h to 144h This is the reason why two quantities are shown in table VIII

Since 1969 two weekly chemical processing has been started and continued until now in this rythm except that since 1970 the mass of tellurium for each irradiation is increased

In this evaluation the radioisotopes ^{127}Te and ^{129}Te have not been considered The ^{127}Te because it has a half life of 9 3h and therefore the activity acquired in the first week decays in the second that acquired in the second decays in the third and that acquired in the forth week decays in the first three days after the and of the irradiation and the start of the chemical processing The isotope son ^{127}I is stable The ^{129}Te decays rapidly to ^{129}I that has a long half life ($1\,7\,10^{7}$ years) and therefore this causes pratically no activity

In this evaluation then we have only considered the activities of the ^{121m}Te ^{121}Te ^{123m}Te ^{125m}Te ^{127m}Te ^{129m}Te ^{131m}Te and ^{131}Te

Yet with reference to the table 9 for the ^{131}Te that has a short half life and therefore decays rapidly we have only considered the activity of the last week

II 3 – CALCULATION OF ACTIVITIES OF TELLURIUM CONTAINED IN MONTHLY COLLECTED I 131 EFFLUENTS

Since the chemical processing to obtain I 131 is done weekly we calculate the weekly activity of the tellurium isotopes and the weekly wasted activity of I 131 considering the annual production and the number of weeks for a year Since the waste is collected each 4 weeks we calculate the total activity in each collection considering the decay of the three preceding processes Table XI gives us the activities at the moment of the collection

In the case of the I 131 we have only considered the wasted activity and not the extra part coming from the decay of the Te 131 because that part resulting from the decay is very small and does not influence the total activity at the waste delivery Furthermore we will see that even the wasted activity is negligible at the time the waste is delivered

Table XI

Radioisotope	Effluent Activity (mCi)
^{121m}Te	16 8
^{121}Te	11 7
^{123m}Te	327 5
^{125m}Te	4825 0
^{127m}Te	238 7
^{129m}Te	164 6
^{131m}Te	136 0
^{131}Te	10104 0
^{131}I	289 5

At the present the international norm for the radioactive waste delivery to the environment reccomends that there should be a study of the capacity that this environment has to receive the waste It is obvious that in such a study for liquid wastes we are concerned with an appreciable series of processes such waste infiltration in soil fixation in the food vegetable for men and for the animal that can be food for man fixation in these last volume and outlet of the stream river and even sea where whould be carried out these wastes and etc for which we do not have at the present numbers of personnal to execute them beyond that the waste quantities that we have whould no justify the amount spent for a such type of study

In view of this we prefer to adapt the oldest system of the concentration limits for drinking water These without any doubt are a lot more restrictive as we will see next giving therefore a major safety to the man and his environment

Considering Annual limit for the intake by ingestion (ALII) for public individual and that these on the average swallow 0 8m^3 of water per year[4] we can calculate the decay time and the dilluition necessarys to deliver all our liquid wastes within the safety standards

For the dilluition we have two retention tanks with the capacity of 10^6 liters each These tanks receive the water used in the laboratory sinks of the S P M R D R B and on the average are filled once each two days The activity concentration resulting from the washing in the own laboratory of this divisions is less than 10% of the concentration derived from the A L I I reason for which is negligible for our calculation

The way followed by the radioactive waste after delivery is Jaguaré stream of which the flow is 0 2 m^3/s Pinheros river whith a flow of 70 m^3/s Billings dam of which the flow is 100 m^3/s and estuary with a flow of 120 m^3/s There is therefore a dilution factor pratically endless

The gallery of pluvial waters that of the retention tanks goes to the Jaguaré stream is situated within the University City in a manner that there is no contact of the water with the vegetable food

In view of all this we establish the convention that we can deliver an activity concentration 10 fold greater than that derived from the A L I I and this we call Annual Limit Concentration (A L C)

The table XII show us the A L I I and the A L C values for the wastes delivered for all radioisotopes produced in the S P M R in a routine manner

Table XII

Radionuclide	A L I I for public individual (μCi)	A L C (μCi/mℓ)
^{32}P	1 5 10	2 10^{-4}
^{198}Au	4 1 10	5 10^{-4}
^{35}S	5 0 10	6 10^{-4}
^{51}Cr	1 3 10^3	2 10^{-2}
^{131}I	1 6	2 10^{-5}
^{131m}Te	4 6 10	6 10^{-4}
^{129m}Te	2 6 10	3 10^{-4}
^{127m}Te	5 0 10	6 10^{-4}
^{125m}Te	1 3 10	2 10^{-3}
^{123m}Te	2 6 10	3 10^{-4}
^{121}Te	2 6 10	3 10^{-4}
^{121m}Te	2 6 10	3 10^{-4}

In the case of the tellurium radioisotope es the I C R P (International Commission on Radiological Protection) and the I A E A (International Atomic Energy Agency) do not give the A L I I for the ^{123m}Te ^{121}Te and ^{121m}Te isotopes and then adapt as limit the most restrictive values between the other radioisotopes in this case the ^{129m}Te

The A L C for a mixture of tellurium radioisotopes was calculated by dividing the concentrations of each radioisotope by its A C L and summing the quotients the result must be less than or equel to 1

After 18 radioactive decay months of the tellurium wastes we will have the concentrations given in the table XIII and that gives a mixture A L C 0 947 if the material is delivered in 5 fractions

Table XIII

Radioisotope	Concentration (μCi/mℓ)
^{121m}Te	$1\,38 \quad 10^{-4}$
^{121}Te	depreciable
^{123m}Te	$8\,5 \quad 10^{-4}$
^{125m}Te	$7\,24 \quad 10^{-4}$
^{127m}Te	$6\,44 \quad 10^{-4}$
^{129m}Te	depreciable
^{131m}Te	depreciable
^{131}Te	depreciable
^{131}I	depreciable

II 4 – DECAY TIME CALCULATIONS FOR THE OTHER WASTED RADIOISOTOPES OF THE S P M R

For this calculation are also take into account the decay time and the dilution in 10^4 liters of water The obtained results are shown in Table XIV

III – RADIOACTIVE WASTES SURVEY COMING FROM THE RADIOPHARMACY SERVICE BELONGING TO RADIOBIOLOGY DIVISION

In this Service are done in a routine manner labeled compounds using ^{131}I and ^{51}Cr The most common are Hippuran Sorum Albumin Macro Aggregated Sorum Albumin Rose Bengal Oleic Acid Triolein Human Sorum Albumimine Brominesulfaleine and EDTA

It has been possible to evaluate the quantity of ^{131}I wastes produced yearly by knowing that the label compound production is weekly and that elapse two days of decay between the delivery of the iodine by the S P M R at this Service and knowing also the labeled compound production

By the S P M R records we had the delivery activities and by the D R B records we had the label compound activities

With these dates it has been possible to build the table XV that shows us the ^{131}I wasted activities of this Service and the figure 6 in porcentual increasing scale as we have done for the S P M R

For the ^{51}Cr we have considered that 80% of the total activity is retained and 20% wasted The results are in the table XVI

We do not done the figure because the activities are very small in regard to the others and have a completely random distribution

As in the case of the S P M R these wastes are collected each 4 weeks in a manner that after the decay calculations in the moment of collection we have 195 mCi of ^{131}I and a insignificant activity of ^{51}Cr Diluting that activity in 10^4 liters of water we will have a concentration of 19.5×10^{-3} μCi/mℓ How the A L C is 6×10^{-3} μCi/mℓ we store this effluent for two weeks before his delivery

Table XIV

Radioisotope	Half Life	Effluent Activity	Decay Time
^{32}P	14 6 d	20 8 mCi	50 d
^{198}Au	2 7 d	12 5 mCi	4 d
^{35}S	87 0 d	4 24 mCi	directly delivered after dilution
^{51}Cr	27 0 d	*1518 8 mCi	80 d

* Based in the maximum production that was in 1968

Table XV

Year	Wasted Activity (mCi)
1965	802 8
1966	1352 7
1967	1448 1
1968	2403 0
1969	2794 5
1970	2963 7
1971	4135 5

Table XVI

Year	Wasted Activity (mCi)
1966	3 27
1967	1 95
1968	7 27
1969	0 91
1970	6 55
1971	15 80

IV – RADIOACTIVE WASTE SURVEY FOR THE CHEMISTRY ENGENEERING DIVISION D E Q.

The radioactive waste produced by the D E Q are coming from natural radioisotopes and are not artificial as that already treated

Departing from the sodium diuranate (SDU) produced by Monazite Production Administration the D E Q obtains uranium nuclearly pure under the form of amonium diuranate (ADU)

In the D E Q initially his been built a Pilot Plant for Uranium Purification by ionic exchange This Plant entered in effective operation in january 1969 and stopped production in march 1970 being replaced by another that operates on the basis of Solvent Extraction in a pulsed column Intrance operation began in July 1970

The total production of ADU by the Ionic Exchange Pilot Plant has been above 1500 Kg[3] In this purification process there are several losses of uranium but they are almost all recoverable because they are losses in the processing yield The only losses that can be considered effective are wasted paste resulting from the dissolution of approximately 90 g of uranium thorium oxalate paste with epproximetely 100 g of uranium and a effluent resulting from the sorption (charge) operation of the uranium which volume is approximately 80 liters with a uranium concentration lower than 20 p p m

The two first are stored in a dum and the last goes to the drain

Considering that in each operation beginning with 60 Kg of SDU containing 40 8 Kg of uranium in the form of SDU 36 5 Kg of uranium in the form of ADU and 190 g of wasted uranium[1] we have that in each Kg of produced uranium of nuclear quality there is 5 2 g of wasted uranium

The table XVII give us the quantity of ADU produced by this process the content of uranium in the ADU and the quantity of uranium wasted in grams and in activity The quantity of ADU has been calculated from the production control maps and the quantity of uranium dividing the grams of ADU by 1 3 The activity of the wasted uranium is been calculated knowing that a $3\,2\,10^{6}$g of uranium corresponds to a 1 Ci of activity[6]

Table XVII

Wasted U coming from the Ionic Exchange U Purification
(year of 1969 and first quarter of 1970)

Time	Produced ADU (Kg)	Produced U (Kg)	Wasted U (g)	Wasted U (mCi)
1969	1157 6	890 462	4630 4	$1\,4\ \ 10^{-1}$
1970 (1st quarter)	428 8	329 846	1715 2	$0\,5\ \ 10^{-1}$
TOTAL	1586 4	1220 308	6345 6	$1\,9\ \ 10^{-1}$

About the Pilot Plant Uranium Purification by Solvent Extraction in Pulsed Column there is not wasted uranium production because the stages in which it is possible to have losses are of much higher yield than the corresponding over of the other uranium purification process private communication Another evaluation to be done is the point of view of the uranium purification we are not concerned with the losses of thorium because we will have to do quantitative determinations of these losses These

determinations will constitute on experimental part of this work For this evaluation we are thinking of determining

1 – thorium content in the dissolution paste of the SDU with HNO_3

2 – thorium content in the precipitates from aqueous phase of the process After the SDU dissolution the thorium goes in the uranyl nitrate From there this aqueous solution enters in contact with an organic phase that will take out the uranium leaving the thorium in aqueous solution This last is treated with NA OH that produces a quantitative precipitation of the thorium together with the other impurities that were in the SDU This precipitated thorium that must be determined

V – OTHER RADIOACTIVE WASTES

We will start with the liquid wastes coming from the reactor building In the basement where there is the heat exchanger and the ionic exchange resin for the swiming pool water purification there is also a sump for the used water of the building

Into this tank that has 800 liters capacity come all the waters from the building sinks and nozzles as well as the washing waters of the ionic exchanger resins

There is a maximum and minimum level for the delivery of the waters of this sump in such a manner that are delivered at once 500 l ters In the top of the sump there is a jordan ionization chamber with the measurement panel in the control room of the reactor wich records the activity level of water In the activity level is high this water is transfered into another tank of larges capacity for decay In the other event it is delivered into a network of storm sewers that goes into the Jaguaré Stream Up till the present time it has never been necessary to transfer the water into the decay tank and the delivery of the water into the storm sewers is done on the average three times a day

The solid wates produced in the reactor building in general consist of irradiated elements and structural elements and until the present momem they are mantained in the storage tubes for decay In these storages tubes among others we meet a lot of reflector elements two irradiation elements for telluric acid solution (EISAT) a thermopile with his supporting arrangement several rabbits with glass sand a lot of ^{192}Ir and ^{152}Eu sources UO_2 pellets a plate of the Argonaut reactor with its support and several capsules with unidentified material

In the several I E A Divisions and Services including the DOMR we mantain heavy paper bags with double walls to collect radioactive wastes such as contamined filter papers contaminated dusters and wipes papers to dry contaminated surfaces (of the blotting paper type) contaminated paper towels etc On these bags we write the name of the laboratory where they are delivered date of the collection for ease identification These bags are changed whenever the local radiation level becomes raised for the workers or when they are filled The colected bags are stored for the decay and are burned only after they have reached an exposure level of 0 5 mR/h close to the bag

The ashes are buried in an appropriate place

Since 1971 we decided simply for easing the work to do two burnings and burials a year In the table (VIII .e give the number of bags collected and burned since 1967

Table XVIII

Time	Nº f bags colected	Nº of bags burned
1967 1st semester	15	–
1968	25	–
1969	25	79
1970	46	38
1971	132	132

The sudden increase in 1971 of the number of bags of radioactive wastes is related to the following facts

1 — We started to collect weekly the bags that stay in the labeled compound laboratory of the D R B because in view of the increase weekly production of this compound the radiation level in that laboratory is exposing the laboratory workers unecessarely

2 — We started also in 1971 to collect the waste bags of the D M N and D E Q because until that date the laboratory personal solved the problem of theirs own wastes

Beyond this type of wastes there are others such as contaminated glasses metals that were used as irradiation covers that were damaged crystals of irradiated chemical compounds that were not used plastics rubber etc that are not burned but only buried in the place already mentioned after radioactive decay

Before finishing we have yet to mention the gas waste processing cells and the laboratory where these cells are located and the reactor building principally the swimming pool All these places give us large radioactive gaseous waste only in the case of accident Normally their radioactive levels are small

The air of the radioisotopes processing cell is conducted by a system ducts to a chimney after passing through a series of absolute filters After the filters we put a Geiger Muller counter with rate meter and recorder We calibrate this instrument with ^{126}Ba ^{131}I and ^{60}Co sources and we can verify that roughly speaking the relation between counts per minute and exposure rate in mR/h is linear and that 30 000 count/min corresponds to 17 mR/h In this manner during the radioisotope processing the radioactive level reaches 15 000 counts/m or 8 5 mR/h and when there is no chemical processing the radioactive level falls to 4 000 count/min or 2 3 mR/h The filters are changed generally when the radiation level during the radioisotope processing reaches 20 000 count/m or 11 3 mR/h When we change the filters they are contaminated and show an exposure rate between 2 and 3R/h close to the filters

In the reactor building we also have a blower and air circulation system with the exhaust delivered to the chimney by ducts

The exhaust system in normal operating conditions has the objective of retaining in the ventilation filters the possible radioactive particles in suspension in the air This air after passing by the filters passes by a counter and leaves by the chimney With the reactor at 2 MW of power this counter records a mean level of 210 count/min or 0 1 mR/h There is also a contamination exhaust system used in the case of an accident With this ventilation system if an accident happeans there is a value that blocks the normal air entrance for the reactor building and at the same time starts the operation of the contaminated air exhaust system causing the air to pass initially by an absolute filter and then by an activated charcoal filter Thus the contaminated gases in the air are retained completely and the air leaves normally but the chimney also passing by the counter In the case of accident the possible contaminant agents in the air would be ^{131}I and ^{41}Ar

The radiation detection in the exhaust system will be modified as can be seen in the Sordi report[12]

VI — EVALUATION OF THE RADIOACTIVE LIQUID WASTE PRODUCTION IN FUTURE YEARS

As we saw in the preceding sections the type of waste that at the present gives us the most trouble is liquid waste since we are at the point that we need to select a new disposal method more practical than that presently used To select a new method that will not become inadequate in the next few years we need to do a more or less approximate evaluation of the quantities of these liquid wastes that we will have in the future

The reactor modification to operate at 10 MW of power is in progress and is being done in three stages At the present we are finishing the second part Once finished, the reactor will operate at 5 or 6 MW (and in the case of necessity 10 MW) but normally only after the conclusion of the third part will it operate at 10 MW Besides the power level we also believe that the operating time will be altered

The predicted program is

1 – Reactor power 5 or 6 MW with the operating time 36 hours weekly

2 – Reactor power 5 or 6 MW with the operating time 72 hours weekly

3 – Reactor power 10 MW with the operating time 120 hours weekly

4 – Reactor power 10 MW with the operating time 20 days without stop

Yet according to Penteado (Private Communication) the actual production of radioisotopes following the above program should be increased by

1 – a factor 2 5 caused only by the reactor power change

2 – a factor 5 being 2 5 caused by the power and 2 by doubling the operating time

3 – a factor 17 that would reach 20 in the case of necessity by the use of the new irradiation positions

4 – a factor 30 This should be the maximum considering not only the operation time but also that at the present 20 reactor positions are being used for irradiation and we have possibilities for 30 more In order to gain an idea we go on to consider that the forth possibility is remote enough and that therefore only the third possibility remains that is that the present production is increased by a factor of 20

We consider also that in the next years the rate of increase is given by the curves of the figures 1 2, 3, 4, 6, and 6 because according to the Pieroni and Barberio information we are already satisfying all the national market for several years and therefore there is not a jump in the growth rate caused by the reactor modification From these facts we can build table 19 where are given the produced radioisotopes, the Annual wasted activity in the saturated production and the time necessary to reach that saturation

Table XIX

Radioisotope	Wasted Activity (Ci)	Time (Years)
^{131}I (SPMR)	198 7	17
^{32}P (SPMR)	8,3	25
^{198}Au (SPMR)	13,8	10
^{35}S (SPMR)	2,0	12
^{51}Cr (SPMR)	580 0	12
^{131}I (Radiopharm)	112 4	13

From the obtained results we believe that a waste disposal method that solves this problem in the next ten years, will be satisfactory Again by the extrapolation of the curves mentioned above we built the table XX that give us the predicted wastes quantities in 1981

Table XX

Radioisotope	Wasted Activity (Ci)
^{131}I (SPMR)	48 1 ± 5 2
^{32}P (SPMR)	1 01 ± 0,28
^{198}Au (SPMR)	13 0 ± 2 1
^{35}S (SPMR)	0 72 ± 0 44
^{51}Cr (SPMR)	299 0 ± 214 0
^{131}I (Radiopharm)	48 7 ± 7 4

The excessively big error that ^{51}Cr waste production presents comes from the fact that the production of this radioisotope has undergone many fluctuations as can be seen in the figure 5

We must therefore to select a disposal method for the quantities given in table XX

This will just be the step that we intend to make next that is to study the different systems that there are for waste disposal and the afterwards select the one most profitable for us

RESUMO

Esta avaliação é parte de um projeto que tem como objetivo selecionar um novo método para a disposição de resíduos radioativos no IEA O método usado atualmente é a diluição em 10 000 l de água Discutimos o armazenamento para decaimento antes da diluição Tentamos estabelecer o padrão de crescimento para os próximos 10 anos do crescimento dos resíduos radioativos produzidos rotineiramente no IEA

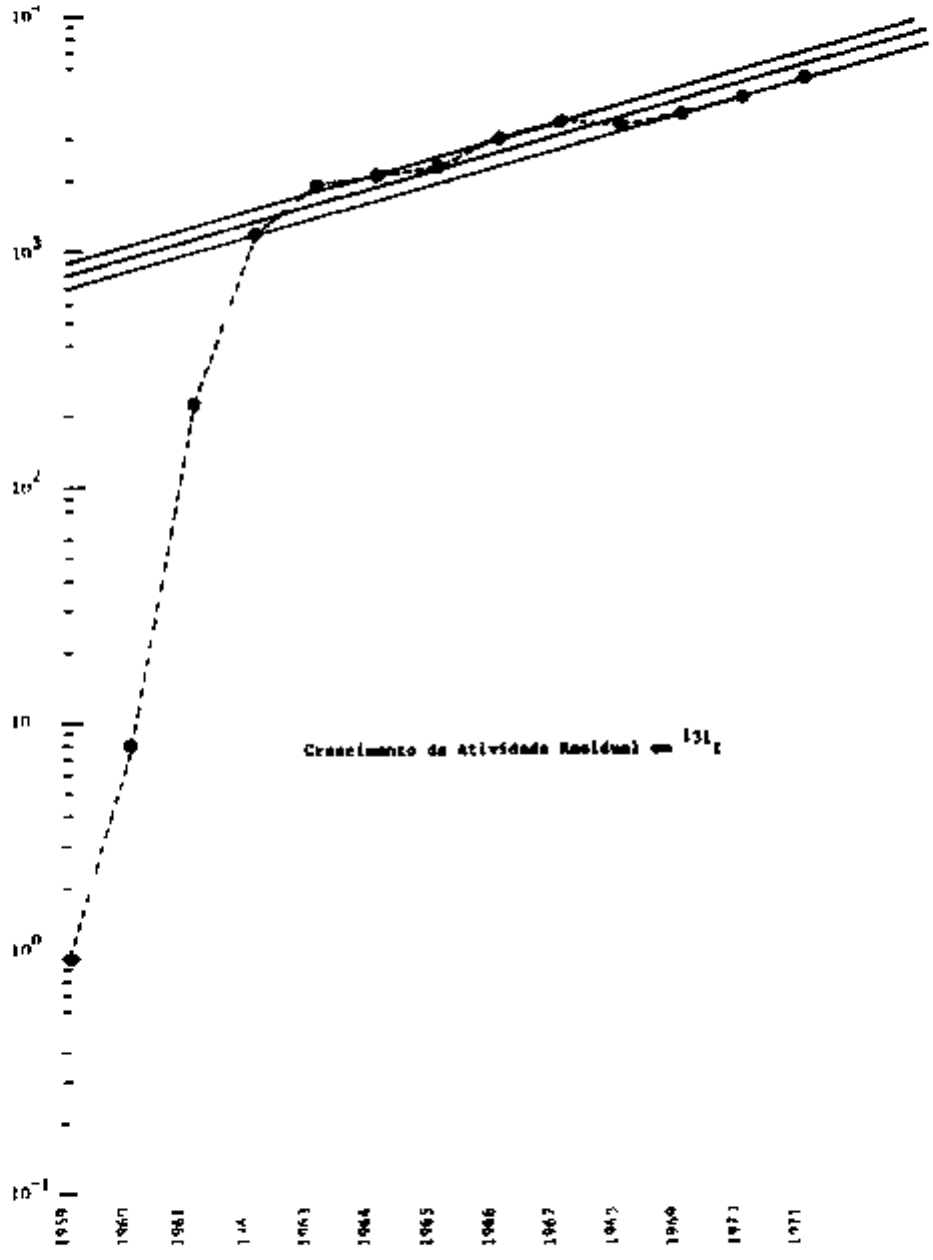


Figure 1 – Increase of the wasted activity in ^{131}I

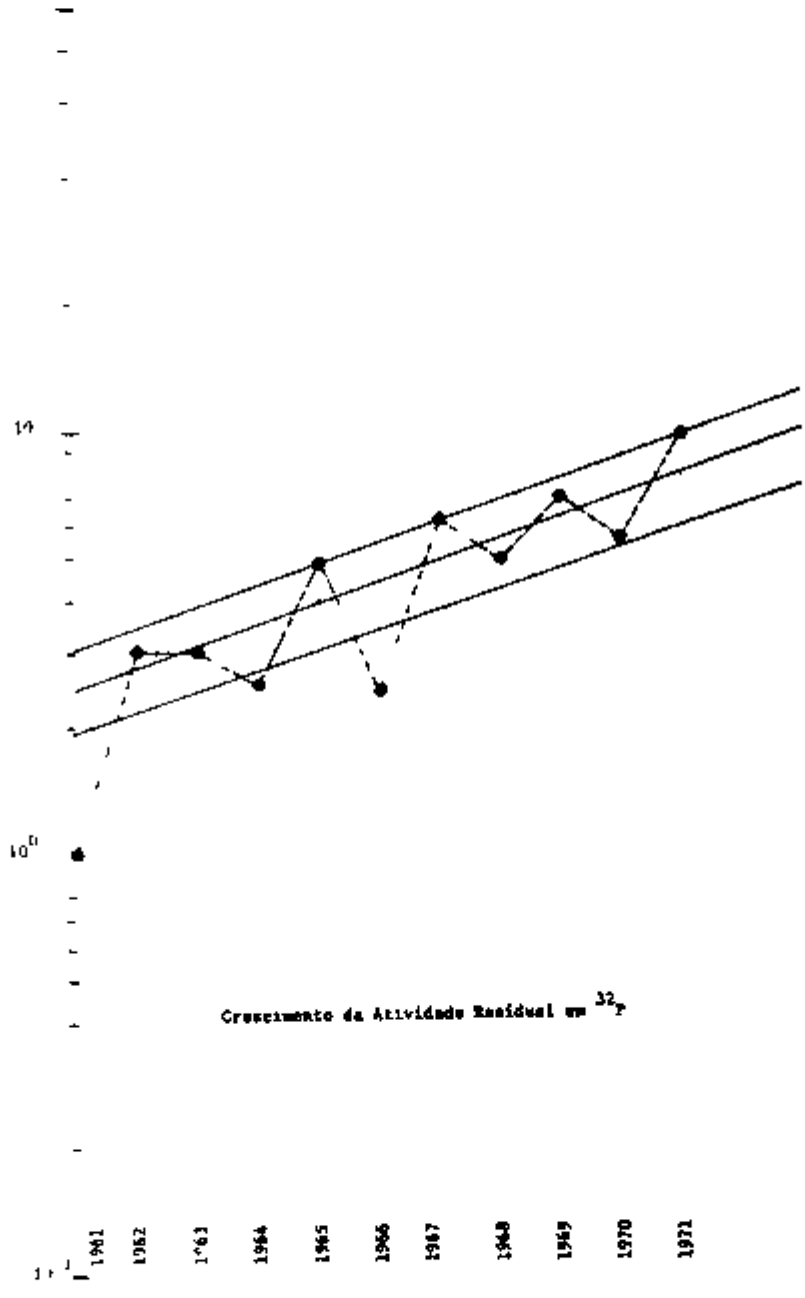


Figura 2 — Increase of the wasted activity in ^{32}P

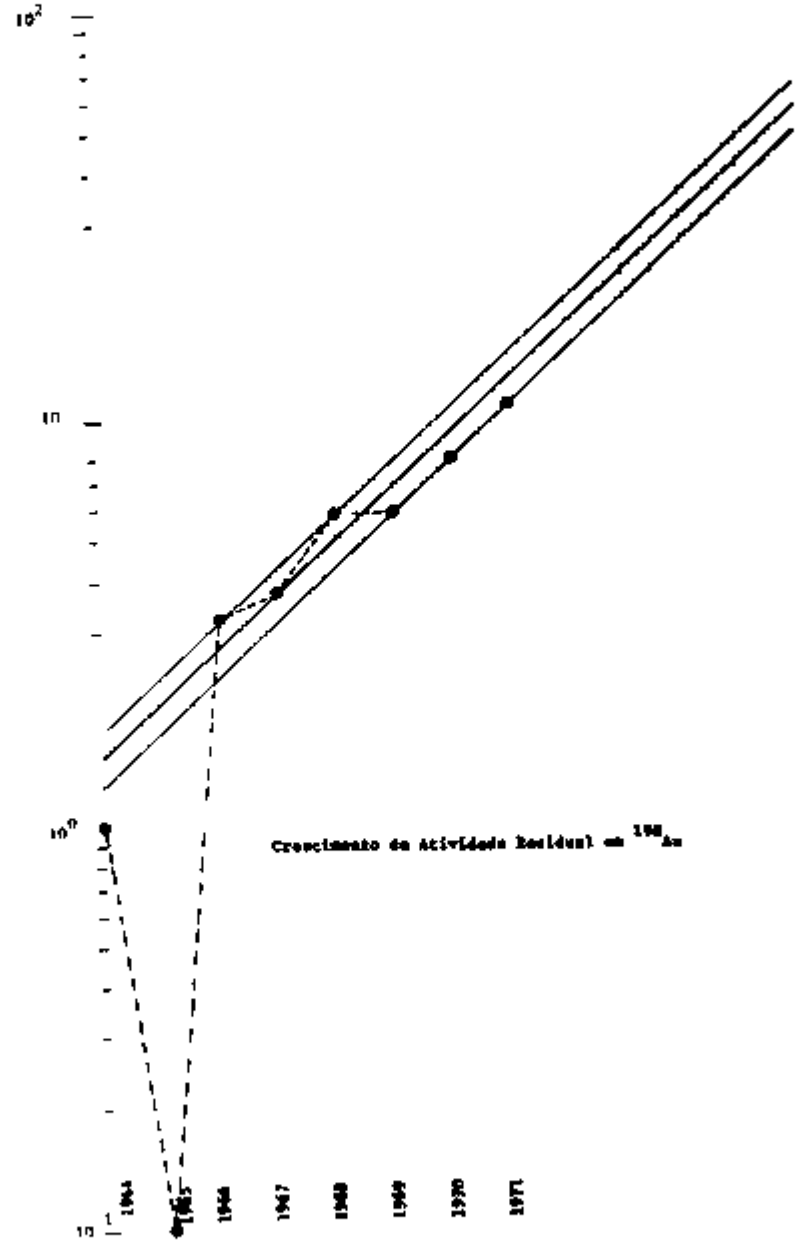


Figure 3 – Increase of the was ed activity in ^{198}Au

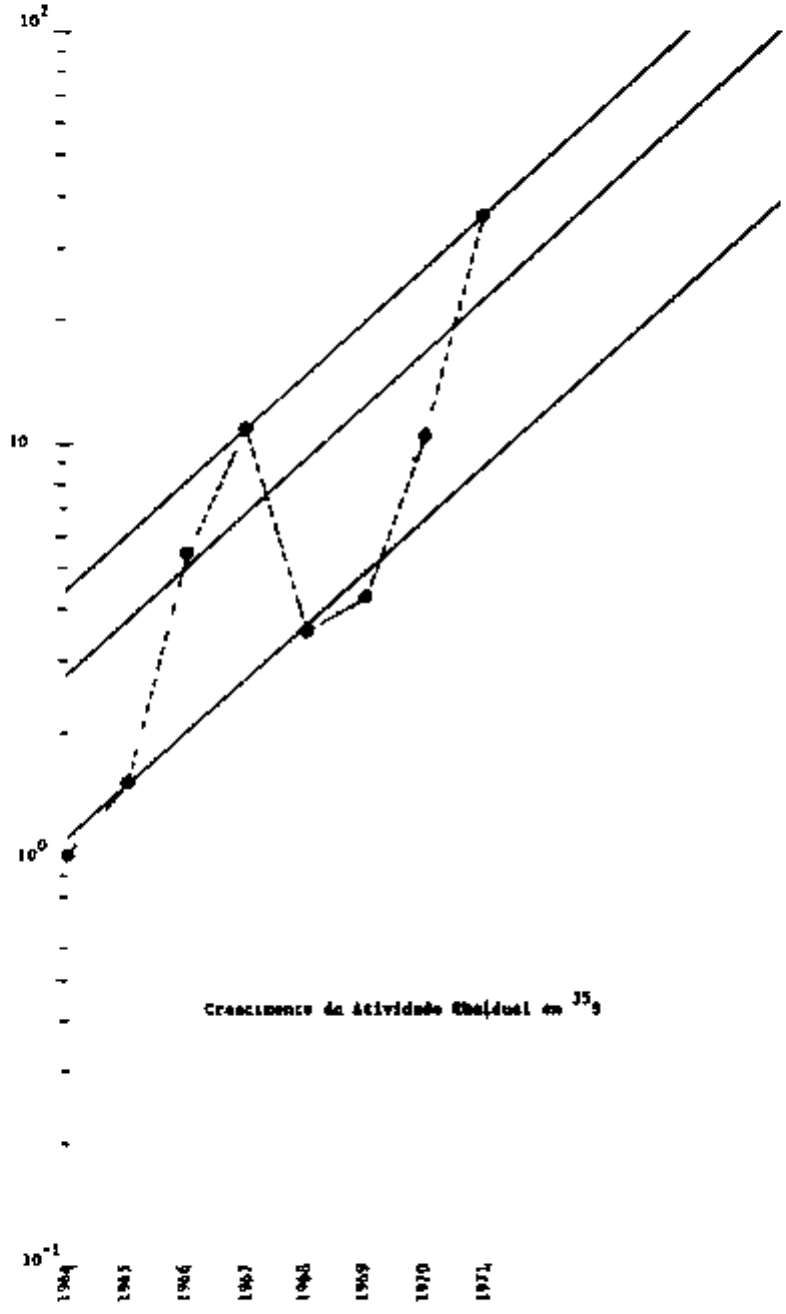


Figure 4 — Increase of the wasted activity in ^{35}S

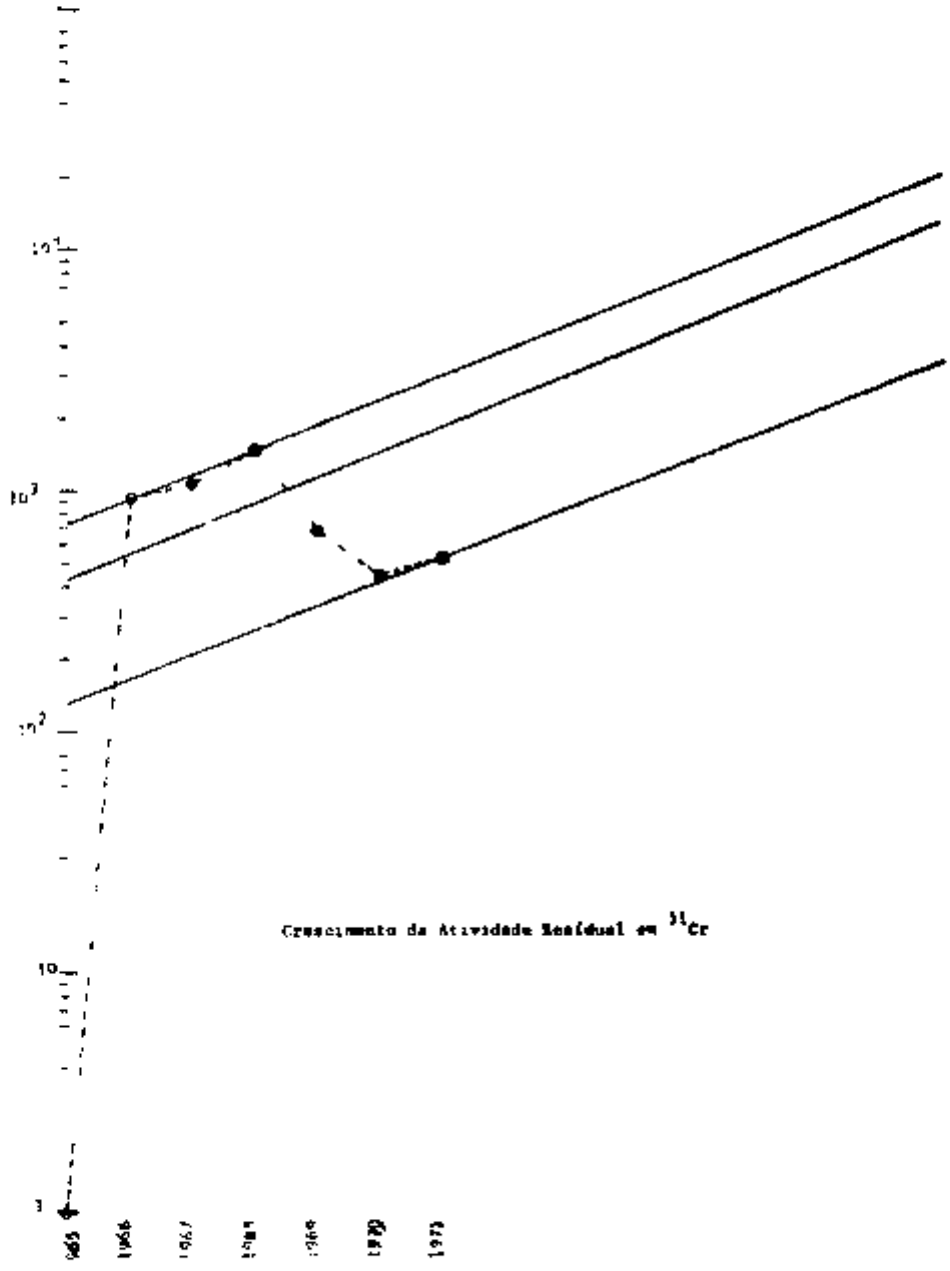
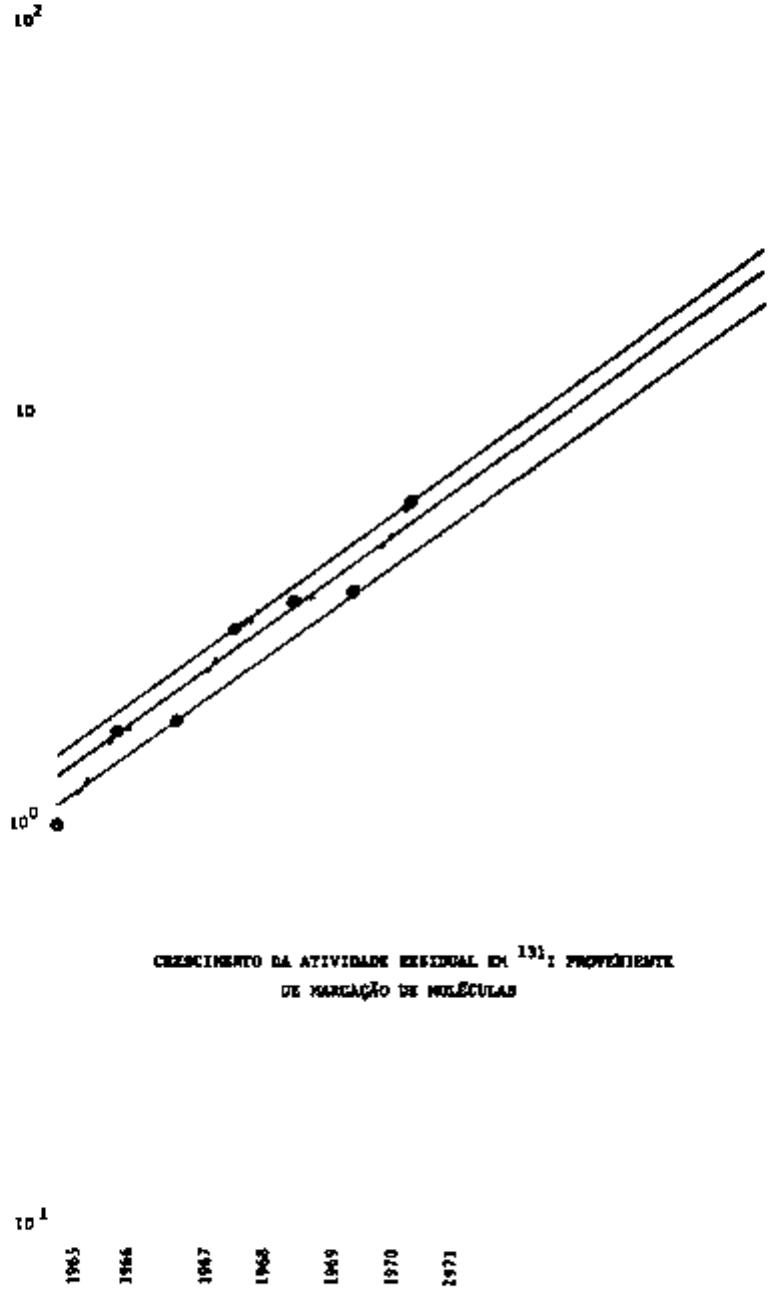


Figure 6 — Increase of the wasted activity in ^{51}Cr



CRESCIMENTO DA ATIVIDADE RESIDUAL EM ^{131}I PROVENIENTE
DE MARCAÇÃO DE MOLÉCULAS

Figure 6 — Increase of the wasted activity (in ^{131}I) coming from labeled compounds

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