The Fusion Fuel Cycle: Material Requirements and Potential Effluents

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October 1980

**Prepared for the
Environmental Control Technology Division,
U.S. Department of Energy
under Contract DE-AC06-76RLO 1830**

**Pacific Northwest Laboratory
Operated for the U.S. Department of Energy
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Under Contract EY-76-C-06-1830

Printed in the United States of America
Available from
National Technical Information Service
United States Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22151
Price: Printed Copy \$____*; Microfiche \$3.00

*Pages	NTIS Selling Price
001-025	\$4.00
026-050	\$4.50
051-075	\$5.25
076-100	\$6.00
101-125	\$6.50
126-150	\$7.25
151-175	\$8.00
176-200	\$9.00
201-225	\$9.25
226-250	\$9.50
251-275	\$10.75
276-300	\$11.00

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ABSTRACT

Environmental effluents that may be associated with the fusion fuel cycle are identified. Existing standards for controlling their release are summarized and anticipated regulatory changes are identified. The ability of existing and planned environmental control technology to limit effluent releases to acceptable levels is evaluated. Reference tokamak fusion system concepts are described and the principal materials required of the associated fuel cycle are analyzed. These materials include the fusion fuels deuterium and tritium; helium, which is used as a coolant for both the blanket and superconducting magnets; lithium and beryllium used in the blanket; and niobium used in the magnets. The chemical and physical processes used to prepare these materials are also described.

This work was supported by the Environmental Control Technology Division of the U.S. Department of Energy as part of the ongoing program that is evaluating the potential effluents and the present control capability and requirements of fusion energy.

SUMMARY

Environmental effluents are released as a result of obtaining and processing the materials needed in the generation of fusion power as they are released from any other power system. Early investigation of these effluents will reveal ways to control and also reduce the consequences of such effluent release. In this report the potential environmental effluents associated with the fusion fuel cycle are identified and the ability of current or planned technology to control these effluents is assessed.

Fusion power plant materials requirements are determined for two tokamak magnetic fusion reactor designs. Option 1 incorporates a liquid lithium-cooled blanket with no neutron multiplier, and option 2 utilizes a helium-cooled blanket with solid lithium aluminate breeding material and beryllium multiplier. The material requirements of these designs for initial inventory, as well as periodic replacement, are determined for the principal fusion reactor materials: niobium, lithium, beryllium, helium, deuterium and tritium.

For the materials of niobium, lithium, beryllium, and helium, the impact on the output from existing industries on a per GWe basis is considered quite small. The increment of any environmental effluents as the result of material requirement for initial fusion plants is therefore negligible.

The deuterium requirements are met by heavy water extraction and electrolytic decomposition. Fusion fuel requirements are approximately 100 kg per GWe-yr, or 500 kg of D_2O per GW-yr.

U.S. production of D_2O at Savannah River is 165 metric tons per year, and hence potential fusion demands are minimal in terms of plant output. Reactor designs using D_2O for cooling could substantially change this, introducing the need to supplement U.S. production with Canadian production of D_2O , currently in excess of 3000 MT/yr. The environmental effluent of greatest concern associated with the production of heavy water is H_2S released to water. During normal operation, the electrolysis will not release any harmful pollutants to either air or water.

The initial tritium inventory for fusion power plants will be obtained from tritium production facilities dedicated principally for strategic materials. The periodic replacement requirements will be obtained from the blanket breeding in lithium and associated tritium extraction systems of the fusion power plant. The only important effluent associated with tritium production is tritium itself, ^3H , a β -emitting radioactive isotope of hydrogen with a half-life of 12.3 years. An unknown but probably small fraction of normal tritium release would be attributed to the increment introduced to meet initial fusion power plant inventories. The Savannah River tritium production plant normally releases about 10^5 Ci/yr to the atmosphere and also 10^4 Ci/yr in water. No standards limit the amount of tritium discharged from facilities. Rather, the standards apply to the allowable dose to individuals, from which maximum allowable concentrations in the environment are determined.

Control of tritium effluents from a fusion power plant will require a multiple-barrier containment system to prevent the release of tritium gas or tritiated water. The overall goal is to reduce the tritium loss to the environment to less than 10 Ci/day. The final tritium barrier can be a containment structure shell enclosing the entire reactor hall. This together with the use of an emergency tritium containment system and a tritium effluent removal system should be adequate to prevent the escape of any abnormal tritium releases within the reactor building to the environment.

Helium production from natural gas or helium releases from the fusion power plant do not release any significant or harmful materials to the atmosphere or water table. Abnormal operating conditions also will not result in harmful pollutant releases to the biota.

The proven U.S. lithium reserve and recoverable resources are approximately 3×10^6 MT. This should be more than adequate to supply the initial inventory and replacement requirements for fusion power plants. The major effluents from lithium mining and metal production are airborne particulates of Li compounds or gaseous effluents; of the gaseous effluents, only H_2S and SO_2 are of environmental significance. The releases are well within the present EPA standards and the limits of current environmental control technology.

Reserves of beryllium are somewhat uncertain. Although there are several known deposits of beryllium crystals or pegmatitic ores, recovery is not economically attractive with current technology. In the mining and fabrication of beryllium, the only effluent release is beryllium dust. However, since beryllium is toxic, potential release of this material is carefully controlled to within EPA standards by current environmental control technology.

At present, there is no domestic production of niobium. In addition, the U.S. has no deposits of niobium that can be classified as reserves. According to the most recent data published by the Bureau of Mines, world-proved reserves are 10^7 MT of contained niobium. Of this amount, Brazil has 77%, the African countries 10%, and Canada 6%. The remaining 7% lies in Eastern Europe and the U.S.S.R.

In the milling and fabrication of niobium, the airborne effluent is primarily niobium dust from thermite reactors. Additional effluents of HF vapor and ammonia gases accompany the niobium-tantalum process for producing niobium metal. These effluents are readily controlled by current environmental control technology.

In summary, estimates of the effluent concentrations corresponding to annual fusion power plant requirements compared to the maximum ambient air concentration levels allowed under current regulations or recommendations are shown in Table S.1. When compared to the maximum ambient air concentration levels allowed under current regulations or recommendations, the present or planned environmental control technology appears to be adequate.

TABLE S.1. Anticipated Effluents of Fusion Fuel Cycle Compared to Present Standards

Annual Effluents Anticipated from Fusion Fuel Cycle		Present Environmental Standards, Annual Average	
Effluent	Air Concentration	Effluent	Air Concentration
LiAlSi ₂ O ₆ (dust)	2x10 ⁻⁵ µg/m ³	Li(LiH)	25µg/m ³ (a)
Li ₂ CO ₃ (dust)	2x10 ⁻⁵ µg/m ³		
Li ₂ NaPO ₄ (dust)	3x10 ⁻⁶ µg/m ³		
Nb (dust)	6.3x10 ⁻⁶ µg/m ³	Nb	50 µg/m ³
Be ore (dust)	4.1x10 ⁻⁶ µg/m ³	Be	0.01µg/m ³
Be(OH) ₂ (dust)	5.6x10 ⁻⁴ µg/m ³		
D	3.0x10 ⁻⁶ µg/m ³	D	No standard
Particulates	0.18 µg/m ³	Particulates	75µg/m ³ (d)
SO ₂	1x10 ⁻⁵ (c) µg/m ³	SO ₂	80µ/m ³
	1.3x10 ⁻² (d) µg/m ³		
CO ₂	0.13 µg/m ³	CO ₂	9x10 ⁶ µg/m ³
NO _x	4.4x10 ⁻³ µg/m ³	NO _x	1x10 ² µg/m ³ (b)
HF	1.2x10 ⁻³ µg/m ³	HF	3.007x10 ⁴ µg/m ³
NH ₃	2.6x10 ⁻² µg/m ³	NH ₃	1.8x10 ⁴ µg/m ³
H ₂ S	10 ⁻⁵ µg/m ³	H ₂ S	1.5x10 ⁴ µg/m ³
T gaseous	10 ¹⁰ µCi/cm ³	T	4x10 ⁻⁵ µCi/mL
liquid	10 ⁻⁶ µCi/cm ³		

(a) 8-hr time-weighted average; occupational number only.

(b) Standards for particulates, SO_x and NO_x vary according to state or region.

(c) From lithium production.

(d) From heavy water production.

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THE FUSION FUEL CYCLE: MATERIAL REQUIREMENTS AND POTENTIAL EFFLUENTS

1.0 INTRODUCTION

The development and commercialization of nuclear fusion for electric power production is a major goal of the U.S. Department of Energy and of several other nations. Fusion power is a potential long-term solution to current problems of world energy supply. It may not only be safer and environmentally more attractive than other energy sources, but its fuel resource requirements are freely available to all nations in virtually unlimited supply.

Effluents are released to the environment as a consequence of the generation of fusion power as they are from any other power system. Early investigation of these effluents will reveal ways of controlling and reducing the consequences of such effluent release. This will ensure that environmental considerations will not limit the development of fusion energy. Knowledge of potential environmental impacts will influence decisions affecting the direction of the fusion development and technology program.

The Pacific Northwest Laboratory (PNL) is conducting a research program to identify the environmental control technology (ECT) requirements associated with all aspects of the materials and resource cycle of magnetic fusion power plants that use deuterium and tritium (DT) fuel. The complete fusion fuel cycle--from raw materials to ultimate waste disposal--is being investigated. Previous studies at PNL considered the procurement⁽¹⁾ and availability⁽²⁾ of materials used in current fusion power plant design concepts⁽³⁾ as well as the disposition of nontritium radioactive wastes generated from such power plants.⁽⁴⁾ A previous report⁽⁵⁾ of the present study summarized an investigation of material requirements and disposition of spent reactor equipment for tokamak fusion power plants fueled with deuterium and tritium. In this report, the potential environmental effluents that may be associated with the acquisition and initial processing of materials for use in the fusion fuel cycle are identified and the ability of current or planned technology to control these effluents is assessed.

A wide variety of power plant concepts are possible at this stage of magnetic fusion development. However, a number of generic system requirements make it possible to identify certain specific material requirements. A schematic of a conceptual magnetic fusion power plant is shown in Figure 1.1. The plasma fuel undergoes the $t(d,n)\alpha$ reaction when heated to extremely high temperatures ($\sim 100 \times 10^6$ K). At these temperatures the plasma is contained in a vacuum chamber where it is confined away from the walls by superconducting magnets to prevent thermal condition and loss of energy. The high energy (14 MeV) neutrons produced in the plasma are absorbed in the surrounding blanket material. Thermal energy is then removed from the blanket by the primary coolant, which transfers the heat likely to steam in a conventional steam turbine generator cycle. The blanket is surrounded by a radiation shield to prevent damage to the magnets from neutron and radiation heating. A biological shield is also present to protect personnel from neutron and photon damage.

The plasma is composed of deuterium and tritium. Whereas the deuterium portion of the fuel is a natural isotope of hydrogen that can be economically extracted from water, the tritium fuel is an unstable isotope of hydrogen that exists naturally in rare amounts. Tritium will be produced in the power plant in self-sustaining quantities by incorporating lithium into the blanket. Neutrons interact with lithium in an (n,α) reaction to produce tritium and helium. The tritium is extracted from the blanket and combined with unconsumed deuterium and tritium separated from the plasma exhaust for further processing and purification in the reactor fueling system. The proper mixture is then re-injected into the plasma.

The fuel cycle of a magnetic fusion reactor comprises the flow of materials used for fabrication, whose inventory is depleted through nuclear fusion or transmutation, or that must be replaced due to radiation damage during the normal operation of the reactor. In addition to the fusion reactor, the facilities of the fusion fuel cycle are: mining, milling, fabrication, waste storage and disposal facilities, and transportation systems needed to convey the materials between these major elements. The fuel cycle is shown schematically in Figure 1.2 for a DT-fueled fusion power plant.

FIGURE 1.1. Schematic of a Magnetic Fusion Power Plant

The 'back end' of this material cycle was the subject of a previous ECT analysis,⁽⁵⁾ of interest in this study are the 'front end' operations of the fuel cycle, as shown in Figure 1.2. Structural materials are mined, refined, milled and fabricated. Lithium is mined and refined, helium is extracted from natural gas wells, and deuterium is produced by heavy water extraction and electrolytic decomposition. The inventory of tritium is maintained by breeding in the lithium blanket with subsequent extraction and recycle to the plasma. The materials analyzed in this study and their functions in the fuel cycle are:

- niobium used for constructing blankets and superconducting magnets
- beryllium used as a neutron multiplier in some blanket designs
- lithium used in its liquid form as a blanket breeding material and coolant; alternative designs use a solid lithium compound blanket cooled by helium
- helium used as a superconducting magnet coolant as well as a blanket coolant
- deuterium used with tritium to fuel the fusion reaction.

The chemical and physical processes of the facilities used to prepare these materials are described and the environmental effluents are identified in the following report. For each effluent, existing standards controlling its release are summarized and anticipated regulatory changes are identified. The ability of existing and planned environmental control technology to limit effluent releases to acceptable levels is evaluated.

FIGURE 1.2. Fusion Fuel Cycle

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2.0 MATERIAL REQUIREMENTS FOR REFERENCE FUSION SYSTEM DESIGNS

In this chapter, the use of niobium, beryllium, lithium, helium and the hydrogen isotopes of deuterium and tritium in magnetic fusion power plants will be described. Reference fusion systems will be developed to identify basic material requirements. Operating and maintenance requirements will also be given such that normal consumption or replacement of materials can be identified.

Magnetic confinement systems will be presented first to identify the use of niobium in superconducting coils. Material requirements in several designs will be examined and a representative references design will be presented.

The next section on blanket system designs will discuss the use of beryllium and lithium in fusion reactors. Because of the various trade-offs between these materials, two reference designs must be considered. Option 1 will use a liquid lithium blanket and liquid lithium coolant. Option 2 will use a solid lithium compound for tritium production with beryllium for neutron multiplication and helium gas coolant.

Finally, the deuterium-tritium fuel systems required in a fusion reactor will be examined to identify material requirements. The fuel system is made up of two major systems; the plasma recycle and blanket tritium recovery system. These will be examined in sufficient detail to present representative designs and fuel inventories.

Using these reference designs, material inventories required for start-up and operation will be summarized at the end of this section. Based on these fusion requirements, estimated material production and associated effluents will be developed in the following chapter.

2.1 MAGNETIC CONFINEMENT SYSTEMS

The conditions necessary to achieve useful energy production by fusion require that the hydrogen fuels be physically isolated to reduce the loss of heat to surrounding materials. It also reduces the influx of impurities to the fusion fuel. In magnetic fusion power plants, this confinement will be

achieved by using a specially designed geometry of magnetic fields to trap the ionized hydrogen isotopes.

These magnetic fields are produced by high ampere currents flowing in coils placed near the vacuum chamber which contains the hydrogen fuel. If normal conducting materials such as copper or aluminum are used exclusively in a full scale fusion power plant, resistance heating in the coils can result in an unacceptably high energy drain. This can significantly lower the net electric energy output from the plant. To eliminate this loss, superconducting materials are almost always specified in power plant design studies.

Only a small number of the many materials which demonstrate superconductivity are being seriously considered for fusion applications. The materials which have received the most interest are the niobium alloys with titanium (NbTi) and with tin (Nb₃Sn). Of the two, Nb₃Sn will allow higher fields with a greater safety margin for the critical temperature. In addition, these higher fields may be necessary to achieve confinement in devices of practical size. But, Nb₃Sn is comparatively brittle and will require extensive development before it can be used with confidence. The Nb₃Sn conductor is very promising and experience is now being gained in the Large Coil Project sponsored by the DOE Office of Fusion Energy, where one of six coils will use Nb₃Sn.⁽¹⁾

For conceptual commercial power plants, three designs have specified Nb₃Sn, however the vast majority of designs still specify NbTi.⁽¹⁷⁾ The material can be handled easily with today's wire and filament drawing techniques, and ease of construction is considered to be a major advantage for first generation commercial plants. Also, only four plant designs to date provide a detailed inventory of materials used in coil design, and all used NbTi. These are the UWMAK I, II, III and NUWMAK Plants.^(2, 3, 4, 5) Because of the preponderance of information on NbTi and its continued use in commercial reactor designs, NbTi in a 50-50 alloy by weight will be used in this ECT analysis of niobium consumption in magnetic fusion reactors.

To identify niobium use in coil construction, the tokamak reactor designs mentioned above will be examined. The three main types of coils used in the tokamak concept are the toroidal (or 'D'), vertical field, and ohmic heating

transformer coils. As an example of their geometry and location, a cross-section of the UWMAK-II design is shown in Figure 2.1.

Ideally, the use of conducting material would be directly related to a number of parameters for coil design and the physical size of the reactor. At this stage of fusion development the latitude in design of the coils and the exact configuration used makes it extremely difficult to apply any parametric relationships. However, when the total material usage for 'D', ohmic heating and vertical field coils is considered on a per MWe basis, the variation in the total inventory of NbTi is not that great between designs. The total inventories for the four designs are given below in Table 2.1. A simple average puts material requirements at 90 MT of NbTi per 1000 MWe or 45 MT of niobium per 1000 MWe.

The 45 MT per 1000 MWe will be used here for niobium requirements in fusion plants. This analysis will assume that the coils last the lifetime of the plant and require no annual replacement.

For completeness, a full reference design will be presented here to identify manufacturing requirements. The UWMAK-II design was used in a previous ECT analysis of fusion and is representative of large tokamak fusion power plants.

TABLE 2.1 UW Tokamak Coil Variables

<u>Design</u>	<u>MWe</u>	<u>NbTi Inventories (MT)</u>			<u>MT/ 1000 MWe</u>
		<u>D Coil</u>	<u>O+VF Coil</u>	<u>Total</u>	
UWMAK I	1500	104	18.14	122.14	81
UWMAK II	1716	164	25.94	129.94	111
UWMAK III	1985	76	65.00	141.00	71
UWMAK	660	58.2	3.90	62.10	94

The principle dimensions of this plant are:

Major Radius	13 m
Minor Radius	5 m
Aspect Ratio	2.6 m

FIGURE 2.1. UWMAK-II Torus Cross Section⁽³⁾

Duty Factor	0.94
Burn Time	5000 sec
Plasma Volume	6415 m ³
Neutron Wall Loading	1.16 MW/m ²
Thermal Power	5000 MWt
Electric Power	1700 MWt

All magnetic fields used in the UWMAK-II design are generated by currents flowing through the NbTi superconductor cooled with liquid helium. The main toroidal field is generated by 24 D-shaped coils. The placement of vertical field and ohmic heating coils is again shown in Figure 2.1

For the toroidal field coils, the current carrying superconductor is actually embedded with insulation in grooves machined or forged into large stainless steel D-shaped disks (like grooves on a record). Windings on the front and back are connected by a cross-over and form one continuous coil. In UWMAK-II, each toroidal field coil is a make up of 19 such disks sandwiched together as shown in Figure 2.2. Each disk represents a complete circuit with its own current leads that can be cut in case of insulation failure. The 19 completed disks are insulated by layers of Micarta, and bolted together with aluminum bolts. This entire structure is placed in a dewar containing liquid helium for operation at a temperature of 4 to 5 K.

The actual current conductor is shown in Figure 2.2 (shaded portion) bonded to a copper stabilizer. The conductor consists of NbTi filaments in a copper matrix with a 1:1 volume ratio. This matrix is constructed by pressing NbTi rods into holes pre-drilled in a copper ingot. Extrusion to a smaller diameter bonds the NbTi to the copper. Wire drawing techniques are then used to reduce the matrix to a 1.2 cm diameter cross section. During this process the NbTi rods are reduced to filaments approximately 0.038 cm in diameter. A twist is also applied during this process. The matrix is then rolled flat to a 0.5 cm thick ribbon and bonded to a copper ribbon of varying thickness as shown in Figure 2.2. When finished, the conductor strip must be 3810 m (2.36 miles) long to form a continuous winding on each stainless steel disk (19 of which form one field coil).

FIGURE 2.2. Toroidal Field Coil Construction

The divertor and transformer coils in the UWMAK-II design use a similar NbTi superconductor, with the addition of a cupro-nickel sleeve added to the NbTi bars before extrusion. The sleeve is added to reduce alternating current losses caused by pulsing the coil during normal operation of the tokamak. Most of the transformer coils, which provide ohmic heating, are wound in the region of the central support stanchion and require no stainless steel for structural support. However, the divertor coils (which generate the vertical field) will require a steel backing. The design parameters and material inventories for the toroidal, vertical field (divertor and ohmic heating coils are given in Tables 2.2, 2.3, and 2.4, respectively.

2.2 BLANKET SYSTEM DESIGN

The blanket region in a fusion reactor is designed to convert the kinetic energy of fusion neutrons to heat and to insure an adequate tritium breeding rate. It is also desirable to keep outer shield requirements to a minimum while protecting the superconducting magnets from neutron damage. A number of variables then came into play in the use of lithium and beryllium in blanket design, as will be discussed here.

All first generation fusion power plants will use deuterium-tritium fuel due to its favorable reaction cross-section. Deuterium is naturally occurring and can be extracted from water. However, tritium is very rare in nature and will likely be produced in the reactor blanket to sustain a fuel supply. Lithium or lithium compounds are used for this purpose, capturing fusion neutrons as shown in the reactions below:

Both isotopes of lithium have exothermic and endothermic reactions, but the two shown are dominant.

Liquid Li metal can maintain adequate tritium breeding with little or no ^6Li enrichment and also can serve as a coolant. However, the liquid Li is

TABLE 2.2. Specifications of the Toroidal Field Magnets

Characteristic	Dimension
1. Field Specifications	
On-axis field at a radius of 13 m, B_0	3.67 T
Field at inside turn of far edge	1.91 T
Maximum field at the conductor, B_m	8.30 T
Maximum field in the self-supported portion	4.48 T
Field ripple along the far edge of plasma	0.05%
Total magnetic energy stored	62 MWh
2. Design Stress	
42K	
Design stress in 316 stainless steel at	4.14×10^8 N/m ² (60,000 psi)
Design Strain	0.002
3. Magnet Components	
Number of Magnets	24
Number of discs per magnet	19
Number of conductor turns per disc	58 (29 each side)
Conductor current	9,065 A ϕ 19 discs 10,131 A ϕ 17 discs 11,482 A ϕ 15 discs
Disc cross section	5 x 9 cm
Superconductor composite: NbTi in copper matrix	500 filaments
(thickness of composite strip = 0.038 cm, filament twisted every 30 cm)	
Copper stabilizer cross section	2.87 cm wide x 190 cm thick on inside turn tapering to 1.70 x 1.13 cm on outermost turn
Insulation between conductor and stainless steel disc	0.10 cm reinforced epoxy
Average copper to superconductor ratio (volumetric)	8:1
Spacer between discs	0.64 cm thick micarta, covering 50% of the area
Aluminum alloy bolts to fasten the discs together against slipping, 1-in. diameter	2630
Lateral support between the magnet and dewar: reinforced epoxy struts on each side of cross section	5500 cm ²
Each magnet weight	710 MT(a)
4. Total Materials Required	
NbTi alloy superconductor	146 MT
Copper stabilizer	8000 MT
Stainless steel in discs	7230 MT
Stainless steel in dewar	941 MT (removable portion only)
Reinforced epoxy in discs	395 MT
Micarta spacers between discs	255 MT
Reinforced epoxy struts in dwars	16 MT
Aluminum alloy bolts to assemble discs	95 MT
Cryogenic insulation	850,000 m ²
Liquid helium inside dwars	160,000 liters
Liquid helium in storage	300,000 liters

(a) MT = metric ton = 10^3 kg.

TABLE 2.3. Physical Parameters of Diverter Coils

Coil	D ₁	D ₂	D ₃	D ₄	D ₅	D ₆
R(m)	8.2	8.8	11.5	11.5	14.6	19.0
Z(m)	6.5	10.0	12.3	9.5	8.0	5.0
Maximum current (MA)	-3.70(a)	4.87	-1.46	3.40	-1.57	-6.82
Maximum B (T)	2.56	2.73	1.69	2.40	1.75	3.07
Maximum tension (10 N)	-6.74	24.5	1.56	-7.46	10.1	42.8
Conductor height, (cm)	10.8	10.8	10	10	9.6	9.6
Number of filaments per conductor	70	75	50	65	55	80
Number of turns (4 conductor/turn)	50(b)	63	20	45	20	84
Number of layers	5	3	4	5	4	7
Magnet minor radius (cm)	32	34x136(c)	22	31	22	44
NbTi (MT)	0.53	0.75	0.21	0.63	0.30	2.40
Copper (MT)	100	140	50	115	60	320
Stainless steel (MT) for reinforcement	0	12	0	0	10.3	57.5
Micarta spacer (MT)	2.25	4.05	1.88	3.17	2.38	10.2
Stainless steel in dewar	3.3	14.9	3.4	4.6	3.9	12.3

(a) Negative value means the coil is under compression.

(b) The number of turns from the top to bottom are 6,12,14,12,6: other coils can be similarly arranged.

(c) Because of the field specification, this should be flat.

TABLE 2.4. Physical Parameters of Transformer Coils

Coil	T ₁	T ₂	T ₃	T ₄	T ₅	T ₆	T ₇	T ₈	T ₉
R(m)	3.61	3.72	3.94	4.25	4.66	5.43	6.50	7.25	8.22
Z(m)	1.00	3.06	4.99	6.96	8.92	13.50	15.05	17.0	19.3
I _{max} (MA)	-4.13	-3.98	-3.61	-3.41	-2.85	-3.36	-3.34	-2.51	-2.20
B _{max} (tesla)	5.72	5.68	5.40	5.10	4.60	5.14	5.60	4.07	3.93
T _{max} (10 ⁶ N)	23.9	22.9	20.2	16.8	8.35	17.4	9.51	6.56	3.57
Conductor height, (cm)	10.3	10.3	10.3	10	10	10	10.3	9.5	9.5
No. of filaments per conductor	250	240	220	180	120	180	240	110	100
No. of turns (4 conductors/turn)	63	63	56	56	48	60	57	39	36
No. of layers	9	9	7	7	12	5	3	3	4
Magnet size (cm)	100x32	100x32	80x36	78x34	132x20	5(12x52)	36x100	34x65	45x46
NbTi (MT)	1.05	1.04	0.90	0.79	0.50	1.08	1.64	0.60	0.55
Copper (MT)	52.0	57.5	50.5	51.2	48.2	70.0	84.7	55.8	58.4
Micarta Spacer (MT)	1.33	1.37	1.24	1.28	0.69	1.72	2.00	1.44	1.55

relatively transparent to the fusion-produced neutrons, and energy deposition (through collisions and capture) is low in a reasonably thick region (20 to 50 cm). The outer shield portion of the blanket, which should operate at relatively low temperatures, must be designed to reflect neutrons back into the blanket region, and thus the size, complexity and cost of the shield are increased.

Solid lithium compounds are more effective than liquid lithium in absorbing the neutron energy in a thin blanket, but they sometimes suffer from low tritium breeding or problems with extraction by diffusion and permeation. These compounds are generally enriched in ^6Li (up to $\sim 90\%$) to take advantage of more favorable cross sections in the attenuated neutron spectrum. A layer of beryllium in the blanket, through its favorable $(n, 2n)$ reaction, can be used to increase tritium breeding.

Exothermic neutron reactions in beryllium ($n, 2n$; n, t ; and n, α) also act as an energy multiplier in the blanket. Neutronics calculations for UWMAK-I⁽³⁾ showed a net energy increase of 9.3% by adding a 4 cm layer of beryllium in the blanket; an 8 cm layer caused a net energy increase of 18.45%. Quantities of beryllium beyond amounts required for adequate tritium breeding may then be used for energy multiplication. Some lithium compounds that typically use neutron multiplication in fusion blankets are LiAl , LiAlH_4 , LiAlO_2 , Li_2SiO_3 , Li_4SiO_4 , LiOH and LiH . Compounds with neutron multipliers such as lead or beryllium do not require additional Be as metal to be present. Examples are LiBeF (flibe), Li_4Pb_2 and Li_3Bi .

The use of lithium and beryllium in several design studies for conceptual demonstration or commercial fusion power plants is given in Table 2.5. As was the case with niobium use in coils, it is difficult to determine if any exact relationship exists between lithium and beryllium inventories per 1000 MWt and the tritium breeding ratio. Conceptually, there should be some trade-off between Li and Be to maintain a tritium breeding ratio; however, additional Be may be used for energy multiplication as mentioned above. From Table 2.5, the following observations can be made:

TABLE 2.5. Lithium and Beryllium Use in Fusion Design

Ref	Design	Power, MWt	Coolant	Fertile Material	Neutron Multiplier	Li MT	Be MT	Breeding Ratio
2	UWMAK-I	5,000	Liquid Li (metal)	Li metal		1,700		1.48
4	ORNL (Fraas)	1,000	"	"		460		1.34
5	Theta Pinch	12,000	"	"	Be	860	50	1.11
6	Mirror	640	"	(4 ⁶ Li)		83		1.1
7	Blascon	4,800	"	Li metal		1,600		1.3
8	Wet Wall	3,745	"	"		"		1.2
9	Suppressed Albatron	270	"	"		"		1.49
10	LASL Laser	3,333	"	"		1,100		"
11	ORNL (Wells)	1,700	"	"		289		"
3	UWMAK-III	5,000	He	"		465		1.27
12	BNL	5,000	He (gas)	LiAl, LiAlO ₂ (90 ⁶ Li)	Be	60	189	1.2
1	UWMAK-II	5,000	"	LiAlO ₂ (90 ⁶ Li)	Be	50	400	1.18
13	PPPL	5,303	"	Flibe	Be	598	875	1.04
14	GA-DEMO	1,676	"	Li ₇ Pb ₂ Li ₄ SiO ₄	Pb	65		1.2
15	ORNL (Steiner)	2,150	HITEC (salt)	Li metal		95		1
18	NUWMAK	2,097	Water	Li ₆₂ Pb ₃₈ eutectic	Pb	40		1.54

(a) UWMAK-II uses an additional 300 MT in the impurity system in the divertor.

- Beryllium is used almost exclusively in designs using solid Li compounds.
- Enrichment in ⁶Li is used exclusively in designs using solid Li compounds.
- Liquid Li metal-cooled designs in general do not require lithium enrichment or beryllium as a neutron multiplier.

Based on these observations, two generic designs will be required to estimate likely lithium and beryllium inventories in fusion plants. One design will use a liquid lithium blanket and coolant with no beryllium. The other design will use a solid lithium compound blanket with helium cooling and include a beryllium neutron multiplier to obtain an adequate tritium breeding ratio. An example blanket configuration was illustrated in Figure 2.1.

Because of the wide variation in blanket configurations for liquid lithium, a reference design will not be described. The UWMAK-II design is representative of solid Li blankets and will be described with Option 2.

2.2.1 Blanket Option 1 - A Liquid Lithium Blanket Cooled with Liquid Lithium

The designs using liquid lithium as a coolant and fertile material as given in Table 2.5 were used to predict the required inventory of lithium as a function of reactor output. The theta pinch design,⁽⁶⁾ which uses beryllium, was not considered. A least squares fit was used to predict the lithium required:

$$y = 1.8 \times 10^2 e^{4.6 \times 10^{-4}x}$$

where

y = lithium inventory in metric tons

x = plant output in MWt

Assuming a 33% thermal efficiency, the required inventory of liquid lithium metal would be about 730 metric tons per 1000 MWe. A chart of lithium requirements by reactor capacity is shown in Figure 2.3.

Lithium purity is an important consideration in defining corrosion of materials exposed to a liquid lithium coolant. It was assumed for this analysis that the lithium is delivered to the plant site as a 99.5% pure commercial grade. Typical impurities in commercial lithium are given in Table 2.6. Purity requirements for reactor operation are also shown.

To achieve the necessary purity for use in the power plant as shown in Table 2.6, additional processing steps are required. It is assumed that these take place at the plant site with no impact on resource acquisition.

Assuming a 33% plant efficiency, 1.2 blanket multiplication and a tritium breeding ratio of 1.3, lithium burnup is ~400 kg/1000 MWe-yr for natural lithium.

Additional lithium will be lost in routine coolant cleanup operations and during blanket maintenance. It is assumed that these losses are comparable to

FIGURE 2.3. Li Inventories in Liquid Power, MWth

the lithium losses caused in tritium breeding. Thus, the assumed annual replacement requirements for lithium are about 0.80 metric tons per year per 1,000 MWe of plant capacity.

2.2.2 Blanket Option 2 - Solid Lithium Compound Blanket Cooled by Helium Gas

As indicated by the designs described in Table 2.5, beryllium would most likely be used only with a solid lithium compound. For estimating lithium and beryllium requirements, the following design was used:

- helium gas-cooled tokamak
- LiAlO_2 solid breeding compound

TABLE 2.6. Purity of Commercial Lithium, ppm by Weight

<u>Major Impurities</u>	<u>Purity of 99.5% Lithium^(a)</u>	<u>Lithium Purity Required for a Reactor^(b)</u>
Na	150	40
K	70	75
Total Alkali Metals	220	115
Alkaline Earths	300	10
Other Metals	150	70
Chlorine	70	50
Nitrogen	100	40

(a) Typical analyses.

(b) Expressed impurity levels are maximum values.

- enriched Li to 90% ^6Li
- beryllium neutron multiplier.

These characteristics are those of the UWMAK-II design,⁽¹⁾ again which has been partially analyzed by the ECT program.⁽¹⁸⁾ The UWMAK-II design is larger than the tokamak designs recently proposed, such as the ORNL-DEMO.⁽¹⁷⁾ However, it is one of the most complete design studies to date where material requirements are specified. The material requirements are normalized here to a 1,000 MWe basis.

A cross section of the UWMAK-II torus was shown in Figure 2.1. The primary structural material is stainless steel with a carbon (graphite) curtain. Helium serves as the primary coolant. The cross-section view in Figure 2.1 shows the helium coolant supply lines, particle collection plates for plasma impurity control, the graphite curtain, blanket, shield and toroidal field coils. The extended D toroidal field coils reach out beyond the shield, allowing the shield to be opened. This allows access to the blanket for servicing or removal.

The UWMAK-II torus is divided into 24 sections, one for each toroidal field coil. The blanket is composed of three parts: the removable wall cells that contain the breeding material, the vertical structure joining the front

wall cells to the helium headers, and the helium gas supply and return headers. The total thickness of the blanket section from the helium supply headers to the front wall is 89 cm.

Figure 2.4 shows a blanket section. There are five cells in each blanket section. There are about 2,840 sections in the reactor. A cell is composed of four material regions: LiAlO_2 breeder material, Be neutron multiplier, more LiAlO_2 and a graphite reflector. The removable portion of the blanket is composed of these semi-cylindrical cells, which are 23.1 cm wide, 27 cm high, and 0.6 to 1.5 m long. The first breeder zone in the cell is made up of 3 cm of LiAlO_2 (90% ^6Li enriched). This is followed by the 18 cm-thick beryllium metal neutron multiplication region. The outer breeder zone is made up of 10 to 13 cm of LiAlO_2 . Tritium collection manifolds are located near the outside of the outer breeding zone. The breeding material and neutron multiplier are placed in stainless steel pins 3.5 cm in diameter, approximately

FIGURE 2.4. UWMAK-II Blanket Section⁽¹⁾

30 cm long, with 0.75 mm wall thickness. These pins are closed at the plasma end and connected to a vacuum pumping system at the outside end. The bred tritium is collected and stored for future use.

The front wall replacement strategy in UWMAK-II is designed to keep downtime to a minimum. The replacement schedule is half the blanket every year, with an estimate of 23 days downtime.

New front wall sections are loaded with fresh breeding material (LiAlO_2 and Be), assembled on the vertical structure containing the graphite reflector, and welded together. After the graphite curtain is attached the refurbished modules are reinstalled in the reactor blanket and welded/sealed to the adjoining sections. The modules weigh 330 MT each, while the largest piece weighs 175 MT. The UWMAK-II blanket material inventories that are to be replaced every 2 years are shown in Table 2.7.

With only 7.5% ^6Li in natural lithium, 320 MT/1,000 MWe of natural lithium will be required to supply the initial inventory of 24 MT of ^6Li /1000 MWe. Tails from this enrichment process could meet divertor lithium requirements;

TABLE 2.7. UWMAK-II Material Inventories

	<u>Material</u>	<u>Inventory MT/1000 MWe</u>
First Wall Structure	SS-316	468
Header Manifolds and Vertical Structure	SS-316	1,300
Breeder and Moderator Cladding Material	SS-316	387
Neutron Multiplier	Be	235
Breeder Material	LiAlO_2	256
	^6Li	24
	^7Li	3
Li in Divertor	natural Li	175

however, the lithium is tied up in various chemical compounds. Therefore, the total initial feedstock to supply the blanket inventory is $320 + 3 + 175 = 498$, or about 500 MT of natural lithium per 1,000 MWe.

Lithium and beryllium replacement in this design will include burnup and processing losses. A burnup of 400 kg/1,000 MWe-yr results from tritium breeding. Assuming most of this is ^6Li , an additional 5.3 MT of natural lithium in the divertor can also be made up from the tails from the enrichment feedstock, giving a total requirement of about 22 MT/1,000 MWe-yr of lithium. Processing losses are assumed to be 10% of the annual requirements. With the blanket replaced every 2 years, as in UWMAC-II, the processing losses will be 1.2 MT/1,000 MWe-yr and 0.15 MT/1,000 MWe-yr for ^6Li and ^7Li , respectively. The feedstock of natural lithium required for 1.2 MT of ^6Li is 16 MT. The ^7Li would be more than supplied by this feedstock.

Beryllium burnup will be minimal (0.1%). However, processing losses will amount to 12 MT/1,000 MWe-yr.

2.3 FUEL SYSTEMS

Fuel systems for magnetic fusion reactors typically consist of two major subsystems: the plasma recycle system and the blanket recovery system. The plasma system collects, processes and recycles unburned tritium and deuterium from the plasma exhaust. The function of this system will not vary greatly with different blanket concepts. The blanket system collects, purifies and stores tritium bred in the blanket. Processing steps in the blanket recovery system can vary significantly depending on the type of blanket used. For this analysis two blanket recovery systems will be described: one for the liquid lithium blanket and one for the gas-cooled solid breeder blanket.

Descriptions of the fuel system include estimates of the deuterium, tritium and helium required for plant operation. The amounts of deuterium and tritium lost through leakage or radioactive decay are also estimated.

2.3.1 Plasma Recycle System

The plasma recycle system for all recent designs is typically based on the use of cryosorption or cryocondensation pumps for vacuum system pumping.

Cryodistillation is used to separate hydrogen isotopes. The generic components of this system are shown in Figure 2.5. Several variations are possible: designs relying on magnetic divertors for impurity control may have high gas flows that require additional pumping in the divertor region; some designs replace neutral beam heating with RF heating. However, the breakdown shown in Figure 2.5 is generally sufficient to identify major inventories of deuterium and tritium.

The plasma exhaust typically consists of unburned fuel and a small concentration of low Z impurities. Constituents expected in the TNS⁽¹⁹⁾ exhaust include:

<u>Constituents</u>	<u>Molecular Species</u>
D	D ₂ , DT
T	T ₂ , DT
He (10-20%)	He
H (1%)	DH, TH, H ₂ O
C (0.02-1%)	Methane/Acetylene
O (3%)	H ₂ O, CO, O ₂
N (0.06%)	N ₂ , ammonia
Ar (0.1%)	Ar
Si (0.02%)	Silanes

High Z impurities that may sputter off the vessel first wall are not expected to be transported back into the fuel purification system⁽²⁰⁾, and will not be discussed here.

The stream from the plasma exhaust is transferred to a fuel purification system where nonhydrogen species are removed. A typical output from the purification process for Tritium Systems Test Assembly⁽²¹⁾ is shown below:

<u>Constituent</u>	<u>Concentration</u>
D ₂	25%
T ₂	25%
DT	48%

FIGURE 2.5. Generic Fusion Plasma Recycle System

HD	1%
HT	1%
H ₂	0.014%

This purified stream is then transferred to the isotopic separations unit. Typical output streams expected in the Tritium Systems Test Assembly⁽²¹⁾ are:

Stream	Concentration %	Stream	Concentration %
H ₂ , HD species waste			
HD	98	D ₂ output	
H ₂	2	D ₂	99.6
HT	10 ⁻⁴	DT	0.31
D ₂	10 ⁻⁶	HT ₂	5 x 10 ⁻²
DT output		HD	3 x 10 ⁻²
DT	92	T ₂	9 x 10 ⁻⁵
T ₂	4.7	T ₂ output	
D ₂	3.6	T ₂	98.9
HT	10 ⁻⁴	DT	1.1
HD	10 ⁻⁷	D ₂	2 x 10 ⁻⁵

The various output streams are then combined in the appropriate mixture for fuel injection.

As an example of the type of system involved, the plasma recycle system for the GA-DEMO is shown in Figure 2.6. In this system, gas exiting the plasma chamber first encounters the flowing plasma boundary (FPB) and auxiliary cryo-sorption pumps. Acetylene and other volatile impurities are expected to be trapped out on the liquid nitrogen-cooled chevrons of these pumps. Normal regeneration (at 80 K or less) of the cryopumps will not release these impurities, but occasional special regenerations at higher temperatures will remove obstructing deposits. It should be possible to collect the high-Z impurities in concentrated form by preceding a special regeneration with a thorough normal regeneration. The tritium content of these impurities is recovered in a batch process by burning (oxidizing) the special regeneration effluent, collecting the water and utilizing the normally idle electrolysis unit of the primary

FIGURE 2.6 GA-DEMO Plasma Recycle System(15)

coolant purification system to break down the water. The hydrogen isotopes thus recovered are returned to the plasma recycle stream prior to liquefaction.

The more volatile principal components of the recycle stream (D, T, H, He) pass through the liquid nitrogen-cooled chevrons and are condensed or sorbed on the 4.2 K molecular sieve (e.g., Linde 4A or 5A) surface of the cryopumps. These pumps are regenerated on a regular schedule for both the FPB and auxiliary pumps by valving them off from the main chamber, heating them up by simply shutting off the liquid helium flow and pumping away the effluent. As short an operating schedule as appears reasonable is selected to minimize tritium inventory. This is every 2 hours in the GA-DEMO.

Oil droplets originating in the pump train, water droplets and any particulates are then removed in a separator/fog filter unit. The gas stream enters a liquid nitrogen-cooled refrigerator-dryer where less volatile impurities (i.e., water vapor) are condensed and the temperature is decreased. Passage through a silica gel purifier at 65 K removes more volatile impurities such as O_2 and N_2 . The gas then enters the liquefier where liquefaction via Joule-Thompson expansion occurs. The product is collected in a liquid D-T storage tank where its isotopic content is monitored. The more volatile helium is presumed to have left the liquid by this stage. When the protium concentration equals 1.0, the cryodistillation isotope separation unit is activated.

The cryodistillation unit consists of two columns and an equilibrator. The feed to the first column is assumed to consist of a high temperature equilibrium distribution of the six molecular forms of hydrogen. With an overall protium (H) atom fraction of 1, the molecular concentrations become: 25% T_2 , 25% D_2 , 48% DT, 1% HT, 1% HD, and 0.014% H_2 . To minimize column size and tritium inventory, about half the D_2 is withdrawn with the overhead of the first column. This results in some losses of both deuterium and tritium.

The purified D-T stream from the distillation units is returned to the storage tank from which a continuous liquid flow is extracted for the FPB pellet injectors. A small amount of the liquid is passed through a warmup coil and returned, as gas, to the gaseous D-T storage tank to provide makeup for the once-per-cycle backfill of the plasma chamber.

Inventories for several components of this system are given below in Table 2.8.

TABLE 2.8. Tritium and Deuterium Inventories
in the GA-DEMO Plasma Recycle System

<u>Component</u>	<u>Inventory (g)</u>		<u>Form</u>
	<u>Tritium</u>	<u>Deuterium</u>	
Cryopumps	1,540	1,030	Solid
Liquefaction	600	400	Liquid
Purification	20	13	Liquid
Pellet Manufacture	1,000	670	Liquid

As can be expected, the inventories of deuterium and tritium will be highly design dependent. The fueling rates will also be highly dependent on fractional burnup of fuel in the plasma, as shown in Table 2.9. The fractional burnup ranges from 4.1 for the GA-DEMO to 7.2 in UWMAK-I. However, the yearly burnup on a per 1,000 MWe basis is consistent between designs, varying with their thermal conversion efficiencies.

As an estimate of inventories in the plasma recycle system, the average fueling rates given in Table 2.9 will be used. A one day supply is assumed, giving 7.2 kg of tritium and 4.8 kg of deuterium for a 1000 MWe plant. Helium production rates are included, and will be used later to determine helium replacement requirements.

2.3.2 Blanket Recovery System

The blanket recovery system deals almost exclusively with tritium. Deuterium will only be present as an impurity due to diffusion or leakage from the plasma fuel system. In the following paragraphs, the blanket recovery systems for the two blanket designs are analyzed; the liquid Li blanket and the gas cooled solid Li breeder.

Blanket Option 1 - A Liquid Lithium Cooled Tokamak

An yttrium metal extraction bed is proposed for the blanket recovery system in a liquid lithium tritium breeding system. The UWMAK-I plant⁽²⁾

TABLE 2.9. D-T Fueling, Burnup and He Production

Design	Fueling Rate (kg/1,000 MWe-yr)		Burnup (kg/1,000 MWe-yr)		He Production (kg/1,000 MWe-yr)
	D	T	D	T	HE
GA-DEMO	2,018	3,027	83	124	166
UWMAK-I	1,435	2,150	103	155	206
UWMAK-II	<u>1,835</u>	<u>2,753</u>	<u>90</u>	<u>132</u>	<u>180</u>
Average	1,763	2,643	92	137	184

gives the most comprehensive design for a liquid lithium cooled tokamak, and therefore the blanket recovery system described below is taken from that design.

The yttrium metal acts as an absorption bed to remove LiT from the flowing lithium steam diverted from the main cooling system. In UWMAK-I, the coolant temperature exiting from the blanket is limited to 500°C with stainless steel, and the yttrium bed would operate at 283°C.

Extraction of the tritium from the yttrium occurs by thermal regeneration. The yttrium bed is isolated from the lithium circuit and the bulk of the lithium is drained from the bed. The temperature of the bed is increased to 400°C and the remaining lithium is distilled by vacuum distillation. At 400°C, good separation between the lithium and the tritium should be possible because the vapor pressure of Li is 9.3×10^{-5} torr while the tritium pressure over the bed is only 7.7×10^{-8} torr. After removal of the lithium, the temperature of the bed is increased to 500°C where the tritium pressure is 1.46×10^{-5} torr so that the tritium gas can be pumped by mechanical techniques into suitable storage containers. It may be necessary, however, to heat the yttrium to 700°C for several hours to dissolve any surface oxide and reactivate the surface. A flow diagram of the entire blanket recovery system for the liquid metal UWMAK-I design is shown in Figure 2.7.

Blanket Option 2 - A Solid Lithium Compound Blanket Cooled by Helium

In gas-cooled blankets that use solid breeding materials, two methods of tritium recovery can be used. The first is to release the tritium to a flowing helium stream directly from the breeding material. The material would be formed and packaged such that the only significant resistance to the flowing tritium would be the breeding material itself. The tritium can then be released directly to the main coolant stream, or to a small helium plenum that will sweep the gas directly to a tritium extraction system external to the blanket.

The second method of recovery is to design the breeding material and container to restrict permeation. The material would then be removed from the blanket at regular intervals for chemical processing in a separate facility.

Both blanket recovery systems have been considered for the UWMAK-II blanket design presented earlier. The breeder material and beryllium are placed inside stainless steel tubes; these tubes are either closed at the end facing the plasma and connected at the other end to a small gas plenum attached to a tritium extraction system external to the reactor, or the exhaust ends of these tubes are closed using a porous metal plug. This plug retains the ceramic powder inside the tube but permits the tritium to diffuse into the helium coolant where it is extracted. This latter technique increases the tritium inventory within the tubes, increases the partial pressure of tritium in the helium coolant, and subsequently increases the permeation of tritium through the primary heat exchanger. The design philosophy chosen here, however, is one that minimizes the tritium contamination in the helium coolant. The tritium is extracted directly from the tubes in a separate recovery system.

A diagram outlining the main coolant flows of UWMAK-II and showing the tritium extraction systems is given in Figure 2.8. The tritium gas is removed from the tubes by a flowing helium stream. The helium coolant inventory in this design is 11.8 MT, or ~ 7 MT per 1,000 MWe. For a helium pressure inside the tubes of 1.4 atm, the tritium pressure can be kept to 10^{-4} torr. The calculated tritium release rate to the environment is 0.8 Ci/day.

FIGURE 2.7. UWMAK-I Blanket Recovery System⁽³⁾

The tritium inventories in the UWMAK-I and II designs were used to establish the inventories in the blanket recovery system for the two reference design options. Tritium inventory in the blanket recovery system equals the total plant inventory minus the tritium present in the fuel recovery system (which includes storage). The latter value is assumed to be 7.2 kg of tritium for any design concept on a per 1,000 MWe basis. The total plant inventories of tritium for UWMAK-I and II are 23.5 and 17.5 kg (15.7 and 10.2 kg/1,000 MWe). This puts the blanket recovery system inventories at 8.5 and 3.0 kg/1,000 MWe for Options 1 and 2, respectively. The liquid lithium design will then have a greater total tritium inventory than the gas-cooled design.

Deuterium inventories in the blanket are negligible.

2.3.3 Loss Terms for Tritium and Deuterium

To determine material balances in the fusion cycle, loss terms other than fusion burn-up must be identified. The following items should be considered:

- permeation
- leakage from fluid system connections
- leakage from valves, pumps, blowers, etc.
- losses of tritiated water from the power system
- maintenance of power system components
- contamination of work areas and personnel during maintenance
- losses associated with handling and disposal of tritium
- contaminated wastes
- radioactive decay of tritium.

Source terms have been estimated for the UWMAK-I and II designs,⁽²²⁾ and upper limits for tritium loss rates are given in Tables 2.10 and 2.11, respectively. Again, the numbers have been normalized to 1,000 MWe output. The question marks indicate rough estimates. The G, L and S stand for gas, liquid and solid.

Though the tritium releases may be important from a radiological standpoint, for a materials balance the numbers are insignificant. A 26,291 Ci/1,000 MWe-yr release from UWMAK-I corresponds to ~ 2.6 g/1,000 MWe-yr. This amount is insignificant compared to the 155,000 g/1,000 MWe-yr used in fusion burnup.

FIGURE 2.8. Coolant Flows for UWMak-II⁽¹⁾

TABLE 2.10. Tritium Leakage Terms for Design Option 1(22)

Source	Tritium Released Ci/1,000 MWe-yr	
	Containment	Environment
1. Blanket(B) and Shield(S)	---	---
a. Diffusion from blanket into vacuum gap	---	---
b. Diffusion from shield to vacuum gaps in shield	---	---
c. Diffusion from shield into helium cooling system	---	---
d. Leakage from B and S fluid connections	<25?(G,L)(a)	<3?(G)
e. Maintenance on B and S equipment	<243?(G,S)	<25?(G)
2. Shield Cooling System		
a. Leakage from valves, pumps, etc.	<25?(G)	<3?(G)
b. Cleanup of shield cooling system helium	---	---
c. Maintenance on shield cooling system	<243?(G,S)	<25?(G)
3. Piping Systems (Li, Na)		
a. Diffusion through lithium system piping	1,533(G)	153(G)
b. Diffusion through sodium system piping	37(G)	4(G)
c. Diffusion through walls of sodium storage tanks	39(G)	4(G)
d. Leaks from valves, pumps, etc.	<25?(G,S)	<3?(G)
e. Maintenance on lithium and sodium system piping	<243?(G,s)	<25?(G)
4. Heat Exchangers		
a. Diffusion through IHX Shell	11(G)	1(G)
b. Diffusion through steam generator shell	1(G)	0.1(G)
5. Power Systems		
a. Diffusion through steam generator into power system	---	2512(L)
b. Leaks from power system	<243?(G,L)	<25?(G)
c. Maintenance on power system components	<243?(G,L)	<25?(G)
6. Tritium Extraction Systems		
a. Permeation losses	4000(G)	400(G)
b. Leaks from valves, pumps, etc.	<25(G)	<3?(G)
c. Maintenance on ³ H extraction systems	<243?(G)	<25?(G)

TABLE 2.10. (cont'd)

Source	Tritium Released Ci/1,000 MWe-yr	
	Containment	Environment
7. Diverter Systems		
a. Diffusion through lithium system piping	8(G)	1(G)
b. Diffusion through sodium system piping	small(G)	negligible
c. Diffusion through IHX and steam generator to power system	---	---
d. Tritium extractions system permeation losses	39(G)	4(G)
e. Permeation losses from vacuum system	68(G)	7(G)
f. Maintenance on diverter system equipment	<243?(G)	<25?(G)
g. Leakage from valves, pumps, etc.	<25?(G)	<3?(G)
h. Contaminated mercury from	---	---
8. Fuel Handling Systems		
a. Losses during fuel preparation and injection	<243?(G)	<25?(G)
b. Losses during storage (assume storage as gas)	18,000 (G)	2,000(G)
c. Leakage from isotopic separation processes	<243?(G)	<25?(G)
d. Maintenance on fuel handling systems	<243?(G)	<25?(G)
9. Waste Handling Systems		
a. Contaminated clothing	---	---
b. Contaminated pump oils	---	---
c. Disposal of contaminated yttrium from tritium extraction beds	- -	---
d. Disposal of contaminated charcoal from diverter	---	---
e. Disposal of contaminated ion exchange resins from power systems	---	---
TOTAL FOR ALL SYSTEMS	26,291	5,400

(a) = Estimate

G = Gas

L = Liquid

S = Solid

TABLE 2.11. Tritium Leakage Terms for Design Option 2(22)

Source	Tritium Released Ci/1,000 MWe-yr	
	Containment	Environment
1. Blanket and Shield (assume shield same as in UWMak-I)	---	---
a. Diffusion from blanket into vacuum gap	---	---
b. Diffusion from shield into vacuum gaps in shield	---	---
c. Diffusion from shield into shield helium cooling system	---	---
d. Leakage from B S fluid connections and penetrations	<22?(G)	<2?(G)
e. Maintenance on B and S equipment	<213?(G,L)	<21?(G)
2. Shield Cooling System		
a. Leakage from valves, pumps, etc.	<22?(G)	<2?(G)
b. Cleanup of shield cooling system helium	---	---
c. Maintenance on shield cooling system	<213?(G,S)	<21?(G)
3. Piping Systems (He Na)		
a. Diffusion through helium system piping	1,049?(G)	105?(G)
b. Diffusion through sodium system piping	<213?(G)	<21?(G)
c. Leaks from valves, pumps, etc.	<22?(G,S)	<2?(G)
d. Maintenance on piping systems	<213?(G,S)	<21?(G)
4. Heat Exchangers		
a. Diffusion through IHX shell	<22?(G)	<2?(G)
b. Diffusion through steam generator shell	<22?(G)	<2?(G)
5. Power Systems		
a. Diffusion through steam generator into power system released via blowdown	---	170(L)
b. Leaks from power system components	<213?(G,L)	<21?(G)
c. Maintenance on power system components	<213?(G,L,S)	<21?(G)
6. Tritium Extraction Systems		
a. Permeation losses	<22?(G)	<2?(G)
b. Leaks from valves, pumps, etc.	<22?(G)	<2?(G)
c. Maintenance on ³ H extraction systems	<213?(G,S)	<21?(G)

TABLE 2.11. (cont'd)

Source	Tritium Released Ci/1,000 MWe-yr	
	Containment	Environment
7. Diverter Systems		
a. Diffusion through lithium system piping	6(G)	1(G)
b. Diffusion through sodium system piping	small	negligible
c. Diffusion through IHX and steam generator to power system	---	<0.1(L)
d. Tritium extraction system permeation losses	107(G)	11(G)
e. Permeation losses from vacuum system	298(G)	30(G)
f. Leakage from valves, pumps, etc.	<22?(G)	<2?(G)
g. Maintenance on diverter system	<213?(G,S)	<21?(G)
h. Contaminated mercury from vacuum pumps	---	---
8. Fuel Handling Systems		
a. Losses during fuel preparation and injection	<213?(G)	21?(G)
b. Losses during storage (assume storage as gas)	15,734?(G)	1,573?(G)
c. Leakage from isotopic separation processes	<213?(G)	<21?(G)
d. Maintenance on fuel handling systems	<213?(G)	<21?(G)
9. Waste Handling Systems		
a. Contaminated clothing	--	---
b. Contaminated pump oils	---	---
c. Disposal of contaminated yttrium from tritium extraction beds	---	---
d. Disposal of contaminated charcoal from diverter	---	---
e. Disposal of contaminated ion exchange resins from power systems	---	---
TOTAL FOR ALL SYSTEMS	19,713	2,326

(a) = Estimate
 G = Gas
 L = Liquid
 S = Solid

Deuterium will be subject to the same control measures as tritium, and hence its leakage will not affect the material balance significantly. However, because of the radioactivity of tritium, the control measures required for tritium containment are very important in fusion systems design. These measures will be discussed in more detail in the next chapter.

Radioactive decay of tritium will in fact present a much more important loss term to the material balance than leakage terms. Decay loss rates will be approximately 5.4% of the steady-state inventory, or 0.85 kg/1,000 MWe-yr and 0.55 kg/1,000 MWe-yr in the reference liquid lithium and gas-cooled designs, respectively. This is based on the previously calculated inventories of 15.7 and 10.2 kg/1,000 MWe for Options 1 and 2, respectively. This decay term must then be added to the fusion burn-up, which was put at 137 kg 1,000 MWe-yr, to determine yearly requirements. This gives 137.85 and 137.55 kg/1,000 MWe-yr for Options 1 and 2, respectively. This will be rounded to ~140 kg/1,000 MWe-yr for both designs.

2.3.4 Helium Inventories and Loss Terms

In the designs presented to this point, helium has been used in a number of systems to provide cooling. This includes liquid helium for cryogenic cooling of field coils and energy storage coils and gaseous helium for blanket cooling in the gas-cooled solid breeder concepts.

Helium inventories estimated for the University of Wisconsin designs are shown in Table 2.12. As can be seen from the table, inventories associated with the coils and storage do not vary significantly between designs. An average of 40 metric tons per 1,000 MWe is calculated. However, the inventory associated with energy storage is very design dependent. The inventories range from 53 MT/1,000 MWe in UWMAK-I to 7.5 MT/1,000 MWe in UWMAK-III. It will be assumed here that the more recent designs of UWMAK-II and III, where inventories are about 10 MT/1,000 MWe, give the best indication of helium requirements for energy storage.

For the gas cooled designs (UWMAK-II and III), helium inventories for blanket cooling are estimated to be ~10 MT/1,000 MWe.

TABLE 2.12. Helium Inventories

<u>Design</u>	<u>Component</u>	<u>Liters (x1,000)</u>	<u>MT</u>	<u>MT/1,000 MWe</u>
UWMAK-I (1500 MWe)	TF Coils	100		
	VF and Ohmic Coils	150		
	Storage	<u>200</u>		
		450	57	38
	Energy Storage		80	53
	Cooling		<u>1</u>	<u>0.66</u>
			138	92
UWMAK-II (1710 MWe)	TF Coils	160		
	VF and Ohmic Coils	150		
	Storage	<u>300</u>		
		610	79	46
	Energy Storage		15	9
	Cooling		<u>11.8</u>	<u>7</u>
			105.8	62
UWMAK-III (1985 MWe)	TF Coils	144		
	VF and Ohmic Coils	100		
	Storage	<u>300</u>		
		544	68.5	34.5
	Energy Storage		15	7.5
	Cooling		<u>20</u>	<u>10.0</u>
			103.5	52

No annual replacement terms have been identified for boil-off from the coils or liquid storage in the above designs. Estimates made for the Princeton design put losses at about 1.5 kg/day.⁽²³⁾ With an output of 2,030 MWe, this indicates a loss rate of 270 kg/1,000 MWe-yr.

Estimates for helium loss rates from the gas-cooled design UWMAK-II are as low as 0.323 kg/day⁽³⁾. However, some consider this highly optimistic given experience in gas-cooled fission reactors.⁽²³⁾ The actual loss rates will hinge on the performance of the tritium control system which removes tritium from the circulating He coolant. If tritium is maintained at ~1 ppm in the coolant, the loss rates of several kg per day of He found for fission reactors will be unacceptable from a radiation standpoint. It is therefore likely that the primary piping will be contained in a second system where leaking helium would be processed to remove tritium. This second system is presumed to be capable of returning helium to the main coolant loop.

As a first estimate of the helium loss constraints that must be met for tritium containment, we will use the release terms in Table 2.12. Item 3 gives routine losses of $1,049 + 213 + 22 = 1,284$ Ci per 1,000 MWe-day. Assuming a 1 ppm concentration of tritium in the helium coolant, this is equivalent to 469 g He lost per 1,000 MWe-day, or leakage of about 0.5 kg/1,000 MWe-day. The loss term for helium from a gas cooled system is therefore estimated at 183 kg/1,000 MWe-yr. The gas-cooled design (Option 2) will then lose $183 + 270 \approx 450$ kg/1,000 MWe-yr of helium from both the cryogenic and blanket cooling systems.

The helium production rate in the plasma was identified earlier as 184 kg/1,000 MWe-yr. Assuming this is recovered without loss as part of the fuel purification process, the helium make-up for design option 1 and 2 will be 86 and 270 kg/1,000 MWe-yr respectively.

2.4 MATERIAL BALANCE

Table 2.13 summarizes the material requirements identified in this chapter for the two design options. All numbers are normalized to a 1,000 MWe fusion power plant.

The levels of material production that will be needed to supply the amounts in Table 2.10 are described in the next section. Potential effluents from material production are also described. Potential effluents from the fusion fuel cycle are summarized in Section 4.

TABLE 2.13. Fusion Material Balance

<u>Material</u>	<u>Design Option 1</u>		<u>Design Option 2</u>	
	<u>Inventory</u> <u>(MT/1,000 MWe)</u>	<u>Replacement</u> <u>(MT/1,000 MWe-yr)</u>	<u>Inventory</u> <u>(MT/1,000 MWe)</u>	<u>Replacement</u> <u>(MT/1,000 MWe-yr)</u>
Niobium	45	0	45	0
Lithium	730	1	500	22
Beryllium	0	0	235	12
Helium	50	0.086	60	0.270
	<u>(kg/1,000 MWe)</u>	<u>(kg/1,000 MWe-yr)</u>	<u>(kg/1,000 MWe)</u>	<u>(kg/1,000 MWe-yr)</u>
Deuterium	8	92	8	92
Tritium	~16	~140	~10	~140

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3.0 PRODUCTION EFFLUENTS FOR MATERIALS

In order to assess the increments of environmental releases that may be attributed to the fusion fuel cycle, the processes for extracting or mining and refining deuterium, tritium, helium, lithium, beryllium, and niobium are described and typical effluents are identified. Using the amounts of materials estimated to be required for a 1000 MWe fusion power plant, as identified in the previous section, the relative amounts of effluents are estimated.

In general, the amounts of effluents released from production of these materials are within current standards. Thus, the increment of effluents due to the fusion economy is assumed to cause no additional need for effluent control.

3.1 DEUTERIUM

Deuterium can be obtained either through electrolysis of heavy water or cryogenic distillation of hydrogen. The cryogenic distillation process is utilized more in Europe than in the United States or Canada. Commercial suppliers of D in the United States use electrolysis of heavy water (99.75 wt% D₂O) to produce deuterium.

The heavy water is obtained from the Department of Energy's Savannah River plant, which produces about 165 MT D₂O per year. Heavy water is also imported, under the supervision of Department of Energy (DOE). The largest supplier of D₂O is Canada, which has several facilities:

Port Hawkesbury, Nova Scotia	360 MT/year
Glace Bay, Nova Scotia	360 MT/year
Bruce, Ontario	800 MT/year
	800 MT/year (under construction)
	800 MT/year (approved)
	800 MT/year (planned)
Gentilly, Quebec	800 MT/year (approved)

The cost of heavy water is about \$225/kg from either domestic or Canadian facilities.

To describe the production of deuterium, the processes at the facility in Ontario are used as a reference.⁽²⁾ Effluents from deuterium production and fusion reactor operation and environmental standards that apply to deuterium are described in following sections.

3.1.1 Deuterium Production

The basic process through which deuterium is produced is shown in Figure 3.1. Water is pumped from a lake and filtered to remove solids. Treatment of water includes adding of sulfuric acid to the water to lower the pH and then degassing the water of oxygen and carbon dioxide. The treated water is fed to the enrichment stages.

The enrichment stages consist of an absorption and desorption section, an extraction section and an enriching section. Enrichment utilizes the Girdler Sulfide (GS) process, which is a bi-thermal process based on chemical exchange between hydrogen, hydrogen sulfide, and liquid water. The absorption and desorption process removes H_2S from depleted water before it is returned to the lake and removes H_2S from the purge gas before it is flared. As the feed water passes through this section, it is partially saturated with H_2S .

FIGURE 3.1. Deuterium Processing Steps

Next in the extraction section, water passes counter-current to a recirculating stream of hydrogen sulfide (H_2S) gas in three large sieve tray towers operating in parallel; these towers compose the first stage and are illustrated in Figure 3.2. By utilizing a two-temperature exchange reaction between hydrogen sulfide (H_2S) and water, deuterium is concentrated in the liquid phase at 30° and deuterium is concentrated in the gas phase at 130° . Each of the three first-stage towers (see Figure 3.2) has two distinct process temperature sections--the top is a cold section and the bottom is a hot section. Deuterium is carried forward to the second stage only in the gas phase, while anything not extracted from the water is returned to the lake from the first stage after it has been stripped of H_2S and cooled.

The second and third stages enrich or concentrate the deuterium that has been extracted. The second stage is a single tower whose operation is similar to the first stage towers. However, there is no heating section at the bottom of the hot tower as the gas coming off the first stage is already hot. To maintain liquid circulation in the second stage, water is pumped from the

FIGURE 3.2. Girdler-Sulfide Process⁽²⁾

bottom of the hot tower through a cooler to the top of the cold tower. The third stage consists of a cold tower and a hot tower. In the third stage, enriched liquid from the bottom of the cold tower is taken forward for further processing. The H_2S in this enriching unit's liquid product is stripped out and returned to the third stage gas loop.

Enriched water at 20 to 30% D_2O concentration from both enriching units is fed to the finishing unit. The finishing unit is a three-stage, four-tower stream vacuum distillation system that concentrates the product from the enriching units into the final reactor-grade heavy water. All finishing unit towers contain sieve trays.

Although the annual deuterium requirements for fusion are small compared to the D_2O production capabilities of Canada, facilities that produce pure D gas have small capacities and are not equipped to supply the large quantities of D required by the mature fusion industry. To avoid the limitations of D gas production, a separate system in the fusion facility would produce pure D from D_2O supplied by DOE. This system would be integrated into the fusion reactor fuel system.

Electrolysis is used to separate deuterium from heavy water. Heavy water electrolysis has a separation factor of about 6. A variation of conventional electrolysis called bipolar electrolysis can achieve separation with lower power consumption (50 to 80% less than conventional electrolysis)⁽⁴⁾ and without production of gaseous hydrogen between stages.

In bipolar electrolysis, a conductive barrier inserted into the voltage gradient between the anode and cathode of a conventional electrolysis cell isolates the two electrodes. The barrier acts as both anode and cathode, thus becoming bipolar. The electrode reactions at the bipolar electrode are similar to a conventional electrolysis but differ in the following ways:

- Gaseous hydrogen is not generated at the cathode surface.
- Atomic hydrogen is sorbed and transported through the electrode by diffusion.
- Oxidation of the hydrogen takes place at the anode surface to form water.

The bipolar electrode process must be porous to hydrogen and also hydrophobic to maintain the separation of the electrolyte. The protium-rich material in the terminal cathode compartment must be isolated from the deuterium-rich solution in the terminal anode compartment. The use of a very thin metal membrane fulfills this requirement. Palladium-25% silver can be used as a bipolar electrode material that readily sorbs and diffuses hydrogen.

Hydrogen flow is established through the cascade toward the terminal cathode. The mass transfer process favors the transfer of the lighter isotopes. In the aqueous phase adjacent to the cathode side of the bipolar electrodes, enrichment of the heavier isotope takes place. To achieve maximum separation, the deuterium-rich aqueous solution is swept from the cathode surface (bipolar electrode) and circulated countercurrent to the movement of the hydrogen through the cascade.

There may be some difficulties with scaling up the bipolar electrolysis process to obtain reasonable cell size and throughput. Throughput can be increased by increasing the temperature; however, this decreases the separation efficiency.

Figure 3.3 shows a diagram of the conventional electrolysis process used to separate deuterium from heavy water. The electrolysis section consists mainly of four electrolysis stacks. The stack has eight cells each with an area of about 48 cm^2 . The current used is 50 amps with a rated stack voltage of 15.8 Volts DC.⁽²⁾ Each stack electrolyzes 2.45 cm^3 of water every minute. This results in 3000 cm^3 hydrogen/minute and 1500 cm^3 oxygen/minute.

3.1.2 Deuterium Release Standards

Deuterium isotope effects on biological systems have been studied since the separation of this isotope in 1932 and have been examined more intensively during the past 15 years. Despite the many detailed investigations of the biosphysics, biochemistry, physiology and pathology of deuterium isotope effects, it is still not fully understood why higher organisms fail to cope with more than 30 to 35% substitution of D_2O for H_2O in body fluids.

FIGURE 3.3. Electrolysis Process for Deuterium Separation⁽²⁾

Deuterium or heavy water in high concentrations has been found to be deleterious in animals. A threshold concentration for ambient water of 3500 mg/l is 1/100 the level at which replacement of body water causes death.

The chemical toxicity of tritium is also based on the above rationale to obtain a desired maximum ambient water concentration of 3500 mg/l. However, since the specific activity of tritium is very high (9.6×10^3 Ci/g) its release to the environment must be based upon its radiological dose consequence.

3.1.3 Effluents and Control of Deuterium Production

The environmental problems associated with the production of heavy water are: 1) H_2S emissions to water and air, 2) thermal releases, 3) solid wastes, 4) SO_2 and NO_x , and 5) salts. The environmental pollutant of greatest concern is H_2S .

The chief disadvantage of using the Girdler-Sulfide process from the occupational viewpoint is the toxicity of H_2S gas and the safety hazards it

poses. If concentrations reach more than $200 \mu\text{g}/\text{m}^3$, H_2S can be smelled by most people. Inhalation of H_2S results in acute effects. Hydrogen sulfide is also flammable and under certain circumstances can be explosive when mixed with air.

Hydrogen sulfide releases to the atmosphere can be limited by the following control measures:

- maintaining a combustible mixture of gases in the stack
- diverting water containing unacceptable H_2S levels to holding ponds
- improving the $\text{H}_2/\text{H}_2\text{O}$ stripper
- minimizing drain water that has to be stripped of H_2S
- accompanying releases of steam or nitrogen and other inerts to the flare by increased propane flows to the flare.

An H_2 recovery system to minimize H_2S and propane usage is being built as part of the expansion at the Ontario plant. This system will further reduce H_2S releases to the flare.

All water streams containing H_2S must be stripped of this compound before the water is returned to the lake. Unstable flows in the towers cause the effluent stripper operating conditions to fluctuate and deter its effectiveness. Changes can be made to tower tray weirs; the addition of antifoam to the process water steadies operating conditions and therefore improves the effectiveness of the stripper. Replacement of unreliable instrumentation and better process control during startup and shutdown will improve performance. A continuous H_2S -in-water monitor that possesses a quick response time is used on the stripper outlet. This monitor plus other effluent stripper process parameters (feed temperature, level and pressure) have been interlocked to automatically divert effluent to the lagoons if a preset concentration of H_2S or the measured variables exceeds defined limits.

The effluents resulting from heavy water production can be seen in Table 3.1. The environmental effluents and control measures used to limit

TABLE 3.1. Environmental Effluents from Deuterium Production^(a)

	<u>Bruce Heavy Water Plant</u>	<u>1000 MWe Fusion Reactor</u>
Deuterium Production (metric tons per year)	640	0.092
Land Area (acres)	10,300	1.5
Water Usage (gallons)	1,400,000	200
Heat Release (megawatts thermal)	1,850	0.27
Release to Water		
H ₂ (kilograms per year)	2,300	0.33
Salts	< 2,000	0.29
Release to Air		
SO ₂ (metric tons per year)	30,000	4.3
NO _x (metric tons per year)	10,000	1.4
Solid Waste	10,000,000	1,400

^(a) estimated for Bruce, Ontario, heavy water plant (3200 MT D₂O/yr or be 40 MT D₂/yr equivalent.

these releases are significant for the entire heavy water production industry but are trivial when considering the fusion increment and its minimal impact on the heavy water industry. The production of purified isotopic deuterium from heavy water by electrolysis requires no special environmental control measures.

The deuterium requirement, 0.092 MT/GWe-year, was determined in Section 2 of this report. Heavy water (D₂O) production losses are assumed to be about 1% of annual production. The quantity released would be about 4.6 kg D₂O/year. This D₂O would end up in bedding tanks entrained with other processing wastes. There will be some releases of D₂ to the atmosphere at the heavy water production facility.

During normal operating conditions the electrolysis unit will not release any harmful pollutants to either the air or water. It is possible that a ruptured electrolytic cell could release D_2O feed water and in some instances the entire quantity of D_2 present in the process streams. The primary concern in this situation would be to avoid an explosive mixture of gases within the electrolytic cell room.

3.2 TRITIUM

Tritium will be used with deuterium to fuel the fusion reactors. Tritium for fusion reactor operations can be obtained from the U.S. stockpile, or from production in either light water reactors (LWR) or a special heavy water production reactor.

To acquire the tritium needed for startup, production must be planned well before reactor operations begin. As illustrated in Figure 3.4, production in fission reactors is estimated to result in about 40 kg of tritium by the year 2000. This would be sufficient to start one to three fusion reactors. Tritium production in an LWR is accomplished by inserting a lithium-bearing target in the reactor.

After startup, this T would be bred in the Li blanket modules of a fusion reactor; the supply of T would be sustained by the fusion reactor, and no make-up shipments would be required. In a mature fusion economy excess tritium from fusion reactors could be used to supply the startup requirements for new fusion reactors.

Descriptions of tritium production methods external to fusion reactors, potential effluents from production transportation, and tritium control measures in general are presented in the following subsections.

3.2.1 Tritium Production Methods

Tritium can be produced by several types of nuclear reactions in fission and neutron irradiation of lithium, boron, and deuterium. Tritium can also be produced by beryllium-neutron reactions and helium-3 thermal neutron reactions. In light water reactors, tritium is produced by neutron irradiation of deuterium, helium-3, boron, and lithium-6 in the reactor coolant. Of these,

FIGURE 3.4. Tritium Production in LWR Industry⁽³⁾

production of tritium in boron is the most significant source. Tritium is produced by fission, however, in much greater quantities than all of the other sources in a light water reactor. Most of this tritium is retained in the fuel rods and cladding and is released during reprocessing. If LWR fuel reprocessing were to be started in the U.S., the annual tritium generation rate at the fuel reprocessing plant could reach 190 grams by 1985, 350 grams by 1990, and 850 grams in the year 2000.⁽⁴⁾ The tritium available for fusion reactors from reprocessing will depend upon detritiation efficiencies, the resumption of LWR fuel reprocessing, and the overall growth schedule of fission reactors.

Deuterium is a source of tritium in heavy water reactors where D_2O is used as a moderator and coolant. Based on a projected growth in CANDU

reactors, annual tritium production may be 19 kg by the year 2000.⁽⁴⁾ Economical tritium separation and detritiation methods must be developed if tritium production by these methods is to be attractive.

3.2.2 Effluents Resulting from Tritium Production

The most significant effluents associated with tritium production will most likely result from the lithium aluminate irradiation process. This process, used at the Savannah River plant,^(a) is used here as a reference for estimating the releases and impact of tritium wastes.

Tritium is separated from irradiated lithium-aluminum targets and then purified. The plant operations are carried out in ventilated rooms and process hoods in which air flow is used to sweep tritium releases out the stacks. Three 61-m stacks and one 23-m stack exhaust a total of 7500 m³/min of air.

Tritium in this exhaust air comes from sources such as waste gas streams, leaks, maintenance work, as well as from processing lithium-aluminum targets. The processing facilities use a pressure varying from full vacuum to 2.5 atm. Leak-tight systems are used to prevent air inleakage and to prevent tritium release to the atmosphere. Process hoods are used for lithium target operations. In general, all equipment in contact with tritium is fabricated from 316 stainless steel because of its low permeability to tritium and its low susceptibility to hydrogen embrittlement. Gasket materials, piping, and valves are constructed so that the radiation and hydrogen embrittlement effects of tritium are minimized.

Tritium processing systems are opened routinely to the atmosphere for process reasons and periodically for maintenance and equipment repair. Any section of process line that is to be opened must be isolated from tritium gas by shutoff valves. It is then evacuated and flushed several times with

(a) The primary function of the Savannah River plant is to produce plutonium, tritium and other nuclear materials for the national defense, governmental use and some civilian purposes. A statement of the exact amount of tritium produced at Savannah River is not available.

argon. The flush gas is processed through the stack stripper, which removes the tritium before the flush gas is stacked. The system is then pressurized with argon before it is opened to the atmosphere.

Waste material or equipment contaminated with tritium is buried at a solid radioactive waste burial site at least 20 ft above the water table. Wells are sampled to detect migration of radionuclides from the burial site. Wastes are packaged within the generating facility. The container may be a metal can, crucible, box or a combination of these depending upon the nature of the waste. During handling, any wastes with contamination of more than 10^4 Ci/ft³ are isolated in welded containers. The trenches are backfilled to reduce the radiation level to 6 mrem/hr at ground level.

No high-level solid tritium wastes (over 1000 Ci/ft³) result from target processing. There are intermediate-level wastes (10 to 1000 Ci/ft³) consisting of spent furnace melts of lithium-aluminum at about 70 Ci/ft³ of tritium. Low-level tritium-contaminated wastes of paper and metal scraps are packaged and buried. Tritium activity in low-level waste is less than 0.1 Ci/ft³.

High-level liquid wastes, such as tritiated water spilled in process hoods, are unusual. In the event of an accident involving T₂O, it will be collected on absorbent material and then packaged using plastic bags and metal cans. Intermediate-level liquid wastes, such as used vacuum pump oil, are collected in polyethylene bottles and packaged with absorbent material in a metal can. Tritium concentration in used pump oil is about 40 Ci/liter. Low-level liquid wastes result from equipment decontamination containing the low levels of tritium. These wastes are drummed and transferred to a permanent disposal site.

Nonradioactive atmospheric emissions of SO₂, NO_x and fly ash result from various operations at the Savannah River plant. Emissions of SO₂ are approximately 3.5 lb/10⁶ Btu and fly ash emissions are 0.3 lb/10⁶ Btu input. There are essentially no harmful liquid effluents that could make their way into drinking water.

Tritium, ^{137}Cs and ^{90}Sr are the only radionuclides of plant origin detectable in Savannah River water. About 10^4 Ci/yr of tritium is released to streams or migrates from earthen basins in ground water. Other radionuclides that reach the river are ^{90}Sr , ^{137}Cs , ^{131}I , ^{32}P , ^{35}S , ^{144}Ce , ^{239}Pu , and U. These radionuclides are released in quantities of less than 10^{-1} Ci/yr. The most significant contributor to consumer exposure results from tritium. The release of these radionuclides is a result of spent fuel reprocessing activities at Savannah River plant. Tritium is also released during spent fuel reprocessing but most results from the processing of the special lithium target assemblies.

Atmospheric releases at the Savannah River plant are either gases and vapors or particulates. About 10^5 Ci/yr of tritium are released from the Savannah River plant. This is based on annual release data given in Reference 5. Other significant gases and vapors such as ^{41}Ar , ^{14}C , ^{85}Kr , ^{87}Kr , ^{88}Kr , ^{131}Xe , ^{133}Xe , ^{135}Xe , ^{129}I , and ^{131}I contribute to the emission rate. Particulates (Co, Sr, Ru, Cs, Ce, U, Pu, Am, Cm) also contribute to the emission rate but in extremely small quantities, less than 10^{-4} Ci/yr. Tritium is by far the major contributor to population dose (about 80%) from normal releases at the Savannah River operation.

The numbers indicated above represent all activities and production processes at Savannah River plant. The release numbers attributable to fusion will be some fraction of these values. Thus the use of the Savannah River plant annual report numbers constitutes a maximum or worst possible case.

3.2.3 Effluents from Tritium Transportation

Tritium can be shipped fixed in a solid such as $\text{Ca}(\text{OH})_2$. The ^1H and ^3H in the stream are fixed as $\text{Ca}(\text{OH})_2$ in suitable equipment. The specific activity of $\text{Ca}(\text{OH})_2$ is 5×10^5 Ci/m³ and it has a heat generation of 2 watts/m³. Tritium could be released from the $\text{Ca}(\text{OH})_2$ in the case of a severe accident. There are three possible ways to receive a dose from the tritium after an accident:

- external dose received by exposure to $\text{Ca}(\text{OH})_2$

- internal dose received from breathing contaminated water vapor driven off from decomposing Ca(OH)_2
- internal dose received from breathing contaminated water vapor that contains dissolved Ca(OH)_2 .

When Ca(OH)_2 reaches a temperature of 1076°F , water vapor contaminated with tritium will be released. This water vapor will enter the atmosphere at a rate equal to the decomposition rate of Ca(OH)_2 . The time required to heat the Ca(OH)_2 to the decomposition temperature will depend on the exposed area during the fire. If the total amount of Ca(OH)_2 was released by an accident and decomposed from the time required to raise the material from 70°F to 1076°F during an 1850°F fire, the rate of release of tritium contaminated water would be 600 Ci/min.

Tritium could also be shipped as a gas. A container similar to those used to ship krypton gas could be used to transfer tritium gas. This container would hold about 1.5 kg of tritium. The LP-12 and LP-50 shipping containers currently in use would likely have insufficient capacities for large scale fusion requirements. An accident causing the vessel to be breached would release tritium. The tritium could be released in three situations: 1) release from a break in the vessel with no fire, 2) release from the vessel during a fire, and 3) release of tritium from the vessel under water.

The tritium in these scenarios will rise because it is lighter than air. The whole inventory of the bottle is assumed to have been released instantaneously. The release of tritium from the vessel during a fire could happen in two ways: 1) the vessel is breached by the accident and tritium burned as it escapes from the bottle producing $^3\text{H}_2\text{O}$, or 2) the bottle is breached by internal pressure caused by the fire. The tritium forms $^3\text{H}_2\text{O}$ and in this case could be released to the environment if the vessel was breached and submerged in water. The total contents of the bottle would probably not dissolve in the water as some of the tritium would bubble to the surface and would be lost to the atmosphere. A certain amount of tritium could return to the earth because of rain or other weather conditions, but in a diluted form.

3.2.4 Tritium Release Standards

There are federal regulations governing the release of tritium through occupational and public radiation dose rates by limiting airborne concentrations to 5×10^{-6} and 2×10^{-7} $\mu\text{Ci/ml}$ (air), respectively (10 CFR 20). The more modern concept of regulating the emission of radioactive material to the environment is to assure regulatory agencies that the release is as low as reasonably achievable (10 CFR 50, App. I).

The EPA (1977) guidelines (effective December 1, 1979) will limit the annual radiation dose equivalent to any member of the general public from planned discharges of radioactive materials from the entire uranium fuel cycle operations to 75 mrem to the thyroid and 25 mrem to any other organ (excluding skin and cornea).⁽⁶⁾ Although these design guides are specific for light-water reactors, there is no reason to hypothesize that the regulations would change for fusion reactor plants.

Present NRC Design Guides for light-water reactors specify that the total quantity of all radioactive material (not just tritium) released in gaseous effluents shall be controlled so that the external dose equivalent to an individual in an unrestricted area does not exceed 5 mrem/yr to the whole body or more than 15 mrem/yr to the skin per reactor. The EPA has reviewed these NRC Guidelines for individual LWR's and have concluded that the NRC would have no difficulty complying with these new EPA standards even at sites with up to 5 LWR's or at multiple sites each with one or two LWR's provided the sites are spaced at least 10 miles apart. For liquid effluents, the design guide is 3 mrem/yr for the whole body and 10 mrem/yr to any organ from all pathways of exposure and all released radionuclides per reactor.⁽⁷⁾

3.2.5 Effluents and Control Measures for Reactor Operations

The total tritium inventory in the GA-DEMO and UWMAK-II facility represents about 10^8 Ci. Most of this tritium is recirculated through the plasma fuel and exhaust cycles. The development of adequate tritium control systems is necessary to maintain the escape of tritium from the fusion reactor at acceptable levels. The location of tritium and pathways for its escape are discussed below. This discussion is applicable for most fusion reactor designs.

The potential sources of tritium leakage during operation of a fusion facility are plasma fuel and recycle components, storage systems, first wall and blanket coolant streams and steam supply systems. Many of these systems contain large tritium and deuterium inventories and have the potential for release of tritium during an abnormal occurrence.

A three-level multiple barrier containment system described in TETR⁽⁸⁾ is representative of tritium control at a fusion facility. This system limits the dilution of tritium within each containment level so it can be recovered before it gets into the next level. This limits tritium releases to the environment. The multiple barrier concept is shown in Figure 3.5.

CAPS: CONTAINMENT ATMOSPHERE PURIFICATION SYSTEM

TERS: TRITIUM EFFLUENT REMOVAL SYSTEM

ETCS: EMERGENCY TRITIUM CONTAINMENT SYSTEM

FIGURE 3.5. Multiple Containment Barriers ⁽⁸⁾

The primary containment barrier consists of the components and pipes which contain tritium, tritium compounds, or tritium bearing materials. These components must demonstrate high integrity and ensure low tritium leakage. All connections are typically welded. Examples are storage containers, cryo-distillation columns, valves and interconnecting plumbing, etc. Materials containing lithium in the reactor blanket may also be required to exhibit high integrity for tritium containment.

Most of the above components will be enclosed in secondary containment such as gloveboxes, particularly equipment requiring periodic adjustment or maintenance. This system is designed to capture any tritium leaking from the primary containment. An inert cover gas would fill the glovebox, with an atmosphere purification system designed to remove any tritium detected. The detritiation process can be continuous, or only as required. The secondary containment between large components would consist of double-walled piping, again connected to the purification system to handle tritium leaks.

The detritiation mode used for secondary containment (continuous or batch) depends on the permissible tritium concentration in the glovebox atmosphere and the permeation rate through the weakest link in the containment. The most probable path for tritium permeation in the glovebox is through the gloves. The gloves are the thinnest barrier and have the highest permeability of any material used in the glovebox. Permeation of hydrogen has been experimentally measured for several rubber materials used to make gloves. The tritium permeation rate for tritiated water is 20 to 100 times faster through a particular kind of rubber than the permeation rate for dry molecular gas HT. The permeation of HTO is slowest through butyl rubber; therefore, this material should be used for fabrication of the gloves. To reduce the permeation of tritium in gloveboxes it is necessary to prevent the formation of HTO (formation of HTO occurs by chemical exchange with water and by direct oxidation). This can be accomplished by removing water vapor and oxygen from the glovebox by using the CAPS system.

For fusion fuel cycle applications, it appears that flush type gloveboxes will be adequate. Filled with nitrogen, the atmosphere would be purged in a batch mode on the detection of high oxygen levels. Oxygen can be held to an

acceptable level ($>19.$) with current technology. The flow from this purging cycle can be exhausted directly to the facility stack. Only on detection of tritium would the atmosphere in the secondary containment be routed to the CAPS system. This method is being used in the Tritium Systems Test Assembly,⁽⁹⁾ which will demonstrate tritium handling technology applied directly to the fusion fuel cycle.

From an environmental standpoint the dry glovebox atmosphere has reduced biological hazard potential because molecular tritium is not as readily absorbed by the body as tritiated water vapor. Building air monitors that distinguish between molecular tritium and tritiated water vapor will allow workers to remain in rooms where tritium levels exceed $5 \mu\text{Ci}/\text{m}^3$ if the HTO fraction is small.

The next level of containment is a Tritium Effluent Removal System (TERS) designed to handle failures of the primary containment. The release would still be confined to secondary containment; however, the release would swamp the CAPS system, which is designed only for routine tritium permeation and leakage.

Failure of the primary and secondary containment systems activates the Emergency Tritium Containment System (ETCS). The building inlet air ducts are sealed in the event of an off-normal tritium release. This reduces the risk of releasing contaminated building air to the environment. Depending on the amount of tritium released and the air flow rates within the reactor building, the ETCS could reduce the room tritium concentration to acceptable levels within a matter of days.

All of the above detritiation systems are currently based on the control technology developed at the national laboratories to eliminate tritium effluent releases. This technology is based on the catalytic conversion of all hydrogen isotopes to water with storage on molecular sieve beds. Detritiation factors of 10^6 across the system are being achieved. This method has proved highly effective in control of releases in laboratory use; however, building wide cleanup of 100 gram quantity releases has never been demonstrated. This is part of the function of TSTA.

The method of catalytic recombination to water has reduced tritium effluents at the laboratories dramatically; however, it introduces several problems. First, tritiated water is the more hazardous form of tritium from a biological exposure viewpoint. Other forms may eventually be more desirable. It is also difficult to separate the tritium from water, as large-scale isotopic separation will be required. The current practice is simply to dispose of the molecular sieve beds upon saturation. The molecular sieve may be an ideal storage media in the laboratory; however, its long-term performance for permanent storage has not been addressed fully.

Mound Laboratory is able to maintain $40 \mu\text{Ci}/\text{m}^3$ in a recirculating highly processed system and Los Alamos is maintaining an $0.5 \mu\text{Ci}/\text{m}^3$ in a work room with a $1 \text{ Ci}/\text{m}^3$ glovebox concentration.⁽⁹⁾ Workers wearing lightweight suits and face masks can be provided protection factors of 100 or more if changed hourly. The formation of HTO must be below 12% by volume of the containment if MPC occupational levels are not to be exceeded. HTO levels can be kept low if water vapor and T_2 gas is not allowed to build up. At $40 \mu\text{Ci}/\text{m}^3$ the HTO formation rate is about $10^{-6} \mu\text{Ci}/\text{m}^3\text{-hour}$. If the tritium contained in a cryopump (1 to 2 kg) is somehow released into containment, then $10 \text{ Ci}/\text{m}^3$ of HTO would be formed in the first hour.

The tritium concentration level for a given room or containment volume depends on the tritium leakage rate from the components and equipment as well as the air flow rate within the building. Using the values for tritium leakage given in Section 2 of this report, approximately 163 Ci/day of tritium makes its way into the containment building for a 1000 MWe fusion plant. Thus the volume rate (air flow) of air required to maintain the building atmosphere at $40 \mu\text{Ci}/\text{m}^3$ would be $1.5 \times 10^6 \text{ m}^3/\text{day}$. This would correspond to about 0.6 volume changes every hour. Approximately 11 Ci/day of tritium gas escapes to the environment. The quantity of tritiated water discharged to the environment is roughly 4 Ci/day.

The reactor containment must also protect workers outside the building from exposure to HTO or T_2O in excess of the maximum permissible concentration (MPC). The largest inventory of tritium, in the GA-DEMO for example, other than that entrained in storage, is seen to reside in the cryopumps (1 to

2 kg of T). The most conservative approach is to assume that work will continue outside the reactor hall area even following a release of the entire tritium inventory contained in the cryopump (about 2 kg). The MPC value of air for a controlled access area is 5 Ci/m^3 so work can proceed as long as the tritium concentration remains below this level. For the release of roughly 200 Ci to the environment after a 2 kg T dump, Galloway⁽¹¹⁾ calculated that less than 5 mrem dose would result.

The overall goal of environmental control technology at a fusion reactor is to reduce the total tritium loss to the environment to less than 10 Ci/day. For a reactor hall size of about 10^5 m^3 , similar to sizes used in reactor design studies, tritium losses to the environment can be minimized if the tritium concentration within the secondary containment is kept low (less than $50 \mu\text{Ci/m}^3$) and if the reactor hall wall surface were hermetically sealed. A stainless steel shell enclosing the entire reactor hall may provide the protection needed. An arrangement similar to this has been studied in an ORNL design study.⁽¹⁰⁾ The reactor building should meet the requirements for low release even if 10^6 to 10^7 Ci of T_2 are ignited and dumped.

In addition to the release of tritium into the containment or building hall, there will be losses of deuterium and helium during routine operations. The release of these elements is a problem of maintaining a working inventory rather than one of toxicity or human health hazards. Losses of deuterium into the containment building were estimated to be less than 4×10^{-3} grams/day. The D_2O losses are much less than 10^{-4} g/day. Helium losses can be quite high and are dependent on whether the blanket is cooled with helium or liquid lithium. A helium cooled blanket design is expected to lose about 1.2 kg He/day of which 220 g/day is lost out of the stack. Neither deuterium or helium losses at the reactor site will be assessed in terms of toxicity or emission standards.

3.3 HELIUM

Helium is used as a coolant in one of the blanket design options considered in this study. Helium production methods, potential effluents and effluent control measures are discussed below.

3.3.1 Helium Production Methods

Helium is separated from natural gas. A helium content of 0.3% is required for economic production of helium. The availability of helium for recovery depends on the rates of production and other conditions established in marketing natural gas.

Figure 3.6 shows a block diagram of the helium production process. Natural gas bearing helium is fed into the production facility at about 400 psi. The gas is pretreated to remove hydrocarbon condensate, carbon dioxide, and water before it enters the low-temperature equipment (cold boxes). In the cold boxes the gas stream is cooled to -250°F and liquefied except for the nitrogen. The liquefied natural gas is returned through heat exchangers in counterflow to the incoming stream to the cold box outlet. The gas is returned to the pipeline and goes on to fuel markets. Crude helium has a composition of 70% helium and 30% nitrogen. This helium is processed further to 99.995 mole% purity.

The current potential production capacity of helium in the U.S. is more than 4 Bcf/year from two Bureau of Mines plants and 10 industrial operators. Production of high purity helium totaled 0.775 Bcf in 1975.⁽⁴⁾ Helium present in the atmosphere, about 5 ppm, might be extracted from exhaust streams of liquid oxygen plants. This might be competitive with low-helium content (0.1%) extraction from natural gas.

3.3.2 Effluents from Helium Production

The separation of helium from natural gas does not routinely release any significant or harmful materials to the atmosphere or water table. Abnormal operating conditions also do not result in harmful pollutant release to the biota.

3.3.3 Helium Release Standards

Helium has been used in the current high-temperature, gas-cooled reactors and is being considered for use as the first wall and blanket coolant for the fusion reactor power plants of the future.

FIGURE 3.6. Helium Production Process

Helium is a noble gas and when present in high concentrations in air acts primarily as simple asphyxiants without other significant physiologic effects. A threshold limit value may not be recommended for each simple asphyxiant because the limiting factor is the available oxygen. The minimum oxygen content should be 18% by volume at STP. However, strange things happen when normal air is used in undersea work. Early on, oxygen was found to be toxic to divers under pressures of a few atmospheres. One toxic mechanism appears to be an impairment of the transport of carbon dioxide from tissues to the lungs under conditions of breathing oxygen at high pressure⁽¹²⁾. Inert gases became a favored base constituent for diving gases. Through experimentation, xenon and krypton were found to have strong narcotic properties.⁽¹³⁾ Helium

although slightly narcotic is currently widely used in combination with oxygen and nitrogen with up to 95% helium. These gases are breathed for many days at great depths during saturation diving.

A desired maximum ambient water concentration of 0.2 mg/l based on solubility of the gas in water at one atmosphere has been suggested.⁽⁴⁾ The rationale used is that the presence of gas in excess of this amount would constitute supersaturation and could have subsequent deleterious effects if consumed prior to establishment of equilibrium.

3.3.4 Control of Helium Releases

Helium can be produced, shipped, stored, and handled by distribution lines with negligible loss, provided good equipment and materials are used and reasonable care is taken. Helium may leak through minute cracks or openings and has a higher leakage rate than most other gases. Maximum use of welded, brazed, or soldered connections, choice of quality valves and pipes and frequent tests for leakage are required. Packless valves can be used to limit the loss of helium in the production lines. Valves having a raft seat are required when tight close-off is needed. These types of valves are not serviceable as metal-to-metal plugs where throttling from a higher pressure to a lower pressure is involved.

Compressor equipment must be carefully designed to avoid contamination of high purity helium. Oil-lubricated compressors require cleanup systems to remove hydrocarbon gas contamination in the helium.

3.4 LITHIUM

Lithium will be used extensively in fusion reactors as a breeding material and possibly as a liquid coolant.

Domestic production of lithium is divided between open-pit pegmatite mines in the spodumene belt of North Carolina, and the concentration of subsurface brine deposits in Nevada and California. Precise production data are considered privileged information by the producing firms and are not available. However, the Bureau of Mines has estimated total domestic production in 1977

to be at 4,000 short tons.⁽¹⁾ This amount has satisfied all domestic needs and more than three-fourths of total world demand.

The North Carolina pegmatite belt is over 25 miles long, up to one-half mile thick and roughly parallels the inner Piedmont Mountain range. Large pegmatite dikes are enclosed in metasedimentary wall rock. Spodumene is the principal ore mineral. All ore is removed by open-pit methods at two locations, one operated by The Foote Mineral Company at Kings Mountain and the other by the Lithium Corporation of America at Bessemer City. The lithium ore is concentrated by a frothing and flotation process.

The Foote Mineral Company also extracts lithium from subsurface brine deposits at Clayton Valley, Nevada. Here the brines are concentrated by solar evaporation and precipitated with sodium carbonate.

Lithium is also produced from brines at Searles Lake, California. At this location the Kerr-McGee Corporation recovers lithium as a coproduct of potash, soda ash, borax, and salt cake. The beneficiation process includes solar evaporation, froth flotation, and precipitation with sodium carbonate.

Currently, two chemical companies are recovering magnesium from brines of the Great Salt Lake, Utah. Because lithium closely follows magnesium in the beneficiation process, these operations have the potential to also recover lithium.

Production methods at these locations are outlined in the following sections. The processes for refining lithium metal and enriching the metal in ^6Li are also described. Effluents from these processes are identified, and compared to present effluent standards.

A complete description of lithium resource assessment is included in Appendix A. Also included in the appendix are proximate environmental factors as they relate to each mineralized area. In addition, complete descriptions of minimizing methods and mining effluents produced at each area are included in Appendix A.

3.4.1 Lithium Production Methods

As mentioned above, pegmatite ores are mined using open pit techniques. The ores are then processed using either a sulfuric acid roast or limestone

calcination to generate lithium carbonate or lithium hydroxide monohydrate. Subsurface brines from the Searles Lake and Clayton Valley playas are extracted through wells. The brines are evaporated either partially or totally by solar methods and then converted to lithium carbonate. The processing methods for pegmatite ores and brines are described in the following paragraphs.

Extraction of Lithium from North Carolina Pegmatites

Either the sulfuric acid roast process or the limestone calcination process are used to extract lithium from North Carolina pegmatites. Both processes consist of three basic steps:

- milling to recover β -spodumene
- conversion of α -spodumene to β -spodumene; and
- extraction and purification of lithium.

The sulfuric acid and the limestone processes are outlined below.

Sulfuric Acid Roast Process. The sulfuric acid roast process is used by the Foote Mineral Company at King's Mountain, North Carolina and by the Lithium Corporation of America at Bessemer City, North Carolina. A flow sheet describing this process is shown in Figure 3.7. The following process descriptions are representative of current methods.

Ore from the mine contains a mixture of quartz, mica and feldspar in addition to spodumene. The ore is fed to progressive crushers and then to a ball mill which reduces the ore to fine, sand-like particles. The ore is conditioned with a frothing agent (probably oleic acid) and is fed to flotation units. Here, α -spodumene is recovered as the froth product. The tails are subjected to further flotation steps to recover feldspathic sand and mica for sale to the glass industry and other markets.

Since natural α -spodumene is essentially unattached by sulfuric acid, the first step in processing is the conversion of the ore to the more reactive β -spodumene. This is accomplished by heating the α -spodumene to about 1100° in a rotary kiln. Susceptibility to chemical reaction is further enhanced by ball-milling the kiln discharge to a very fine powder, thus increasing the surface area.

FIGURE 3.7. Sulfuric Acid Roast Process

The powdered β -spodumene is mixed with sulfuric acid (93X) and is introduced into an acid-roasting kiln. The mixture is heated in the kiln to approximately 250°. The chemical reaction in which hydrogen ions replace the lithium ions in the mineral and form lithium sulfate is:

The acid-roasted ore is leached with water and the excess acid is neutralized with ground limestone. The slurry is then pumped to a series of rotating vacuum filters. The solids removed by the filters are stockpiled for sale to brick and ceramic industries. The filtrate is treated with hydrated lime and soda ash to remove calcium and magnesium impurities. After filtration, the solution is concentrated by evaporation to about 200-250 g/liter of lithium sulfate.⁽⁸⁾ At this point, soda ash is added to precipitate lithium carbonate. Centrifuging separates the precipitated lithium carbonate from the mother liquor, which contains sodium sulfate, excess soda ash and some dissolved lithium carbonate. The mother liquor is pumped to another tank and cooled to 35°F. Glauber's salt (hydrated sodium sulfate) crystallizes at this temperature and is ultimately sold in an anhydrous form. Sodium, lithium and carbonate ions remaining in solution are recycled to the purification step so that no lithium is lost.

Limestone Calcination Process. A flowsheet describing the limestone calcination process used by the Foote Mineral Company plant at Sunbright, Virginia is shown in Figure 3.8.

The pegmatite ore fed to the mill consists of approximately 20% spodumene, 40% feldspar, 30% quartz, 7% mica and 3% miscellaneous materials. The ore is crushed and wet ground in a rod mill and then fed to flotation cells, which use a fatty acid (probably oleic acid) as a frothing agent. The ore is concentrated from about 1.5% as Li_2O to 6.2% as Li_2 in the frothing product. The tailings from the spodumene flotation cells are fed to other flotation circuits to produce mica, feldspar and a quartz-feldspar mixture. These materials are marketed.

FIGURE 3.8. Limestone Calcination Process

Spodumene concentrate from the flotation cells is mixed with finely ground limestone in the ratio 1 part ore to 3.5 parts limestone.⁽⁵⁾ The slurry is fed to a rotary, coal-fired kiln where water is evaporated and spodumene changes from the α to the β form. The β -spodumene reacts with calcined limestone to form calcium silicate and other compounds according to the following reaction:

A one-inch clinker is discharged from the kiln at 1900°F.

After cooling, the clinker is wet-ground in a ball mill to about 100 mesh. The ground clinker is hot-leached with water in a six-stage thickener and filtered. The solids from filtration contain less than 0.10 to 0.15% lithium oxide and can be used as a binding material in construction. The filtrate, an impure lithium hydroxide, is concentrated in a triple-effect evaporator-crystallizer. The crystalline lithium hydroxide monohydrate formed is separated from the mother liquor by centrifugation. Additional lithium hydroxide is recovered from further processing of the mother liquor. These products are presumably further purified by recrystallization. The final mother liquor from the impure crystals is a 50% solution of sodium hydroxide; this is sold as a by-product.

Extraction of Lithium from Clayton Valley Brines.

Mineral brines at Clayton Valley are extracted from a system of wells drilled into the playa. The wells, 24 in. in diameter, are drilled on 2,000-ft centers and vary in depth from 300 to 800 feet. Brine is then piped to a series of 10 evaporation ponds that vary from 760 to 13 acres in size and range from 1 to 4 ft deep.

Solar evaporation results in a 20-fold increase in lithium concentration, from 300 ppm to 6,000 ppm. In addition to concentrating lithium, evaporation causes the precipitation of unsalable salts. Sodium chloride is deposited in all ponds after the lithium concentration reaches 680 ppm. Lime is added in the intermediate stages of the pond system to precipitate calcium, magnesium

and sulfate. As the brine reaches a lithium concentration of 2,000 ppm, glaserite ($K_3Na(SO_4)_2$) and potash (KCl) are precipitated.

Brine reaches the final pond in the system at a concentration of 6,000 ppm lithium. This rich brine is then sent to the milling plant in Silver Peak to be converted to salable lithium carbonate.

Officials of the Foote Mineral Company indicate that unsalable salt sediments are not removed from larger ponds in which the brine has a lithium concentration less than 2,000 ppm. When these ponds become full of precipitate, they are bypassed and abandoned. However, precipitated salts are removed from the smaller rich-brine ponds and stockpiled for possible recovery of potassium chloride. Company sources estimate the rate of deposition in these smaller ponds to be approximately 9 in. per year.

Lithium Carbonate Production from Clayton Valley Brines. Foote Mineral Company produces lithium carbonate from Clayton Valley brines at Silver Peak, Nevada. The production process consists of three basic steps:

- solar evaporations
- precipitation
- lithium chemical production.

A flowsheet describing the process is shown in Figure 3.9. The process and the effluents associated with the production of lithium carbonate are attained below.

The sodic (NaCl) brines of Clayton Valley are first concentrated by a factor of 12 by solar evaporation in ponds, to a final concentration of over 0.6% LiCl. Approximately 10,000,000 gallons of brine a day are pumped from the playa. The concentration of the well brines is slightly above 300 ppm Li. As the brine concentrates in the solar evaporation ponds, sodium chloride and other salts are deposited on the pond floors.

Lime is added midway through the pond system in order to precipitate magnesium. Eventually the brine reaches a concentration of 6,000 to 7,000 ppm Li.

FIGURE 3.9. Lithium Carbonate Production from Clayton Valley Brines

Salts deposited in ponds with concentrations of less than 2,000 ppm are not harvested, but salts deposited in the strong brine ponds are removed in order to continue utilizing these more expensive and smaller ponds. Approximately nine inches of salt build up each year. The salt removed from the ponds is stockpiled and has no commercial value at this time.

Brine fed to the plant at 6,000 to 7,000 ppm Li is further purified by adding small quantities of soda ash and lime. The impurities are then filtered out and the brine is treated with soda ash to precipitate lithium carbonate. The lithium carbonate is filtered out and dried. The mother liquor produced in the filtering process is recirculated to the ponds.

Extracting Lithium from Searles Lake Brines

Mineral brines at Searles Lake are extracted from a system of wells drilled into the playa. Currently there are approximately 200 wells supplying brine to the processing plant. Because brine is selectively produced from the upper and lower salt layers, well depths are generally either 70 ft or 130 ft. The brine is lifted with a surface pump at each well and collected in a pipeline network for transportation to processing facilities in Trona and Westend.

A major portion of the primary brine is used as direct feedstock to the processing plant where borax, potash, soda ash and salt cake are extracted. However, some of the brine goes through intermediate concentration in solar evaporating ponds before reaching the plant.

In addition to mineral brines, brackish water used in the process stream enters the plant from wells north of Searles Lake. Also, small amounts of potable water are imported from Indian Wells Valley to be used for select process applications that require purity.

The solution flow is somewhat circular in that the mining and milling process includes a recharge phase. That is, end liquors and process water from the milling plant are released to the playa to prevent feedstock depletion.

Lithium Extraction from Searles Lake Brines. Dilithium sodium phosphate (Li_2NaPO_4) is one of the minor coproducts of the Searles Lake Trona plant, which produces potash, borax, salt cake and soda ash as major products.

Lithium recovery is accomplished as a two-stage foam-flotation process. A flow sheet of this process is shown in Figure 3.10. Lithium extraction takes place in three major steps:

- lithium saponification in the main cycle evaporators;
- lithium separation from a sodium sulfate process stream by flotation;
- dilithium sodium phosphate purification.

This process yields a product which consists of about 10% Li. The extraction of Li is reported to be 90%.⁽¹⁶⁾ The product contains 50% P_2O_5 , 17% Na_2CO_3 , 2% Na_2SO_4 , and 2% H_2O .

The lithium content of the raw brine fed to the main plant evaporator cycle is 0.008% Li. Heating and concentrating the brine causes lithium to precipitate mainly in the first effect evaporator as dilithium sodium phosphate. A fatty acid derivative of coconut oil is added to the main evaporators to control foaming. The fatty acid saponifies and the soap formed, together with the subsieve colloidal lithium, becomes occluded by the burkeite ($Na_2CO_3 \times Na_2SO_4$) precipitate.

Flotation is carried out in two stages, with air added at both feed pumps. Release of air in the flotation units results in the formation of tiny air bubbles, which collect the soap-coated Li_2NaPO_4 particles and float them to the surface. The lithium froth overflows to the second stage and the liquor discharged from the bottom and is routed to the salt cake plant. The second stage consists of a pneumatic foam separator, which further separates the foam from the liquor.

After flotation separation, the foam is agitated and heated to break down the foam. The lithium salt is then filtered batchwise and the cake is washed. Conveyor belts carry the cake to the roaster where organics are removed by roasting and bleaching with sodium nitrate in a gas-fired Herreshoff furnace. The dried concentrate is packed in 100-lb bags for shipment.

Lithium Carbonate Production at Searles Lake. In the lithium carbonate process, dilithium sodium phosphate is converted to a refined lithium carbonate (Li_2CO_3) product and a 78% crude orthophosphoric acid (H_3PO_4).

FIGURE 3.10. Lithium Extraction from Searles Lake Brines

The lithium carbonate production process is shown in Figure 3.11. This process consists of three major chemical processes:

- Conversion of dilithium sodium phosphate to lithium sulfate (Li_2SO_4)
- Conversion of lithium sulfate to lithium carbonate
- Recovery of aqueous lithium carbonate from end liquor.

Annual production is approximately 153,000 kg as lithium or 900 tons/year of lithium carbonates.⁽¹⁷⁾ This represents a lithium recovery of 88% from dilithium sodium phosphate and an approximate recovery of one-third of the lithium in the Searles Lake brine. Lithium losses are confined to that dissolved in concentrated phosphoric acid and in the treated end liquor, which is recycled to the head end of the process.⁽¹⁶⁾ Essentially no lithium is returned to the lake.

In the lithium carbonate process, the Li_2NaPO_4 , which was extracted from the brine, is first digested with 93% sulfuric acid in an agitated tank.

The lithium and sodium sulfates are centrifuged from the digester tank, washed and dissolved in water. The filtrate, a 45 to 50% orthophosphoric acid solution, is evaporated to a 78% phosphoric acid solution containing approximately 1% lithium. After evaporation, Li_2SO_4 and Na_2SO_4 solids are settled out and recycled back to the digester.

The dissolved Li_2SO_4 and Na_2SO_4 are added to a saturated solution of sodium carbonate at 200°F. Lithium carbonate precipitates out of solution according to the following reaction:

The lithium carbonate solids are centrifuged out, washed, and dried in a steam-heated rotary-drum air dryer. The product is packaged in 50-lb sacks and 325-lb drums for shipping.

FIGURE 3.11. Searles Lake Lithium Carbonate Production

Lithium recovery is increased by treating the end liquor (filtrate from lithium carbonate centrifuge) using dilute phosphoric acid and then evaporating to near Na_2SO_4 saturation. The phosphate ions cause precipitation of Li_3PO_4 according to the following reaction:

The Li_3PO_4 is settled out and recycled to the acid digester.

Lithium Recovery from Great Salt Brines

Currently, no lithium is being produced from Great Salt Lake brines. However, process facilities exist that have the potential to extract this lithium.

In 1970, the Lithium Corporation of America and Sulzedetfurth A. G. of West Germany completed construction of a brine evaporation and processing complex near the north arm of the Great Salt Lake. The solution mining phase of this complex consists of 20 square miles of ponds in which salts precipitate as the brine is concentrated by solar evaporation. Industry officials report a brine volume of 20,000 acre-feet per year enters the pond system. A period of 2 years is required for the brine to completely pass through the ponds. During this time, approximately 95% of the liquid will be evaporated and 20 million tons of salts will be precipitated.

Initial plans called for harvesting solid salts of potassium, sodium and magnesium sulphate from pond sediment as well as extracting magnesium, lithium and bromine from ion-rich residual bitterns. Because the Dow Chemical Company canceled its long-term contract to purchase magnesium chloride, construction on the bitterns processing plant was stopped just short of completion. Under current market conditions it is not feasible to process the residual brine just for the recovery of one ion such as lithium.

Although no lithium is currently being produced from Great Salt Lake brines, processing facilities for this purpose exist near Little Mountain, Utah. Potassium sulfate and sodium sulfate are produced as major products in an operation similar to the process at Searles Lake. Lithium, if produced,

would be a byproduct of these processes. Information on the lithium extraction process is not available. The current brine processing method is shown in Figure 3.12.

Potassium and sodium compounds precipitate in the main solar evaporation ponds. These salts are ultimately fed to nearby plants, operated by Great Salt Lake Minerals and Chemical Corporation (GSL), for purification and refining. The bitterns remaining after precipitation of these salts are rich in magnesium chloride compounds precipitate and become the feed for the production of electrolytic magnesium metal and chlorine. After the magnesium compounds have precipitated, the remaining end-stage solution is enriched in both lithium and bromine. Recovery of lithium from this solution would probably

FIGURE 3.12. Lithium Recovery from Great Salt Lake Brines

be similar to the Clayton Valley process. It is estimated that compounds containing up to 500 tons per year of lithium can be recovered.⁽¹⁸⁾

3.4.2 Lithium Metal Production

Both lithium carbonate and lithium hydroxide monohydrate can be used as the basic lithia source for the production of lithium metal; the choice depends on the purity of the metal required. Lithium hydroxide monohydrate is usually used for higher purity grades.

Lithium carbonate or lithium hydroxide monohydrate is first converted to lithium chloride by the addition of hydrochloric acid, with purification and crystallization steps following. The chloride conversion reactions that occur are:

The lithium metal cell is constructed of a low-carbon steel shell, which acts as a cathode, a container of fused salt, and graphite rod anodes. The cell is initially charged with a mixture of 55% lithium chloride and 45% potassium chloride. The electrolyte mixture melts at about 400°C. Auxiliary heaters are used to initiate the melting process to start electrical conduction. Once conduction begins, the heat generated by the internal resistance of the cell is sufficient to sustain the melt. Lithium reduced at the steel cathode floats to the top of the cell where it is skimmed off the electrolyte and poured into steel ingot molds. Five kilograms of chlorine gas are generated at the anode for every kilogram of lithium metal produced. The chlorine gas is collected and sold commercially. The reactions that occur are:

The electrolysis process has a high consumption of electrical energy with a typical use of about 46 kWh for each kilogram of lithium metal. This does not include the startup power needed to melt the electrolyte.

3.4.3 Lithium Isotopic Enrichment

Some reactor designs require that the tritium breeding material be enriched (up to 90%) in lithium-6 as opposed to natural lithium (7.5% lithium-6). There are currently no governmental or private facilities for lithium-6 separation. A facility at Oak Ridge, Tennessee last operated in 1963. Isotopic separation of lithium was achieved using an amalgam/aqueous hydroxide technique called the Colex process.⁽⁴⁾

In the Colex process the enrichment of lithium-6 was achieved by counter-current flow of lithium-mercury amalgam and aqueous lithium hydroxide solution in packed columns. Lithium-6 preferentially moves into the amalgam and lithium-7 into the aqueous phase. The lithium amalgam was produced by electrolysis of lithium hydroxide and mercury. Lithium amalgam is pumped from the lower portions of the separation column into the upper portions of the other columns arranged in cascade. The amalgam is then allowed to flow downward countercurrent to the liquid lithium hydroxide phase which was pumped upwards.

Lithium-6 in the desired isotopic concentration can be withdrawn as the hydroxide once the amalgam decomposes from water contact in catalysts. The lithium-6 hydroxide not extracted as product is returned to the top for recycle, while the mercury is recycled to the amalgam cells.

At the tail (lithium-7) end of the Colex plant, some of the lithium hydroxide depleted in lithium-6 is withdrawn for disposal (storage, stockpile, or resale) while the remainder is returned to the amalgam cells.

Enriched lithium-6 is stored as the hydroxide monohydrate. Lithium metal is produced electrolytically and packaged in aluminum cans under an argon atmosphere. Metal preparation involves neutralizing the hydroxide with hydrochloric acid, drying, and electrolyzing lithium chloride and potassium chloride mixture.

Although any enrichment could be obtained, the product of the plants under the operating parameters at that time was 95.5 wt% lithium-6.⁽⁴⁾ No effort is being made to maintain these facilities. Modification to the plant would be necessary to meet the current environmental and safety guidelines. The handling of large quantities of mercury needed to operate this plant would pose occupational as well as public environmental hazards.

Another process which might be used to enrich lithium-6 is called the crown chemical exchange process. The possible advantage to using this process is that no mercury would be required. Isotopic separation of calcium based upon the chemical exchange reaction between calcium salts and macrocyclic polyethers, a new class of metal complexing agents, is also applicable to the isotopic enrichment of lithium-7 or lithium-6. Lithium-7 tends to concentrate in the aqueous phase and lithium-6 in the organic phase of polyether. Counter-current liquid-liquid extraction columns can be used to achieve multistage enrichments.

Another lithium isotope enrichment process is based primarily on the calcium isotope enrichment work and preliminary laboratory-scale experiments with lithium.⁽⁴⁾ Experiments have shown equilibrium separation factors of 1.01 with the first few polyethers chosen. In order for the process to be economically competitive, it is estimated that a value greater than 1.03 will be needed.

Figure 3.13 shows a schematic of the chemical exchange cascade for separation of lithium isotopes. The process consists of a series column with an overhead reflux system and a bottom reflux system. The facility could be used for either lithium-6 or lithium-7 enrichment. Lithium bromide is used as the feed material, polyether crown as the complexing agent, and bromoform as the organic solvent. This process has potential because none of the constituent materials is flammable.

The lithium isotope separation takes place in the series column train when an aqueous solution of lithium bromide contacts a countercurrent organic solution of lithium bromide complex in bromoform/crown solution. The aqueous LiBr solution tends to become enriched in the heavy lithium-7 isotope while the organic LiBr solution tends to become enriched in lithium-6.

FIGURE 3.13. Lithium Isotope Production Process

3.4.4 Lithium Release Standards

The inherent toxicity of the lithium ion is high, a few milli-equivalents in the plasma gives rise to serious neurologic signs and symptoms. These include anoxeria, nausea, tremor, muscle twitches, apathy, mental confusion, blurring of vision, coma and death.

Lithium may be released to the environment as Li, Li₂O or LiOH as a result of a lithium fire. A lethal chemical dose can result from a 1-hour exposure to 200 µg/m³ LiOH.(19)

Industrial experience has shown that a reduction in atmospheric concentration of lithium hydride to $25 \mu\text{g}/\text{m}^3$ is required to attain just a sneezing level. This concentration was adopted by the Occupational Safety and Health Administration for a time-weighted average.⁽²⁰⁾

McKee and Wolf recommended that lithium in water used for drinking and cooking purposes should not exceed $5 \text{ mg}/\text{l}$.⁽²¹⁾ No supporting documentation was offered. Doudoroff and Katz reviewed the literature on the toxicity of industrial wastes to fish.⁽²²⁾ They reported on earlier work which found that lithium chloride at concentrations between 320 and 620 mg/l as Li, can kill freshwater fish in about 24 hours. Goldfish are killed rapidly when kept in water containing 320 to 400 mg/l Li. Strand and Poston reported that 100 mg/l of LiCl is toxic, but 33 mg/l is harmless to fish.⁽²³⁾ There is quite a lot of variability in these data. It is assumed that "freshwater fish" are composed of more than one species and that experimental conditions of these workers probably varied. However, with these problem areas in mind, the mean of these data is about 310 mg/l of Li.

Historically, application of factors have been attached to toxicological data in an attempt to predict safe concentrations in surface waters. Factors of either 0.01 or 0.1 were used. If the toxicant was considered to have long-term effects such as carcinogenicity, the value of 0.01 was used; if it was a simple toxicant, 0.1 was multiplied by the LC50 dose to arrive at the safe limit.

If it is assumed that 310 mg/l would be a safe level as perhaps supported by the quote from Meinck, et al (1956) as 33 mg/l being harmless.

Lithium hydroxide used now in industry, became a test compound in the mid fifties. An acute inhalation toxicity study was performed on animals using an atmospheric concentration of between 5 and 55 mg/m^3 lithium hydroxide.⁽²⁴⁾ This material proved to be intensely irritating and corrosive. Levels in excess of $10 \text{ mg}/\text{m}^3$ eroded the body fur and skin on the legs of animals and produced severe inflammation and irritation of the eyes on occasion, as well as destruction of the external nasal septum in some rats. All levels resulted in emphysematous changes in the lungs. The toxicology of a few lithium compounds is presented in Table 3.2.

TABLE 3.2 Lithium Toxicity in Animals(20, 36)

<u>Compound</u>	<u>Animal</u>	<u>Toxicity</u>	<u>Concentration (mg/kg)</u>	<u>Route/ Administration</u>	<u>Comments</u>
Lithium Carbonate	human	TDL ₀ ^(a)	7	orally	
	rat	LDL ₀ ^(b)	710	orally	
	rat	TDL ₀	300	orally	6-15d preg. effect-tera- togenic
	dog	LD50 ^(c)	500	orally	
Lithium Chloride	rat	TDL ₀	190	intraperitoneal	9-16 preg. effect-tera- togenic
	rat	LD50	757	orally	
	rat	TDL ₀	1062	parenteral	pregnant
	mouse	LD50	480	subcutaneous	
	mouse	LD50	604	intraperitoneal	
	rabbit	LD90 ^(d)	850	orally	
Lithium Hydroxide	rat	LCL ₀ ^(e)	22 mg/m ³	-4 hrs. inhal- ation	

- a. lowest published toxic dose
- b. largest published lethal dose
- c. lethal dose 50% kill
- d. lethal dose 90% kill
- e. lowest published lethal concentration

3.4.5 Effluents from Lithium Production

Effluents produced in the mining and milling of North Carolina pegmatite ores and in the extraction and purification of brines from Searles Lake,

Clayton Valley and the Great Salt Lake are discussed below. Effluents generated during the production of lithium metal from lithium carbonate or lithium hydroxide monohydrate are also summarized.

Effluents from Processing Pegmatite Ores

Effluents generated in the production of lithium compounds from pegmatite ores include mine water, solid waste from overburden and gangue material, fugitive dust from mining and milling operations, and carbon dioxide, residual ore, and neutral aqueous solutions from milling and chemical refining processes.

At both mine locations all solid and aqueous wastes are utilized. Mine water is cycled to milling facilities for use as process water, overburden is used in road and tailings dam construction and gangue is sold to aggregate producers.

Fugitive dust from mining and milling operations is controlled with water sprays and bag filters. It is estimated that 0.65 lb of fugitive dust are released per ton of ore.

Mica and feldspathic sand, wastes from the milling process, are sold to glass producers and some of the solids from chemical refining are sold to brick and ceramic producers. Residual ore and tailings water are collected in ponds. The residual ore settles out and the clarified, neutralized water is allowed to seep into nearby streams. The ponds will eventually be landfilled and seeded. About 0.95 ton of carbon dioxide is generated per ton of ore processed and is diluted through stack dispersion.

Effluents from Processing Brines

Effluents from the lithium extraction processes at Searles Lake, Clayton Valley and the Great Salt Lake consist of dilute aqueous brines, precipitated salts, and various quantities of airborne particulates and stack gases. In all three cases, the aqueous effluents are recycled to the lakes. This eliminates the problem of liquid waste disposal and maintains the brine salinity of the lake. At Clayton Valley and the Great Salt Lake, unsalable precipitated salts are allowed to accumulate in the smaller evaporation ponds, which are abandoned when filled. At Searles Lake, the quantity of precipitated salts is negligible. Accumulation rates of solid wastes are given in Table 3.3.

TABLE 3.3. Accumulation Rate of Solid Waste from Brines

<u>Location</u>	<u>Accumulation Rate (T/Day)</u>
Clayton Valley	6,000
Great Salt Lake	20,000
Searles Lake	--

Generally, hydrogen sulfide gas (H_2S) is associated with brine deposits in the western U.S. If present in significant amounts, H_2S would be released to the atmosphere during the evaporation process. Currently, H_2S emissions vary at each site depending on processing equipment used. At Clayton Valley particulate emissions are about 0.17 MT/yr and at Searles Lake particulate emissions attributable to lithium operations are about 5 T/yr.

Effluents from Lithium Metal Production

Effluents associated with lithium metal production from lithium carbonate are carbon dioxide and chlorine gas. About 0.5 tons of carbon dioxide are generated for every ton of lithium chloride produced. The chlorine gas, which is generated when lithium is produced from lithium chloride, is controlled by a chlorine gas-collection system.

Effluents from Lithium Isotopic Separation

The main waste associated with lithium-6 enrichment is lithium tails. These tails are approximately 98-99% lithium-7. It is possible that the lithium-7 tails could be used as tritium breeding material in other fusion reactors (designs not requiring lithium-6 enrichment). Lithium-7 tails could also be used in pH adjustment in PWR coolant. The use of impure lithium would result, however, in an increased production of tritium in PWR coolant streams.

The most hazardous effluent associated with lithium enrichment will be mercury-contaminated wastes. Adequate facility descriptions are required before estimates of the quantity of mercury-contaminated wastes can be made.

3.5 BERYLLIUM

Beryllium will be used with the solid lithium compound blanket to provide neutron multiplication. Most of the present requirement for beryllium is met by importing foreign beryl (a mineral containing beryllium). Large amounts of beryl are contained in the North Carolina pegmatite belt. However, because of low grade and small particle size, this beryl is not economically recoverable with the present state-of-the-art technology.⁽²⁵⁾ Well-zoned pegmatite deposits in New England, South Dakota, and Colorado are known to contain about 10,000 tons of large, readily recoverable beryl crystals. However, these deposits are also not classified as reserves because the high cost of hand sorting prohibits recovery. There are also significant domestic reserves and resources in the low-grade nonpegmatite deposits of western Utah and the Seward Peninsula of Alaska.

The only significant domestic production of beryllium in the U.S. is from deposits at Spor Mountain in west central Utah. Bertrandite is the principal ore mineral in these deposits, which occur in localized units of water-laid volcanic tuff. Precise dimensions of the units are lacking; however, it is known that they vary from 2 to 20 ft thick and some units are at least 200 ft long. The U.S. Bureau of Mines places proved reserves within the deposits at 28,000 short tons of contained beryllium metal.⁽²⁶⁾

Domestic beryllium deposits also occur in economically significant quantities in the central York Mountains, Seward Peninsula, Alaska. Here a suite of five beryllium minerals was formed by hydrothermal alteration where dikes cut marine carbonate rocks. Proved reserves of beryllium in the York Mountain deposits are at least 4,500 short tons of contained metal.⁽²⁷⁾ Beryllium is not now being produced from these deposits.

Although data is lacking on total domestic resources of beryllium, the Bureau of Mines estimates that proved reserves plus subeconomic resources would yield at least 80,000 tons of beryllium metal.⁽²⁶⁾ Deposits at Spor Mountain and the Seward Peninsula have been estimated to each contain approximately 35,000 tons of this metal. The remaining 10,000 tons are found in small low grade nonpegmatite deposits at various locations in the U.S. The mining of

beryllium ore at Spor Mountain and production of metal are described in the following subsections. Complete descriptions of beryllium ore resources and mining methods are given in Appendix B. In addition, environmental factors associated with each other as well as mine effluents produced are presented in Appendix B.

3.5.1 Beryllium Mining

Beryllium ore at Spor Mountain is extracted by a unique system of surface mining. Pit areas are developed to supply mill feedstock over a specific time horizon. At the end of this time, a new pit is developed and exploited.

Currently, all deposits in the area are being developed by the Brush Wellman Company on a 5-year planning horizon. More specifically, a drilling program is used to define the boundaries of a 5-year ore supply. Once the ore body is identified, all overburden is stripped and detailed drilling programs are conducted to define ore grade. Next, the ore is selectively removed and blended in stockpiles. This blending step is necessary to ensure a homogeneous feedstock of 0.6 to 0.7% BeO required by the concentrating plant.

Under current mining procedures, only 40 days/year are devoted to actual ore removal. The remaining time is spent blending ore as well as drilling and stripping new pit areas.

Officials of the Brush Wellman Company indicate near future plans do not include underground mining. However, they do not exclude underground mining as an alternative at some time in the distant future.

3.5.2 Beryllium Metal Production

Production of beryllium metal consists of two basic steps: (1) extraction of beryllium oxide from bertrandite ore, and (2) conversion of beryllium hydroxide to the metal. These steps are described in the following paragraphs.

The bertrandite ore is wet-ground and screened at the Spor Mountain mine to form a slurry that is transported to the processing plant. Here, it is leached with a 10% sulfuric acid solution. The insoluble ore residue is

filtered and discarded. The remaining beryllium sulfate solution is treated by a solvent-extraction process using di-2-ethylhexyl phosphoric acid in kerosene. The beryllium in solution is subsequently converted to beryllium hydroxide for further processing to the oxide or the metal. A flow sheet describing the process is shown in Figure 3.14.

The reduction of beryllium hydroxide is normally carried out in two major steps: (1) conversion of beryllium hydroxide to beryllium fluoride, and (2) conversion of beryllium fluoride to beryllium metal. Figure 3.15 shows the steps of the magnesium reduction process as used by the Brush Wellman Company.

FIGURE 3.14. Extraction of Beryllium Oxide from Bertrandite

FIGURE 3.15. Reduction of Be(OH)_2 to Be Metal

As seen in the figure, beryllium hydroxide ($\text{Be}(\text{OH})_2$) is first dissolved in ammonium bifluoride solution. The following reaction occurs:

Calcium carbonate is then added to precipitate aluminum as the hydroxide, while PbO_2 is added to oxidize manganese and chromium, which are precipitated as MnO_2 and PbCrO_3 . After filtering, ammonium polysulfide is added to precipitate lead, nickel and copper as sulfides. After another filtration, the solution is evaporated and the remaining ammonium beryllium fluoride decomposes at 900° to 1100°C in an induction-heated graphite retort, according to the following reaction:

Beryllium fluoride flows continuously from the retort onto a casting wheel, where it cools and forms into pellets.

Pelletized BeF_2 is charged to a graphite crucible and a stoichiometrically deficient quantity of magnesium metal is added. The mixture is heated slowly and the following reaction occurs:

The excess beryllium fluoride acts as a flux that allows the reaction to occur at a lower temperature. The beryllium floats on the molten slag permitting separation after solidification. The cooled beryllium metal is milled under water to remove the slag. The beryllium metal product at this point is in crude form and further purification must be accomplished.

The preferred purification method for commercial processes is vacuum melting.⁽²⁸⁾ Usually the melting is performed in a tilt-pour furnace housed in a vacuum chamber. Beryllium is melted in a beryllia or a lined alumina crucible. The molten metal is poured into a graphite mold to form a billet of pure beryllium. It is usually possible to produce beryllium billets weighing 75 to 90 pounds by this method within a 4-hr furnace cycle.⁽²⁸⁾

3.5.3 Beryllium Release Standards

Early animal studies indicated that beryllium was not a highly toxic element; however workers engaged in the extraction of the element from its ores suffered from a number of ailments, including dermatitis, tracheobronchitis and pneumonitis, which was generally fatal. Most of these workers were exposed to soluble beryllium salts, but some cases of acute pneumonitis resulted from handling beryllium oxide.⁽²⁹⁾

Later on, a severe chronic lung disease developed among employees of fluorescent lamp plants. A fine beryllium oxide was a component of fluorescent powders. Additional cases of berylliosis have since been reported from various industries where dusts or fumes of metallic beryllium or its compounds were present. Workers have developed pneumonitis where beryllium concentrations in excess of 1 mg Be/m^3 in soluble compounds.⁽³⁰⁾

Subchronic exposure of animals to beryllium sulfate in concentrations of $40 \text{ } \mu\text{g Be/m}^3$ resulted in pneumonitis, as did exposure to finely divided beryllium oxide in concentrations of 10 mg/m^3 .⁽³¹⁾

The EPA set an emission standard to the atmosphere from stationary sources of $<10 \text{ g/24-hour period}$ or an ambient concentration limit for public exposure of $0.01 \text{ } \mu\text{g/m}^3$ average over a 30-day period.⁽³²⁾

The occupational threshold limit adopted by the U. S. Atomic Energy Commission in 1949 of $2 \text{ } \mu\text{g Be/m}^3$. The threshold limit value (TLV) suggested by the ACGIH (1976) is also $2 \text{ } \mu\text{g/m}^3$ for an 8-hour exposure.⁽³³⁾ Whereas NIOSH (1977) recommends $0.5 \text{ } \mu\text{g/m}^3$ for a 130-minute exposure.⁽²⁰⁾

3.5.3.1 Beryllium As A Water Pollutant. Beryllium could enter surface waters in effluents from certain metallurgical plants. Although the chloride and nitrate salts of beryllium are very water soluble, and the sulfate is moderately so, the carbonate and hydroxide are almost insoluble in cold water.⁽⁷⁾ McKee and Wolf reported that beryllium is not likely to achieve significantly toxic levels in natural ambient waters.⁽²¹⁾ Out of more than 1500 U.S. surface water samples collected at 130 locations, prior to 1967, only about 5% contained beryllium from 0.01 to $1.22 \text{ } \mu\text{g/l}$ with a mean of $0.19 \text{ } \mu\text{g/l}$. On the basis of literature cited, the EPA has set a criterion for the

protections of aquatic biota in hard water 1.1 mg/l. The criterions for soft water is 0.011 mg/l. The EPA also set a criterion of 0.10 mg/l beryllium for irrigation waters for use on all soils except 0.50 mg/l for use on neutral to alkaline fine-textured soils.⁽⁷⁾

3.5.3.2 Beryllium Toxicity to Plants. Haas reported in 1932 that 2.5 mg/l beryllium was toxic to some varieties of citrus seedlings. Romney found that beryllium at 0.5 mg/l in nutrient solutions reduced the growth of bush beans.⁽³⁴⁾ Three years later Romney and Childress reported that beryllium concentrations of 2 mg/l in nutrient solutions and/or 4% of the cation-exchange capacity in soil reduced the growth of tomatoes, peas, soybeans, lettuce, and alfalfa.⁽³⁵⁾

3.5.3.3 Beryllium Toxicity To Animals. NIOSH reports that toxic neoplastic effects were observed in rabbits following an intravenous (IV) injection of 20 mg/kg body weight.⁽³⁶⁾ Osteosarcomas have been produced in animals by IV injections of suspensions of metallic beryllium.

The aquatic toxicological literature on beryllium is very limited. Hann and Jensen reported an aquatic toxicity rating for the beryllium as a 96-hr median tolerance limit (TLM) range from 10 to 100 ppm. In 1960, Tarzwell and Henderson reported on the acute toxicity of a variety of metals to the fathead minnow and bluegill.⁽³⁷⁾ Their report included data on the 6-hr TLM of the sulphate, nitrate and chloride of beryllium, showing a 10 to 130-fold greater toxicity of beryllium in soft water (20 mg/l as CaCO_3) than in hard water (400 mg/l). Guppies were found to be much more resistant (X100) to beryllium in hard water than soft.⁽³⁸⁾ In 1973, Slonim and Slonim undertook to determine the precise relationship between the TLM for BeSO_4 in guppies and water hardness. They conducted several static bioassays and from the data computed the regression equation and correlation coefficient. Two additional bioassays were run to test the equations. The observed TLM was in agreement with the calculated value within the 95% confidence range in both bioassays.⁽³⁹⁾ Table 3.4 reviews some of this toxicity data.

3.5.4 Effluent Control in Beryllium Production

Effluents from the mining, milling and beryllium metal production processes are primarily solid wastes and Be dust. Solid wastes include overburden

TABLE 3.4. Beryllium Toxicity To Aquatic Animals - 96-hr TLM.

<u>Compound</u>	<u>Organism</u>	<u>Concentration (mg/l)</u>		<u>Comments</u>
Beryllium Nitrate	fathead minnows	0.15	Be	soft water
		20.	Be	hard water
Beryllium Chloride	fathead minnows	0.15	Be	soft water
		15.		hard water
Beryllium Sulfate	fathead minnows	0.2	Be	soft water
		11		hard water
	bluegill	1.3	Be	soft water
		12	Be	hard water
	guppy	0.19	Be	soft water
		20	Be	hard water
	salamander larvae	4.7	Be	soft water
		26	Be	hard water

waste from mining, ore residue from milling, and cake waste containing $\text{Al}(\text{OH})_3$, MnO_2 and PbCrO_3 and slag slurry from metal production. Brush Wellman estimates that current overburden waste ranges from 5.5 to 10 yd^3 /ton of ore extracted. Ore residue from the milling process is estimated at 1.6 tons residue per ton of beryllium hydroxide produced.

Emissions of beryllium dust after control are estimated as 0.2 lb/ton of beryllium produced in the mining phase, and 30 lb/ton of beryllium produced in milling and metal production.⁽⁴¹⁾

A lack of precipitation and groundwater eliminates the significance of liquid effluents at the Spor Mountain mine. Liquid effluents from milling and metal production are primarily recycled. Those that are not recycled are assumed to be neutralized and collected in tailings ponds.

3.6 NIOBIUM

Total domestic consumption of niobium in 1977 was estimated to be near 4 million pounds of contained metal. Of this amount, 80% was consumed as steel alloy, 16% was used in the manufacture of special superalloys, and 4% was

consumed as pure metal or ferrotantalum-niobium.⁽⁴¹⁾ The use of super conducting alloys for fusion reactor structures will increase the demand for this metal.

The U.S. has no deposits of Nb that can be classified as reserves, and is estimated to have only 3% of total world resources, which are not economically recoverable. The U.S. imports all niobium in the form of pyrochlore ore concentrate, niobium-tantalite concentrate, tin slag, ferroniobium. Niobium metal, synthetic concentrates, or special alloys are imported. Brazil is the chief source of niobium and accounted for 31% of total U.S. imports in 1977. Other significant exporters of niobium are West Germany and Canada, who supplied the U.S. with 24% and 20% of total 1977 imports, respectively. Nigeria and Thailand combined contributed 20% to total U.S. imports of niobium in 1977.⁽⁴²⁾

Pyrochlore $((\text{Na,Ca})_2(\text{Nb,Ta})_2\text{O}_6(\text{O,OH,F}))$ and niobium-tantalite $((\text{Fe,Mn})(\text{Nb,Ta})_2\text{O}_6))$ ores and concentrates are sold on the basis of % total pentoxides of niobium and tantalum. Niobium concentrate generally has 65% total pentoxide with a Nb_2O_5 to Ta_2O_5 ratio that ranges from as low as 8-1/2:1 to a high of 10:1. Pyrochlore concentrate generally contains 50 to 55% Nb_2O_5 and less than 1% Ta_2O_5 .

Ferroniobium is a ferroalloy produced as an intermediate product and contains from 60 to 70% niobium. Grade depends on the %age of tantalum present which can range from 0.5% to 6.0%.

Niobium pentoxide is another intermediate product whose grade also depends upon the percentage of tantalum present. Purity can be as low as 89.0% or as high as 99.7% with the Ta_2O_5 present ranging from 0.25 to 10.0%.

Niobium metal is marketed in many physical shapes and forms with purity ranging from 95 to 99.9%. The production of Nb metal through a presently used process and a conceptual process is described in the following subsection. Effluents that result from these processes are identified and compared with current standards. Data on niobium resources as well as information on the current domestic niobium industry is given in Appendix C.

3.6.1 Niobium Metal Recovery

Niobium and niobium alloys are processed domestically by Kawecki-Berylco Industries, Inc., Boyertown, Pennsylvania; Mallinckrodt, Inc., St. Louis, Missouri; and Fansteel, Inc., Chicago, Illinois. These companies primarily recover high-purity niobium oxide by digesting niobium-tantalite ores in hydrofluoric acid, followed by solvent extraction and purification. The solvent-extraction method perfected by the Bureau of Mines uses hydrofluoric acid, sulfuric acid and methyl isobutyl ketone to recover more than 99% of the contained niobium and tantalum.

Currently, pyrochlore concentrates are chiefly utilized in the manufacture of ferroniobium used in steel production.⁽⁴³⁾ Large-scale demand for niobium for fusion applications will necessitate use of pyrochlore as a base material rather than niobium-tantalite ores. Use of pyrochlore ores is more economical because of the low tantalum content. Studies performed by Teledyne Wah Chang at Albany indicate that small tantalum values do not affect niobium's superconducting properties. Thus, tantalum need not be recovered.

Since current processing techniques do not represent future niobium production methods, a conceptual process, developed by the Canadian Bureau of Mines is used as a reference. This process for recovering Nb_2O_5 from pyrochlore uses solvent extraction methods to purify the niobium. First the pyrochlore concentrate is decomposed using sulfuric acid. After decomposition, hydrofluoric acid is added and the solution enters a solvent extraction battery with methyl isobutyl ketone (MIBK) as the extractant. Niobium is back-extracted with an ammonium fluoride solution and precipitated with ammonia. Finally, the filtered and washed precipitate is calcined to Nb_2O_5 .

The process yields a 99.9% niobium product. Ninety-six percent of the original niobium is recovered.⁽⁴⁾ The process is described in more detail below; a flow diagram is shown in Figure 3.16.

FIGURE 3.16. Nb_2O_5 Recovery from Pyrochlore Concentrate

Digestion of Pyrochlore

Dry ground pyrochlore concentrate is mixed with 93% sulfuric acid, heated, and allowed to react for approximately one hour at a temperature of approximately 200°C. The resultant slurry is then cooled and gradually diluted with water until the acid composition is 40 to 45% H₂SO₄.

Filtration

The slurry is filtered in a drum filter and washed twice, once with 35% H₂SO₄ solution and then with water. The primary and wash filtrates are combined prior to extraction, and the residue is sent to waste.

Solvent Extraction

Hydrofluoric acid is added to the niobium-bearing concentrate and wash solutions to a concentration of 5% HF. The acid solution is contacted with the methyl isobutyl ketone in mixer settlers or pulse columns. Niobium-bearing liquor then enters a stripping column where the niobium is extracted from the organic using a 5% solution of aqueous ammonium fluoride. The now barren MIBK enters a third column and is scrubbed with a 4.75% H₂SO₄ solution and recycled. The used scrub solution is sent to waste.

Precipitation

The solution of Nb₂O₅-NH₄F is neutralized with ammonia in agitated tanks. The Nb₂O₅ precipitates and is filtered in drum filters. The filtrate is washed with 4 ml of water per gram of precipitate and dried in a rotary dryer at 100°C.

Calcination

The dried precipitate is calcined at 900°C for one hour. The highly pure (99.9%) Nb₂O₅ is packaged and stored.

3.6.2 Niobium Metal Production

Niobium metal alloyed with titanium is necessary for use in fusion reactors. Teledyne Wah Chang, of Albany, Oregon, one of the major suppliers of superconducting alloys, plans on expanding their present facilities during

1979 and 1980 to approximately 10 million lb/yr of niobium metal. This capacity should meet all potential requirements for the fusion industry. Other metal producers include Fansteel, Inc. and Kawecki-Berylco Industries, Inc.

A process similar to the one utilized by TWCA to produce niobium metal and niobium/titanium alloys is used as a reference. The process is shown in Figure 3.17. Process steps include reduction of Nb_2O_5 , election beam melting and arc melting.

Reduction of Niobium Oxide

To reduce niobium oxide to niobium metal, the oxide is mixed with aluminum powder. Other oxidizing agents may be added. The powders are placed in a steel vessel lined with refractory material and ignited by the addition of a little water. The reaction proceeds rapidly and forms a solid chunk of Nb metal. Gases are also formed and sent to plant scrubbers before being release to the environment. The solid Nb metal is sent to electron beam melting.

FIGURE 3.17. Niobium Metal Production

Electron Beam Melting

The niobium metal is placed in an electron beam furnace. After a high vacuum is generated, a tungsten electrode is heated by direct resistance until electrons are emitted. Niobium metal passes through the electron beam, melts, and drops into a copper mold. The process is repeated a minimum of four times. Carbon, nitrogen and oxygen gases that form during the process are directly vented out the stack. Gaseous metallic elements are removed in a cold trap.

Arc Melting

The metal electrode is prepared by welding titanium to the niobium ingot. The metals are then charged in a consumable electrode arc melter. After a vacuum is generated, the arc is struck by passing current through the electrodes and lowering them to the base of the furnace where a small charge to the same material as the electrode is placed. The electrode is drawn back and the arc is struck. The electrode melts off and drops into a copper mold at the bottom of the furnace. The process is repeated a minimum of two times to achieve homogeneity in the resultant Nb/Ti alloy.

3.6.3 Niobium Release Standards

The ability of niobium to move through the biosphere appears to be very limited. Niobium is found naturally in all waters at the ppb level. The average crustal abundance of niobium is in the ppm range. Niobium is strongly adsorbed by clay particles or is usually found as an insoluble oxide or hydride. Root uptake is very limited which explains why environmental bioaccumulation factors for general food types; e.g., cereals, grains, meat, poultry, dairy and vegetables are generally near 0.05. The maximum reported concentration factor in animal tissues or organs is 0.06 for mouse kidney.⁽⁴⁵⁾

Spinach leaves were found to efficiently accumulate niobium from surrounding air. Zirconium-niobium concentrations were 3000 to 4000 times greater than air. However, plant to terrestrial animal bioaccumulation factors are found to be the same, 0.008, for two different species of birds feeding on different foods with quite different ⁹⁵Nb concentrations. The concentrations of Ar-Nb is diminished by a factor of 10 in the grass-cow-milk food chain.⁽⁴⁵⁾ These animal data suggest a biological discrimination against niobium in these test animals.

3.6.3.1 Niobium As An Air Pollutant

The discharge of niobium to the uncontrolled atmosphere is not regulated by the federal government, however, occupational exposure is. The presently regulated eight-hour occupational threshold limit values (TLV) is 50 mg/m³. Ottinger reduced this value by a factor of 100 to represent an acceptable ambient level of 0.05 mg/l m³ for continuous exposure to members of the general public.⁽⁴⁶⁾

3.6.3.2 Niobium As A Water Pollutant

Niobium is found naturally in all waters at the ppb level. The discharge of niobium to water as an environmental effluent is not regulated by the Federal Government. One citation found recommends a maximum permissible concentration for niobium in the aquatic environment (specifically reservoir water). The Russian paper recommends an MPC of 0.01 mg/l of niobium based on their experimental enzyme alteration data. Dawson offers a desired maximum ambient water concentration of 0.02 mg/l based on acute toxicology data taken from the literature.

3.6.3.3 The Chemical Toxicity of Niobium to Terrestrial Animals

No toxic effects in humans have been reported. Laboratory experiments using rats have shown that niobium is transported across the placental barrier when maternal exposure is great. Niobium catalyzes the oxidation of 5-hydroxytryptophan, an intrinsic brain metabolite.⁽⁴⁸⁾ Scanlon summarizes that appreciable environmental pollution with niobium could be associated with fetal abnormalities, especially of the central nervous system.⁽⁴⁸⁾ In animals, niobium has generally been observed to be more toxic than tantalum and some of the rare earth elements; toxicity is due to its interference with metal activated enzyme systems.

A substantial fraction of inhaled niobium oxide is retained by dogs, with the oxalate showing a biological half-life of over 300 days. Inhalation of niobium elevates the amount of the metal in the lungs, skeleton, kidneys and muscle over levels observed from intravenous injections. Analysis of human tissue indicates the affinity of niobium for fat, liver, spleen and red blood cells.⁽⁴⁵⁾

Over half of the normal daily intake (about 620 μg) is excreted in the urine in the same time period.⁽⁴⁵⁾ A long-term pulmonary retention half-life has been calculated as 224 days.

Single intravenous doses of 20 mg Nb/kg or less to rats, rabbits and dogs were not fatal. Easterly and Shank selected a few references from the literature to compare some lethal doses of niobium to small mammals (Table 3.5).⁽⁴⁵⁾

3.6.4 Effluent Control from Niobium Production

The following discussion of effluents from niobium metal recovery pertains to the tantalum-niobium process since no effluent data is available on the conceptual pyrochlore process. Effluents from both processes are expected to be similar in type, although quantities are expected to differ.

Basically, gases from the dissolution step are scrubbed and routed out the stack. Liquid wastes are neutralized with lime in clarifying ponds; the precipitate (CaF) is land filled while clarified pond liquid is sent to a sewage treatment plant.

The following emissions data were provided by Kawecki-Berylco for their Boyertown plant:

- Hydroflouric Acid gas:
 - Scrubber Input -- 0.135 lb HF/lb $\text{M}_2\text{O}_5^{(a)}$
 - Stack Output -- 0.0054 lb HF/lb M_2O_5
 - Scrubber Blowdown -- 3.0 gal/lb M_2P_5
 - Ammonia Gas
 - Exhausted to Atmosphere -- 0.12 lb/lb Nb_2O_5

(a) M refers to tantalum-niobium metal.

TABLE 3.5 Comparison of Lethal Doses of Niobium per Kilogram of Body Weight to Small Mammals(20,45)

<u>Compound</u>	<u>Animal</u>	<u>Toxicity(1)</u> <u>(LD₅₀/No. Days)</u>	<u>Concentration(2)</u> <u>(mg/kg)</u>	<u>Route of</u> <u>Administration</u>
Niobium Penta- Chloride	Rat	(LD ₅₀ /10)	14 mg Nb/kg	IP
	Mouse	(LD ₅₀ /7)	21 mg Nb/kg	IP
	Mouse	(LD ₅₀ /7)	323 mg Nb/kg	Oral
Niogium Chloride	Rat	(LD ₅₀)	40	IP
	Rat	(LD ₅₀)	1400	Oral
	Mouse	(LD ₅₀)	829	Oral
Potassium Niobate	Rat	(LD ₅₀ /7)	725 mg Nb/kg	Oral
	Rat	(LD ₅₀ /7)	92 mg Nb/kg	IP
	Mouse	(LD ₅₀ /7)	23 mg Nb/kg	IP
Potassium Oxotetra- fluoroniobate	Mouse	(LD ₅₀)	130	Oral

- 1) LD₅₀ = lethal dose to 50% of the population within the specified time period.
- 2) Concentration in mg/kg unless otherwise stated
- 3) IP = Inter-peritoneal

- Liquid Acid Wastes (mostly HF/H₂SO₄) -- 11 gal/lb Nb₂O₅
 - Liquid Lime to Neutralize/Control Flouride -- 9 lb/lb Nb₂O₅
 - Ammonia Recovery Unit
 - From Process -- 3.8 gal/lb Nb₂O₅
 - Lime for Reaction/Flouride Control -- 1.4 lb/lb Nb₂O₅
 - NH₃ Recovered -- 1.0 lb/lb Nb₂O₅
 - Volume to Waste Plant -- 5.2 gal/lb Nb₂O₅

Effluent control equipment for niobium metal production consists of scrubbers for gas generated during reduction. Cold traps are utilized during electron beam melting to condense any gaseous metallic elements. No appreciable gases are generated during metal production.

Kawecki-Berylco reports that at their Boyertown plant dust from thermite reduction amounts to:

- 0.425 lb/lb Nb Dust Collector Input
- 4.25×10^{-5} lb/lb Nb Stack Output (99.9% Efficiency)

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4.0 EFFLUENTS IN THE FUSION FUEL CYCLE

In the preceding sections, the fusion material requirements for lithium, beryllium, niobium, helium, deuterium and tritium were identified for reference fusion power plant design. The effluents associated with the acquisition of these raw materials and their preparation for use in the fusion fuel cycle were also identified.

In this section, effluent source terms that represent the fusion increment in material production are modeled for release into the environment. The resulting concentrations will be compared to federal standards, if they exist, for exposure to the general public. On the basis of these comparisons, the adequacy of control technology to limit the release of effluents will be addressed in light of fusion material requirements.

4.1 EFFLUENT RELEASES

The types of effluents associated with material acquisition in the fusion fuel cycle that were identified in the preceding chapter are summarized in Table 4.1 through 4.10. As can be seen the major effluents are associated with the mining and processing of lithium, niobium and beryllium. The processing steps where the effluents arise are also given. Effluents are indicated by (E) in the tables. Effluents are nonproduct materials that are discharged to the environment. Wastes (W) are defined as nonproduct materials that are disposed of with the intent of isolation from the environment.

4.2 INCREMENTAL RELEASES FOR FUSION REQUIREMENTS

In order to evaluate the impact of fusion on materials acquisition, the major effluents identified on the preceding tables will be summarized here in terms of metal production in addition to deuterium products.

For lithium, shown in Table 4.11, airborne particulates will be released as $\text{LiAlSi}_2\text{O}_6$ from ore crushing operations and as Li_2CO_3 and Li_2NaPO_4 in the processing of brines from Clayton Valley and Searles Lake. Gaseous effluents included H_2S , SO_2 , H_2O and CO_2 . Processing losses are based on an estimated annual production of approximately 1000 and 200 metric tonnes of

TABLE 4.1 Limestone Calcination Process at North Carolina Mines

Process	Waste/Effluent	Accumulation/Release Rate
Crushing, Handling, Screening, Pile Losses	(E) Fugitive Li Ore Dust	0.65 lb Ore Dust/Mt Ore Processed
Flotation	Usuable Processing Products Mica, Feldsp, Quartz	90 lb/day Suspended Solids
Rotary Kiln	(E) CO ₂	7 MT CO ₂ /MT of LiOH Produced
Ball Milling	(E) Fugitive Dust (W) Tailings Water to Pond	Small Quantities of Dust
Leaching/Thickening	(W) Tailings Water Ore Residue	18 MT Ore Residue/MT of LiOH Produced
Recrystallization	(W) Waste NaOH	

TABLE 4.2 Lithium Carbonate Production-Acid Roast Process at North Carolina Mines

Process	Waste/Effluent	Accumulation/Release Rate
Crushing, Screening, Handling, Pile Losses	(E) Fugitive Li Ore Dust	0.65 lb Ore Dust/Mt Ore Processed
Ball Milling	(E) Fugitive Li Ore Dust	~ Small
Flotation	Usuable Processing Products- Mica and Feldspathic Sands	90 lb/day Suspended Solids
Rotary Kiln	(E) Particulates	
Secondary Ball Milling	(E) Fugitive Dust (W) Tailings Water to Ponds	Small Quantities of Dust
Acid Roast Kiln	(E) Particulates	
Leaching and Neutralizing	(W) Ore Residue to Ponds	
Rotary Filtration	(W) Solids to Stockpile	
Dryer	(E) Particulates	

Table 4.3 Lithium Carbonate Production at Searles Lake Brines

Process	Waste/Effluent	Accumulation/Release Rate
Evaporator	(W) Aqueous Processing Wastes, H_3PO_4 with some Li	H_3PO_4 ~1300 MT/year Li ~ 8 MT/year
Washing	(W) Wash Water Waste Returned to Lake	~700 MT/year
Secondary Evaporator	(E) Airborne Gases, Vapors, CO_2 , H_2O	~300 MT CO_2 /year
Settling Tank	(W) Waste Water	~980 MT/year
Rotary-Drum Dryer	(E) Airborne Particulates	400 MT/year
Bagging Filter Operations	(E) Fugitive Dust (Li_2CO_3) (W) Solid Wastes	89 lb Dust/year <10 lb Solids/year

TABLE 4.4 Lithium Extraction at Searles Lake Brines

Process	Waste/Effluent	Accumulation/Release Rate
Main Cycle Evaporators	(E) H_2S (W) Usable Processing Products Suspended in Liquids	H_2S <0.03 ppm
De-Watering	(W) Unsalable Suspended Solids NaCl Returned to Lake	2500 MT NaCl/day
Soda Ash Plant	Na_2CO_3	300 MT/day
Salt Cake Plant	Na_2SO_4	600 MT/day
Flotation/Separation	(W) Waste Water-Foam Recycled to salt Cake Plant	
Filtration-Washing	(W) Wash Waste Returned to Lake (W) Filter Cake Returned to Salt Cake Plant (E) Water Vapor	200 MT Wash Water/yr 2000 MT Filter Cake/yr 100 MT/year
Gas-Fired Roaster	(E) Airborne Particulates	1 lb/hr
Bagging-Filtering	(E) Fugitive Dust (Li_2NaPO_4) (W) Solid Wastes	0.05 lb Dust/day 0.045 lb Solids/day

TABLE 4.5 Lithium Carbonate Production at Clayton Valley Brines

Process	Waste/Effluent	Accumulation/Release Rate
Solar Evaporation	(W) Solid Wastes, Brine Wastes	~6000 MT Solids/day
Precipitation	(W) Salts: $\text{Mg}(\text{OH})_3$, Na, K, Ca	13.5 lb Salts/100 lb Brine
	(E) H_2S	Atmospheric Concentration <0.03 ppm
Filtration	(W) Solid Wastes	
Boiler	(E) Airborne Particulates, SO_2	4×10^{-6} MT Part./MT Li Product 1.6×10^{-4} MT SO_2 /MT Li Product
Rotary Dryer	(E) Airborne Particulates	0.05 MT Part./MT Li Product
Conveyor	(E) Airborne Particulates	0.008 MT Part./MT Li Product
Pellet Rotary Dryer	(E) Airborne Particulates	0.0015 MT Part./MT Li Product for K, Na, Mg

TABLE 4.6 Salt Lake Lithium Recovery

Process	Waste/Effluent	Accumulation/Release Rate
Solar Evaporation	(W) Unsalable Salts, NaCl, K, Mg	2.5×10^6 MT Lake Brines/year Processed for K, Na, Mg
	(W) Tailings Water to Ponds	Waste Water: 7.5×10^6 MT/year accumulated brine waste

Particulate, SO_2 and H_2S Releases Similar to Clayton Valley Brine Process

TABLE 4.7 Lithium Metal Production

Process	Waste/Effluent	Accumulation/Release Rate
Li ₂ CO ₃ or Li(OH) ₂ Conversion to LiCl	(E) CO ₂ and H ₂ O vapor	0.5 MT CO ₂ /MT of LiCl Produced
Electrolytic Cell	Chlorine Gas Collected for Commercial Use	5 kg Cl/kg Li Metal

46 kWh Required to Produce 1 kg of Li Metal Product

TABLE 4.8 Niobium Ore Processing

Process	Waste/Effluent	Accumulation/Release Rate
Filtration-Washing	(W) Ore Residues	?
Centrifugation	(W) ZrO ₂	?
Extraction	(W) Used H ₂ SO ₄ Solution Sent to Holding Tank	?
Metal Production:		
Thermite Reactor (Reduction)	(E) Airborne Nb Dust	4.25 x 10 ⁻¹⁵ lb Nb Product
Electron Beam Melting	(E) Airborne Particulates and Gases, C, N ₂ , O ₂	?
Arc Melting	Insignificant Wastes	?

TABLE 4.9 Niobium-Tantalum Process for Metal Production

Process	Waste/Effluent	Accumulation/Release Rate
Dissolution	(E) HF Vapors	0.0054 lb HF/lb (Nb Ta) ₂ O ₅
	(E) NH ₃ Gases	0.12 lb NH ₃ /lb (Nb Ta) ₂ O ₅
	(W) Acid Wastes (HF/H ₂ SO ₄)	11 gallons/lb (Nb Ta) ₂ O ₅
NH ₃ Recovery Unit	(W) Aqueous Wastes Recovered NH ₃	5.2 gallons/lb (Nb Ta) ₂ O ₅

TABLE 4.10 Beryllium Production

Process	Waste/Effluent	Accumulation/Release Rate
Mining	(W) Overburden, Rock	5 to 10 yd ³ /MT Be Ore
	(E) Fugitive Ore Dust	0.2 lb Ore Dust/MT Be Metal
Milling-Processing	(E) Airborne Be Ore Dust	30 lbs Ore Dust/MT Be Metal
	(W) Slag Slurry Wastes	
	(W) Ore Residues	1.5 MT/MT Be
Solvent Extraction	(W) Aqueous Acid Wastes	?
Chemical Conversion	(W) Process Wastes (NH ₄) ₂ SO ₄	?
Filtration	(W) Cake Wastes	?
	Al(OH) ₃	
	MnO ₂	
	PbCrO ₃	
Thermal Decomposition	(W) NH ₄ F	?
Evaporation	(W) Precipitated Solids	?

TABLE 4.11 Effluents From Lithium Production

Effluent	Source Term (kg/Mt Li)	Release	
		Initial Fusion Inventory (kg/1000MWe)	Annual Fusion Requirement (kg/1000 MWe-yr)
Li Ore Dust (LiAlSi ₂ O ₆) ^(a)	42	223	7
Li ₂ CO ₃ Dust	0.24	193	5.8
Li ₂ NaPO ₄ Dust	0.04	32	0.97
Particulates	54.	4.4 x 10 ⁴	1.3 x 10 ³
	2.3 (2300)	1851 (2.8 x 10 ⁶)	55.8 (5.6 x 10 ⁴)
H ₂ S	0.03 ppm		
CO ₂	1640	1.3 x 10 ⁶	4 x 10 ⁴
H ₂ O (vapor)	10 ⁵	8 x 10 ⁷	2.4 x 10 ⁶
SO ₂	0.15	121	3.6

(a) Both Mine and Brine Lithium Recovery

lithium metal from Clayton Valley and Searles Lake, respectively. The releases for the above effluents are given per metric tonne of Li. Also shown is the fusion increment to supply the initial fusion inventory as well as yearly requirements.

For niobium, airborne effluents are mainly niobium dust from the thermite reactor. No estimates are available for effluents from the electron beam melting process. For the niobium-tantalum process, additional effluents are HF vapor and ammonia gases. Again, these effluents are summarized in Table 4.12, based on metric tonne of niobium metal produced. The effluents associated with amounts produced for fusion inventory are given. Annual requirements of Nb are zero.

TABLE 4.12 Effluents From Niobium Production

Effluent	Source Term (kg/MT Li)	Release	
		Initial Fusion Inventory (kg/1000 MWe)	Annual Fusion Requirement (kg/100 MWe-yr)
Nb Dust	.04	2	0
HF Acid Vapor	8.2	372	0
NH ₃ Gas	184	8285	0

For the production of beryllium, the only effluents identified are the release of ore dust. This will be in the form of $2\text{Be}_2\text{SiO}_4\text{H}_2\text{O}$ and $\text{Be}(\text{OH})_2$. Table 4.13, gives the releases associated with supplying the initial inventory to the plant and the yearly requirements. The beryllium requirements for design option 2, a gas-cooled tokamak reactor, are used here.

TABLE 4.13 Effluents From Beryllium Production

Effluent	Source Term (kg/MT Li)	Release	
		Initial Fusion Inventory (kg/1000 MWe)	Annual Fusion Requirement (kg/100 MWe-yr)
Be Ore dust ($2\text{Be}_2\text{SiO}_4\text{H}_2\text{O}$)	0.1	26	1.3
$\text{Be}(\text{OH})_2$ Dust	13.6	3523	180

For deuterium production, effluents are associated primarily with the production of D_2O at a heavy water production facility. The releases assume an inventory of 8 kg of deuterium at the fusion plant. Effluents from deuterium production are identified in Table 4.14.

TABLE 4.14 Effluents From D₂O Production

Effluent	Source Term (kg/MT Li)	Release	
		Initial Fusion Inventory (kg/1000 MWe)	Annual Fusion Requirement (kg/100 MWe-yr)
SO ₂	9400	376	4300
NO _x	3100	124	1400
Salts (water)	0.62	0.25	0.29
H ₂ S (to water)	0.72	0.03	0.33
(atmospheric)	7.5 x 10 ⁻⁵	3 x 10 ⁻⁶	3.4 x 10 ⁻⁵
D ₂ O	10	.4	4.6
D	2	.08	0.92

A number of liquid wastes have been identified in the preceding tables, including tailing waste water, brines, etc. However, provisions are always made to control these wastes, typically in settling ponds. Under routine operating conditions, no discharge of liquid waste occurs and these wastes are not identified with release effluents. Holding ponds have often leaked under abnormal or accident conditions, however the quantification of accidental releases is beyond the scope of this study.

4.3 EFFLUENT CONCENTRATIONS

In order to determine the impact of supplying materials to the fusion fuel cycle, the annual release terms given in the previous section will be evaluated here for release into the environment.

The resource production facilities considered in this report (mining, milling, brines, metal production, etc.) operate under existing standards for liquid, solid and airborne effluent releases. To evaluate the impact on the surrounding environment as a result of supplying the fusion material requirements, airborne effluents identified in the preceeding text were modeled for atmospheric release. The releases correspond to the annual requirements for a fusion power plant producing 1 GWe.

As a first rough estimate of downwind concentrations, releases are assumed to be made from a stack height of 45 meters. Airborne concentrations are measured one mile downwind with an assumed dilution factor of $9.9 \times 10^{-8}/\text{m}^3$. The results are shown in Table 4.15 for all the materials considered.

TABLE 4.15 Effluent Concentrations Corresponding to Annual Fusion Requirements of 1 GWe.

<u>Effluent</u>	<u>Air Concentration ($\mu\text{grams}/\text{m}^3$)</u>
$\text{LiAlSi}_2\text{O}_6$ Dust	2×10^{-5}
Li_2CO_3 Dust	2×10^{-5}
Li_2NaPO_4 Dust	3×10^{-6}
Particulates	0.18
CO_2	0.13
SO_2	$1 \times 10^{-5}(\text{a})$
	$1.3 \times 10^{-2}(\text{b})$
Nb Dust	6.3×10^{-6}
HF	1.2×10^{-3}
NH_3	2.6×10^{-2}
Be Ore Dust	4.1×10^{-6}
$\text{Be}(\text{OH})_2$ Dust	5.6×10^{-4}
D	3.0×10^{-6}
H_2S	$\ll 10^{-5}$
NO_x	4.4×10^{-3}
T (Gaseous)	$\ll 10^{-10} \mu\text{Ci}/\text{cm}^3$
(Water)	$\ll 10^{-6} \mu\text{Ci}/\text{cm}^3$

(a) Lithium Production

(b) D_2O Production

Table 4.16 lists the maximum ambient air concentration levels recommended for the materials released in the fusion materials cycle. There are no existing environmental standards for Li_2CO_3 , $\text{LiAlSi}_2\text{O}_6$ and Li_2NaPO_4 . The occupational value for LiH, $25 \mu\text{g}/\text{m}^3$, is used for comparative purposes.

TABLE 4.16 Maximum Ambient Environmental (Air)
Concentrations Allowed Under Current
Regulations^(a)

Li(Lih)	25 $\mu\text{g}/\text{m}^3$ ^(b)
Nb	0.05 mg/m^3
Be	0.01 $\mu\text{g}/\text{m}^3$
T	4×10^{-5} $\mu\text{Ci}/\text{ml}$
D	No Standard
Particulates	75 $\mu\text{g}/\text{m}^3$ ^(c)
SO ₂	80 $\mu\text{g}/\text{m}^3$ ^(c)
CO ₂	9000 mg/m^3
NO _x	100 $\mu\text{g}/\text{m}^3$ ^(c)
HF	3 ppm
NH ₃	18 mg/m^3
H ₂ S	15 mg/m^3

-
- (a) Annual average
 (b) 8 hour time weighted average, occupational number only
 (c) Particulates, SO₂ and NO_x standards vary according to state or region

There are also no standards for Be ore dust or Be(OH)₂. The Be metal standard, 0.01 g/m^3 , is used to evaluate the Be-related releases. The regulations governing the release of particulates, H₂S, CO₂, SO₂, NO_x and HF depend upon state regulations and will thus vary from region to region.

4.4 EFFLUENT CONTROL

A comparison of Tables 4.15 and 4.16 indicates that the effluent concentrations corresponding to the annual release associated with the fuel cycle of a 1 GWe fusion power plant will be well within the maximum ambient environmental concentrations allowed under current regulations. It would appear that existing environmental control standards and the technology to keep releases within these standards are adequate for a fusion industry of several hundred

fusion power plants. However, the toxic chemical release problems of the heavy metal industry will certainly be exacerbated by a fusion industry. Whether this becomes a significant environmental control problem will have to be assessed in greater detail and depth as the fusion industry matures.

APPENDIX A

LITHIUM ASSESSMENT

THE LITHIUM INDUSTRY TODAY

Lithium was first isolated in its elemental form in 1955. Because of its unique chemical properties, lithium is regarded as a strategic metal and the industry had experienced rapid growth prior to and during World War II. Currently, the most significant use of lithium is in the smelting of aluminum where it lowers costs and increases production by reducing both potline temperature and fluoride emissions. Lithium is also utilized in ceramic production, as a base for high temperature grease, and in the production of batteries and synthetic rubber (refer to Section 3.4 for a discussion of commercial producers).

DOMESTIC LITHIUM RESOURCES

The U.S. Bureau of Mines has put proved lithium reserves at 407,000 short tons, or 18% of proved world reserves.⁽¹⁾ In addition to these proved reserves, other identified domestic resources are estimated to contain 3.2 million tons of recoverable lithium.

The two major sources of lithium in the U.S. are from pegmatite dikes in North Carolina and subsurface brines in Nevada. Also, minor amounts of lithium are recovered as a co-product of alkali salts from brines in California. Production figures by area are not available as this information is considered to be privileged by the three producing firms.

In addition to these producing areas, other resources have been identified. Industry sources believe it may be commercially feasible to extract up to half of the lithium present in the Great Salt Lake, Utah. This would add 286,000 short tons to domestic resources.

Hard Rock Pegmatites

A significant portion of domestic lithium reserves occur in granite pegmatites. More specifically, 363,000 tons of lithium are estimated to be contained in the spodomene belt near Kings Mountain, North Carolina.⁽²⁾ This unzoned pegmatite belt contains ore bodies up to 250 feet thick and 3,000 feet long, along a mineralized region 25 miles long and one-half mile wide. Recent

drilling has determined the pegmatites to be at least 600 feet in depth. Geologic interpretation of the area has hypothesized the mineralization may prevail down to the 4,500 foot level. Should this prove correct, total reserves in the area would increase to over 2,500,000 tons of lithium.

Mineral Brines

Clayton Valley, Nevada is the most significant lithium-rich brine deposit now in production. Using 1978 recovery techniques, the reserves in this deposit are placed at 44,000 short tons.⁽³⁾ However, industry sources believe higher market prices would make existing technology economical to the extent that a total of 120,000 tons could be recovered.⁽²⁾

At Searles Lake, California lithium is recovered as a co-product of soda ash, potash, borax and salt cake. The U.S. Bureau of Mines estimates the proved reserves of this deposit to be 1,000 short tons of lithium, while industry sources say total resources could run as high as 24,000 tons.

Another potential source of domestic lithium is the Great Salt Lake in Utah. Evans, of the Committee on Nuclear and Alternative Energy Systems (CONAES), has estimated it may be commercially feasible to extract about half the lithium from this brine if the recovery program includes the other minerals present. Should such a process prove economically feasible, lithium resources would increase by 286,000 short tons.

Domestic Reserves and Resources

Table A.1 lists domestic tonnages of proved lithium reserves by area and producing firm. These reserves were ascertained by exposure and/or exploration. Also, a 25% open-pit mine loss is assumed.

Table A.2 indicates lithium resources that are highly probable as inferred by geologic evidence. In the North Carolina pegmatite belt, these resources refer to deposits within the developed mine areas beyond those that have been proved by exploration drilling. Also, these resources include undeveloped areas in the belt. Brine resources include lithium known to be present in deposits now being produced but is not recoverable at present technology and prices. In addition, brines in the Great Salt Lake are included.

TABLE A.1. Proved Domestic Lithium Reserves (000 Short Tons)

CAROLINA PEGMATITE BELT

<u>Location</u>	<u>Li Ore Minable by Open-Pit</u>	<u>Li%</u>	<u>Li Reserves</u>
Kings Mountain	31,371	.68	213
Bessemer City	21,241	.70	<u>149</u>
			362

WESTERN U.S. BRINES

<u>Location</u>	<u>Li Brine</u>	<u>Li%</u>	<u>Li Reserves</u>
Clayton Valley, Nevada	222,880	.02	44
Searles Lake, California	20,000	.005	<u>1</u> 45
Total Reserves.			407

NORTH CAROLINA PEGMATITES

Location and Demography

The North Carolina pegmatite belt lies in the south western part of the State, 30 miles West of Charlotte. The mineralized zone starts near the South Carolina border and strikes northeast parallel to the inner Piedmont Mountains between the communities of Grover and Lincolnton.

At present, open-pit mines are operating at two locations along the belt. The southern most mine is located near the community of Kings Mountain, which has a population of 8,465. The other mine site lies 5 miles north of Kings Mountain near Bessemer City, a community with a population of 4,990. Other communities along the mineralized belt may be impacted by future mining operations. Archdale and Crouse both lie within undeveloped areas of the ore belt. The populations of these two communities are 4,870 and 600, respectively. Also, Cherryville, a community with a population of 5,258, lies a few miles to the west of large, undeveloped ore bodies.

TABLE A.2 Potential Domestic Lithium Resources (000 Short Tons)

CAROLINA PEGMATITE BELT				
<u>Location</u>	<u>Li Ore Mineable by Open-Pit</u>	<u>Li Ore Mineable By Underground</u>	<u>Li%(a)</u>	<u>Li Resource</u>
Bessemer City	7,521	7,439	.65	97
Kings Mountain	5,207	6,943	.70	85
Undeveloped Areas	3,306	378,250	.69	<u>2,631</u> 2,813

WESTERN U.S. BRINES			
<u>Location</u>	<u>Li Brine</u>	<u>Li%</u>	<u>Li Reserves</u>
Clayton Valley, Nevada	425,648	.02	85
Searles Lake California	523,450	.005	26
Salt Lake, Utah	8,265,000	.0035	<u>286</u> 397
Total Potential Domestic Lithium Resources 3,210			

(a) Because pegmatites are unzoned, underground and surface ore grades are assumed to be the same.

Topography

The North Carolina pegmatite belt trends northeasterly along the foothills of the inner Piedmont province. Approximately 80% of the mineralized area is characterized by gently sloping terrain, varying between 100 feet to 300 feet in local elevation difference. Bedrock is overlain with red-yellow to gray hapludult soil that has a subsurface horizon of thin clay accumulation as well as appreciable weatherable minerals. Oak, hickory, and pine forests cover the undeveloped areas.

Meteorology

The mineralized belt lies in an area with an average precipitation rate of 45 inches per year. Snowfall is less than 8 inches per annum. Annual runoff ranges from 15 to 20 inches while the mean yearly pan evaporation rate ranges from 48 to 64 inches.

Prevailing winds in the pegmatite belt area tend to be east to northeast from January through late summer, and from the south in the fall. Average wind velocity and direction as well as barometric pressure are listed in Table A.3

The annual average mixing depth above the pegmatite belt at which pollutants disperse during the daily cycle ranges from 1.0 to 1.5 km. Average annual wind speed through this mixing depth ranges from 6 to 8 m/sec. Ventilation is the product of mixing depth times wind speed through the mixing depth and is proportional to the volume of air available to dilute pollutants emitted into the atmosphere. Afternoon ventilation data over the mineralized area given in Table A.4.

Mean annual temperature in the belt is 75°F. Seasonal temperature extremes range from 45°F to 100°F.

TABLE A.3 Wind and Barometric Data for the North Carolina Pegmatite Belt

<u>Month</u>	<u>Wind Velocity</u>	<u>Direction</u>	<u>Barometric Pressure</u>
January	8 mi/hr	East	1020 Milibars
April	7 mi/hr	N.E.	1016 Milibars
July	13 mi/hr	N.E.	1016 Milibars
October	14 mi/hr	South	1018 Milibars

TABLE A.4 Afternoon Ventilation Data for the North Carolina Pegmatite Belt (000 m²/sec.)

<u>Season</u>	<u>Afternoon Ventilation</u>
Spring	12-28
Summer	6-12
Fall	6-12
Winter	6-12

Ecology

The pegmatite belt of North Carolina is located in the Piedmont Mountains. The dominant vegetation in this region is white oak, post oak and hickory on the mountain slopes, and willow and alder in the stream valleys.

Dominant wildlife in the area is deer and turkey. Animals of lesser importance are listed in Table A.5.

Present or future mining operations within the mineralized area appear to be of no threat to any species of wildlife.

Hydrology

No major aquifers underlie the pegmatite belt. Ground water is from subsurface circulation of precipitation and runoff. Because country rock in the Piedmont belt has low permeability and lacks deep fractures, water circulation is generally confined to a zone in the upper 30 feet of bedrock. These consolidated rocks shunt water laterally to surface seepage areas and perennial stream channels in a rather direct path along interconnecting fractures.

LeGrand⁽⁴⁾ collected data on the chemical character of ground water within the pegmatite belt. These data indicate the water is slightly acidic, with relatively few dissolved minerals in solution. Table A.6 summarizes the findings on an average.

The study also indicates that the solvent power of ground water increases with increased carbon dioxide content and time of contact. These may be important considerations when implementing an effluent abatement program.

TABLE A.5 Animals That Occur in the North Carolina Pegmatite Belt

Bear	Skunk
Fox	Ruffed Grouse
Gray Squirrel	Song Birds
Fox Squirrel	Rabbits
Raccoon	Copperhead Snake
Opposum	Rattlesnake

TABLE A.6 Chemical Composition of Ground Water Within the North Carolina Pegmatite Belt (Values in ppm, Except pH)

<u>Contaminant</u>	<u>Concentration</u>
Dissolved Solids	85
Silica	27
Iron	0.16
Calcium	8
Magnesium	4
Sodium/Potassium	6
Bicarbonate	43
Sulfate	1.7
Chloride	8
Fluoride	0
Nitrate	5
Calcium Carbonate	37
pH	6.5

Geology and Mineralization

The prevailing geology of the North Carolina pegmatite belt is that of metasediments in contact with a granite intrusion of quartz-monzonite. Impalcement of the batholith caused additional alteration of the metasediments, but failed to produce regional melting of the country rock. There is evidence of this in the well preserved sedimentary structure of the Kings Mountain series.⁽⁵⁰⁾ The resulting deformation did, however, produce wall rock alterations of two types. The first is a coarse mica schist containing oligoclase, staurolite and tourmaline. Lithium pegmatites are not found in this wall rock as a result of diversely origented biotite reducing permeability and sealing out volatiles from the magma. Of greater interest is the second type of altered wall rocks that enclose lithium pegmatites. Here the magma has caused zones of brecciation and the alteration of amphibolite and biotite to chlorite before implacement of pegmatites. It has been suggested that the older breccias provided areas of low resistance by which pegmatites were concentrated, thus forming local ore bodies within the mineralized area.⁽⁶⁾

Implacement of the pegmatites occur at close intervals along the strike of the inner Piedmont and parallel to the foliation of mica schist. These lithiated pegmatite bodies range up to 250 feet wide and 3,000 feet long, along a mineralized region 25 miles long and one-half mile wide. Drilling programs to date in 1976 had reached depths of more than 600 feet without encountering significant changes in the mineralogy of pegmatites or wall rocks.

Kesler⁽⁶⁾ has provided detailed descriptions of mineral composition from long range analysis of milling ore in the early 1960's. These studies have determined the pegmatites to be composed of five major minerals in the proportions listed in Table A.7.

The longest of these minerals are spodumene and microcline which can reach crystal sizes up to 3 feet in length, but more commonly are less than one foot. Spodumene crystals in the Kings Mountain deposits have an average Li_2O content of 7.5%. The crude ore has a concentration of about 1.5% Li_2O . Also, because these pegmatites display no zoning, ore grades remain uniform throughout the deposit. As a result, feldspar and muscovite can be recovered as a byproduct of the milling process.

Recent geologic interpretations, by Kesler⁽⁵⁾ and Norton⁽⁷⁾ of the North Carolina pegmatite belt, will influence future exploration and mining programs. These authors have suggested that lithium in the pegmatite deposits was originally deposited as shallow water evaporites in the Kings Mountain sediments. It has been hypothesized that the lithium was then leached from the metasediments by volatile fluids being driven off the granitic magma as it

TABLE A.7 Mineral Composition of Kings Mountain Pegmatites

<u>Mineral</u>	<u>% Composition</u>
Spodumene	20
Quartz	32
Muscovite	6
Feldspar	
Albite	27
Microcline	14
Trace Minerals	1

crystallized. Because the metasediments are steeply dipping West, this hypothesis implies the source wall rock and lithiated pegmatites should persist at depth Westward under the present mining area. This conceptual model will provide a basis for future exploration drilling. If it is determined that spodumene deposits do, in fact, extend Westward and to greater depths, this may alter present mining techniques. That is, underground mining methods may have to be adopted rather than surface mining.

Mining

Currently, open-pit mining operations are being carried out at two locations along the North Carolina pegmatite belt. Although the mines are being operated by different mineral firms, the process is similar at both locations.

Prior to mining, vegetation and overburden are removed from the pit area. Marketable trees are harvested. Overburden in the area ranges from 20 to 100 feet and is composed of soil and weathered ore and gangue material. Initial waste to ore stripping ratios are generally greater than 3, while steady-state stripping ratios average 1.8. Overburden is used in the construction of loads and tailings dams.

A system of slope benches ranging from 20 to 40 feet wide is used to excavate the pit. Lithium-bearing crystals are segregated with loaders once the ore has been fractured with explosives. The gangue material, primarily amphibolite, is sold to a local aggregate producer. Currently, the pit area near the community of Kings Mountain is approximately 55 acres. The mine to the North near Bessemer City covers approximately 20 acres.

Industry officials report neither operation has plans for underground mining within the next 20 years. However, should underground mining become a viable alternative at some future date, no unique engineering problems are anticipated.

CLAYTON VALLEY, NEVADA BRINES

Location and Demography

The Clayton Valley brine deposits are located approximately 35 miles Southwest of Tonopah, Nevada, near the California border. The small community

of Silver Peak, with a population of 150, supports the nearby mining operations. Another small mining community of Goldfield, population 300, lies 25 miles to the East. Tonopah, the nearest metropolitan center, has a population of 1,650. Areas South and West of the brine fields are unpopulated.

Topography

Prevailing surface features around the Clayton Valley brine deposits are characterized by plains with low surrounding mountains. Local elevation difference ranges from 1,000 to 3,000 feet, while 50% to 80% of the total surface area has less than 8% slope incline. Regional soil type is argid with a sub-surface clay horizon.

Meteorology

Precipitation in Clayton Valley is minimal. Rainfall and snowfall each amount to less than 8 inches per year. Because of the dry climate, annual runoff amounts to only 0.2 inches. Average annual pan evaporation rate ranges from 96 inches to 112 inches. Moderate easterly winds prevail throughout most of the year while barometric pressure fluctuates close to 1014 millibars year around.

The annual average mixing depth at which pollutants disperse above Clayton Valley during the normal daily cycle is greater than 2.4 km. Wind speed through the mixing depth ranges from 6 to 8 m/sec. on the average annually. Ventilation available to dilute atmospheric pollutants above the brine is given in Table A.8.

Mean annual temperature in the Clayton Valley is 66°F. Yearly temperature extremes range from 15°F to 112°F.

Ecology

The Clayton Valley lies near the western edge of the great basin and is characterized by a semidesert environment. The chief plants present are deciduous microphyll shrubs, sage brush, salt brush, greasewood and shadscale. Outstanding and widely distributed animals are jack rabbits, kangaroo rats, kangaroo mice, pocket mice, grasshopper mice, and antelope ground squirrels.⁽⁸⁾ Influent species found in the region are given in Table A.9.

TABLE A.8 Afternoon Ventilation Data for the Clayton Valley Area
(000 m²/sec.)

<u>Season</u>	<u>Afternoon Ventilation</u>
Spring	18-24
Summer	18-24
Fall	6-12
Winter	6-12

TABLE A.9 Wildlife Occurring in the Clayton Valley Area

Coyote	Prairie Falcon
Mule Deer	Swanson's Hawk
Kit Fox	Golden Eagle
Bobcat	Bald Eagle
Horned Owl	Red-Tail Hawk

Because the Clayton Valley brine fields lack surface water, aquatic ecology is not affected. Also, it is assumed the mining process will have negligible impacts on terrestrial ecology.

Hydrology

Clayton playa is one among a series of playas in a diversified complex formed by a large topographic depression southwest of Tonopah, Nevada. This depression not only has internal surface drainage, but is closed hydrologically as well. Because the Clayton playa occupies the lowest elevation within the playa complex, it receives large amounts of discharge water circulating through deep surrounding aquifers. Precise data is lacking; however, the Nevada Department of Conservation and Natural Resources estimates about 22,000 feet of water annually enters Clayton playa from local precipitation and from subsurface inflow.⁽⁹⁾ All water entering the playa is discharged by evaporation, surface springs or utilized by Man.

Subsurface water in the valley increases in mineral content near the center of the playa. Springs near the margin support fresh water vegetation and is potable enough to serve the Silver Peak public water system. Near the

center of the playa vegetation decreases to sparse salt grass characteristic of highly mineralized water. Also, wells in this central region yield water with specific conductance of 242,000 μ ohms.⁽¹⁰⁾

Geology and Mineralization

The Clayton Valley, Nevada lithium brines are unique in that it is the only brine deposit being mined exclusively for the recovery of lithium in the U.S. The deposit consists of a subsurface hydrologic brine system percolating through a closed basin filled with quaternary sediments. These sediments are composed of lacustrine siltstone, hectorite clay, and rhyolitic volcanics. Evaporite chloride and sulfate salts were deposited near the surface of the playa that covers approximately 32 square miles.

Chemical analysis of Clayton Valley brines show high concentrations of sodium chloride with lesser amounts of lithium, potassium, magnesium and calcium. Lithium content varies from 0.02% to 0.17% with the average of about 0.035%.^(3,11) Typical data observed in Grime holding ponds are listed in Table A.10.

It is generally agreed that the source of lithium at Clayton Valley is from hydrothermal leaching of volcanics within the basin. The rhyolitic tuffs are known to contain lithium. Also, hydrothermal ore deposits occur West of

TABLE A.10 Clayton Valley Pond System⁽¹²⁾

<u>Pond No.</u>	<u>Area (Acres)</u>	<u>Brine Volume (Thous. Gallons)</u>	<u>Li Conc. (ppm)</u>	<u>Solid Phases</u>
1	760	200,000	680	None
2	520	120,000	780	NaCl
3	90	25,000	930	NaCl
4	86	22,000	1200	NaCl
5	46	15,000	1400	NaCl, gypsum Mg(OH) ₂
6	41	5,000	1900	NaCl, CaCO ₃
7	17	1,500	2400	NaCl, Glaserite,
8	13	1,000	3100	NaCl, Glaserite, KCl
9	14	3,000	6000	NaCl, Glaserite, KCl

Clayton Valley. These hydrothermal fluids are assumed to have reached Li/Cl ratios of 0.01 or higher and were localized within the closed evaporation basin prior to the precipitation of lithium bearing salts.⁽¹¹⁾ Subsequent hydrologic activity within the basin has redissolved the evaporites to form the current brine system.

A system of production wells drilled into the playa indicate lithium-rich brines permeate the strata to depth of at least 800 feet.

SEARLES LAKE, CALIFORNIA BRINES

Location and Demography

The Searles Lake brine deposits are located in a remote section of Southern California, North of the Mojave Desert and Southwest of Death Valley National Monument. The brine wells are located on the West side of the Searles Lake playa near the small community of Westend. Trona, a somewhat larger community, lies 5 miles North of Westend and on the Northern edge of the salt playa. The populations of Westend and Trona are 100 and 1,500, respectively. With the exception of these two communities, the Searles Lake area is unpopulated.

Topography

Searles Lake is a salt playa lying to the East of the Cascade Mountain range. Relief is typical of the surrounding basin and range topography. That is, low mountains border featureless plains. These plains comprise 50 to 80% of the total surface area and have slopes with less than 8% incline. Soil is of the arid type with a loamy clay horizon. Intermittent areas of rock land with little soil cover are also present. Vegetation is limited to sparse creosote brush.

Meteorology

Annual rainfall in the Searles lake area averages from 0 to 8 inches. Mean annual snowfall is also less than 8 inches. The lack of precipitation reduces runoff, which ranges from 0.2 to 1 inch per year. A dry, arid climate accounts for high evaporation rates at Searles Lake that range from 112 to 128 inches per year.

Prevailing winds are moderately high year around and tend to come from the west. Wind and barometric data are given in Table A.11.

Atmospheric pollutants above Searles Lake are effectively dispersed at a mixing depth of 1.0 to 1.5 kilometers elevation. Wind speeds through this mixing depth range from 4 to 6 meters/sec throughout the year. Table A.12 lists the volumes of air provided by these conditions.

Annual mean temperature of the Searles Lake area is 62°F, with extremes reaching a low of 15°F and a high of 112°F.

Ecology

The Searles lake salt playa lies within the Mojave Desert biome. Dominant vegetation in the area is creosote bush and salt brush. However, burrobrush and yucca are also present. Dominant mammals of the desert are the kangaroo rat and pocket mouse. Wildlife of lesser density are listed in Table A.13.

Because life within the playa area is virtually absent, mining operations pose no threat to ecology.

Hydrology

The Searles lake playa lies in Searles Valley which is a closed hydrologic basin. That is, the only natural way ground water can escape the basin is by solar evaporation. Basement-complex rocks are impermeable and only yield small quantities of water from fractures, joints, or bedding planes. The principal fresh water-bearing unit in Searles Valley is a porous pleistocene alluvial deposit of sand, gravel, silt and clay. Water quality ranges from 2,00 to

TABLE A.11 Wind and Barometric Data for the Searles Lake Area

<u>Month</u>	<u>Wind Velocity</u>	<u>Direction</u>	<u>Barometric Pressure (millibars)</u>
January	15 mi/hr	South	1019
April	13 mi/hr	East	1015
July	15 mi/hr	Northeast	1011
October	20 mi/hr	East	1013

TABLE A.12 Afternoon Ventilation Data for the Searles Lake Area
(000 m²/sec.)

<u>Season</u>	<u>Afternoon Ventilation</u>
Spring	6-12
Summer	0-6
Fall	0-6
Winter	3-6

TABLE A.13 Wildlife Occurring in the Searles Lake Playa Area

Kit Fox	Mourning Dove
Desert Coyote	Desert Iguana
Spotted Skunk	Sidewinder Rattlesnake
Quail	Desert Tortoise
Roadrunner	Phainopepla Bird

12,00 ppm dissolved salts. The commercial brine deposits are located in pleistocene lake deposits of clay, silt sand, and salt.

In the playa area the water table is at or near the surface, resulting in moisture loss to the atmosphere and deposition of salts on the playa surface.

Geology and Mineralization

Searles Lake lies within Searles Valley in the Southwest corner of the Great Basin and range district. A combination of limited internal drainage and high evaporation rates have produced numerous salt playas within the district. The Searles Lake area is third in a chain of six such playas that were interconnected by a pleistocene river draining into Death Valley. Subsequent evaporation and gravity caused increasing downstream concentrations of minerals in the sediments and brines.

Currently, Searles Lake is nearly a dry salt flat with surface area of approximately 100 km². Two distinct salt bodies lie within 50 m of the surface. The uppermost of these deposits has an approximate volume of 1 Km³, while the lower salt body is about half that volume.⁽¹³⁾ Lacustrine mud

layers are found at the surface of the playa, between the evaporite deposits and below the lower evaporite deposit.

These evaporite layers are composed of a variety of mineral salts and are the basis for current mining efforts by the Kerr-McGee Corporation. Minerals present in the salt layers are listed in Table A.14. However, data on percent composition was not available.

Because lithium is highly soluble, no lithium minerals are found in the evaporities. Rather, all lithium has either remained in brine solutions or has been trapped in the lattice of clay minerals in lakes and river beds.

Lithium is currently being recovered at Searles Lake as a co-product of potash (KCl), borax ($\text{Na}_2\text{B}_4\text{O}_7 \cdot 10\text{H}_2\text{O}$), salt cake (Na_2SO_4) and soda ash (Na_2CO_3). In this process, brines are pumped from wells in the dry lake bed and minerals are recovered by fractional crystallization and chemical precipitation. Lithium content of the brine averages .008% Li or .015% Li_2O .⁽¹³⁾

TABLE A.14 Minerals Present in the Searles Lake Evaporation Deposits⁽¹⁴⁾

<u>Mineral</u>	<u>Chemical Composition</u>
Aphthitalite	$\text{K}_3 \text{Na}(\text{SO}_4)_2$
Borax	$\text{Na}_2 \text{B}_4 \text{O}_7 \cdot 10 \text{H}_2\text{O}$
Burkeite	$2 \text{Na}_2 \text{SO}_4 \cdot \text{Na}_2 \text{CO}_3$
Halite	NaCl
Hamksite	$9 \text{Na}_2\text{SO}_4 \cdot 2\text{Na}_2\text{CO}_3 \quad \text{KCl}$
Mirabilite	$\text{Na}_2\text{SO}_4 \cdot 10 \text{H}_2\text{O}$
Nahcolite	NaHCO_3
Northupite	$\text{Na}_2\text{CO}_3 \cdot \text{MgCO}_3 \cdot \text{NaCl}$
Sulfohalite	$2 \text{Na}_2\text{SO}_4 \cdot \text{NaCl} \cdot \text{NaF}$
Teepleite	$\text{Na}_2\text{B}_2\text{O}_4 \cdot 2 \text{NaCl} \cdot 4 \text{H}_2\text{O}$
Thenoldite	Na_2SO_4
Tinculconite	$\text{Na}_2\text{B}_4\text{O}_7 \cdot 5 \text{H}_2\text{O}$
Trona	$\text{Na}_2\text{CO}_3 \cdot \text{NaHCO}_3 \cdot 2 \text{H}_2\text{O}$
Tvchite	$2 \text{Na}_2\text{CO}_3 \cdot 2 \text{MgCO}_2 \cdot \text{Na}_2\text{SO}_4$

The primary source of this lithium is assumed to be from geothermal fluids emanating from the nearby Long Valley caldera.⁽¹⁴⁾ Because these source fluids no longer feed the producing wells, depletion and dilution accompany recovery.

Chemical analysis of present geothermal solutions from the Long Valley caldera suggest that as much as 95% of the primary lithium has been lost along the route of the Owens River before reaching Searles lake. This has exploration implications in that lithium-rich brines or clay deposits may exist in the upstream regions of the prehistoric lake chain.

THE GREAT SALT LAKE BRINES

Location and Demography

The Great Salt Lake covers an area of 1,000 square miles and is located in the northcentral portion of the Utah panhandle. The Great Salt Lake Desert extends west and south of the lake to the Nevada border. Three metropolitan areas border the lake on the East side along a 60 mile route. First in the chain is the state capital, Salt lake City, having a population of 172,000. Further to the North along the lake are Bountiful and Ogden, having populations of 29,900 and 69,000, respectively. The Great Salt Lake Desert west of the lake is virtually unpopulated.

Topography

The Great Salt Lake lies in a large basin formed by the evaporation of ancient Lake Bonneville. Surrounding terrain is characterized by smooth plains bordered by high mountains. Local elevation differences exceed 3,000 feet. Soil in the lowlands is composed of evaporite salts and natragids low in organic matter. Vegetation in the basin is sparse, consisting mainly of greasewood and tule marsh. Large areas of barren desert are also common.

Meteorology

The Salt Lake basin and surrounding mountains get an average annual rainfall that ranges from 8 to 16 inches. Snowfall in the area averages 32 to 64 inches per annum. Surface runoff into the Salt Lake basin is less than

2 inches a year due to the relatively dry climate. The annual pan evaporation rate ranges from a low of 48 inches to a high of 64 inches.

Prevailing winds in and around the Salt Lake area are north to northwest and range from 5 to 10 miles per hour. Seasonal wind velocities and direction as well as barometric pressure are listed in Table A.15.

The mixing depth at which pollutants disperse above the Great Salt Lake ranges from 2 to 2.5 kilometers. Average annual wind speed through the mixing depth is 6 to 8 m/sec. Afternoon ventilations available to disperse pollutants are given in Table A.16.

Average yearly temperature in the Salt Lake Basin is 46°F with annual extremes ranging from a low of -22°F to a high of 105°F.

Ecology

The processing of brine from the Great Salt Lake poses little threat to the surrounding ecosystem. The entire western edge of the lake is salt desert, devoid of life. Dominant vegetation on the east side of the lake is saltbrush

TABLE A.15 Wind and Barometric Data for the Salt Lake Basin

<u>Month</u>	<u>Wind Velocity</u>	<u>Direction</u>	<u>Barometric Pressure (millibars)</u>
January	10 mi/hr	Northwest	1022
April	4 mi/hr	North	1011
July	3 mi/hr	Northwest	1012
October	5 mi/hr	Northwest	1016

TABLE A.16 Afternoon Ventilation Data for the Great Salt Lake Basin
(000 m²/sec.)

<u>Season</u>	<u>Afternoon Ventilation</u>
Spring	12-18
Summer	18-24
Fall	6-12
Winter	3-6

greasewood and tule marsh. Wildlife found on the eastern side of the lake are Kangaroo rats, rabbits, pocket mice and Mormon crickets. Seagulls are common along the entire lake shore.

Hydrology

The Great Salt Lake lies in the northeast portion of the basin and range province. Arid climatic conditions minimize surface water and runoff into the Salt Lake Basin. Minor fresh water unconsolidated aquifers near the lake occupy intermontane valleys buried by alluvial and glacial deposits.

Brine aquifers occur to the west and southwest of the lake in the Salt Lake desert. Data on the volume of the aquifers, however, is not available. Tritium isotopic studies reveal that the only significant natural recharge is from rainfall. This contributes about 2.3 inches per year.⁽¹⁵⁾ Lateral and vertical migration of brine is negligible. Another source of recharge is brackish water imported from distant wells by the Kaiser Chemical Company to be used in a flotation separation process. Depletion is from natural evaporation and evaporation of expelled milling brine.

Geology and Mineralization

Prehistoric Lake Bonneville was formed during the pleistocene by internal drainage into the basin and range district. During the late pleistocene, climatic conditions became more arid, reducing internal drainage and increasing evaporation rates. These conditions have continued up to the present and have reduced the lake to a series of salt playas and the remanent brine waters of the Great Salt Lake in northern Utah.

Lithium ions in the Salt Lake brines originated from volcanic and tectonic activity during late eocene times. The lithium was concentrated as evaporites and brines in playas prior to the subsequent formation and evaporation of Lake Bonneville.⁽¹⁶⁾

Brines in the Great Salt Lake have three distinct areas of lithium concentration. The area north of the railroad causeway contains 64 ppm lithium, the deep southern arm contains 43 ppm, and the shallow southern arm contains 22 ppm lithium. The northern arm is becoming richer in lithium because of

evaporation and restricted inflow past the causeway. These enriched Northern brines tend to enrich the deep southern arm. The shallow southern arm is becoming fresher due to inflow from surface runoff.⁽¹⁷⁾

At least 2 brine aquifers are known to exist beneath the Bonneville Salt playa. One, a shallow aquifer, has been sampled and described in the literature in three independent studies.^(15, 18, 19) A summary average of this data is given in Table A.17.

TABLE A.17 Chemical Composition of Shallow Aquifer Brines in the Salt Lake Desert

<u>Ion</u>	<u>Concentration in Grams per Liter</u>
Cl	119.38
SO ₄	5.69
Na	69.57
K	3.62
Li	.003
Ca	1.38
Mg	1.83

The above studies also sampled a deep brine aquifer at depths greater than 325 meters. Data from these tests are averaged and summarized in Table A.18.

TABLE A.18 Chemical Composition of Deep Aquifer Brines in the Great Salt Lake Desert

<u>Ion</u>	<u>Concentration in ppm</u>
Ca	1,550
Mg	1,440
Na	43,550
Li	17
K	1,900
SO ₄	6,150
Cl	71,400

Currently, potassium and magnesium are being recovered from brines in the southern shallow arm and the northern arm of the Great Salt Lake as well as from the shallow aquifer in the Salt Lake desert. Operations on the shallow arm of the lake have design capacity to remove 41,000 metric tonnes of magnesium per year. The facility on the northern arm of the lake has design capacity to recover 220,000 metric tonnes of potassium sulfate and 90,000 tonnes of bischofite ($\text{MgCl}_2 \cdot 6 \text{H}_2\text{O}$) per year. Lithium follows magnesium in the recovery process and is recoverable at a ratio of .0055 to 1. If 90,000 tonnes of bischofite yields 8,020 tonnes of magnesium, the Salt Lake brine operators have the potential to recover 240 tonnes of lithium per year.⁽¹⁵⁾

Kaiser Chemical Company currently recover potassium and magnesium from the shallow aquifer in the Great Salt Desert. It has been estimated that at present production rates, the operation could continue for 25 to 40 years before the brine deteriorated from 1% KCl to 0.5% kCl.⁽¹³⁾ This rate has also been used on an estimation of lithium deterioration, placing the lifetime potential of the shallow brine aquifer at a yield of 475 to 760 metric tonnes of lithium.⁽¹⁵⁾

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APPENDIX B

BERYLLIUM ASSESSMENT

THE BERYLLIUM INDUSTRY TODAY

Beryllium has gained worldwide importance in the past few decades because of its desirable chemical properties. A high thermal conductivity and melting point as well as rigidity and low density has made beryllium a valuable commodity in the aerospace and electronics industries. Also, new applications for the use of beryllium as a neutron multiplier in the fusion reactor fuel cycle are being investigated.

Within the past year, apparent domestic beryllium consumption was approximately 50 short tons of metal. Consumption%ages by industry were as follows: as a metal in nuclear reactors and aerospace applications, 39%; as an alloy in electrical equipment, 36%; in electronic components, 15%; and other, 10.⁽¹⁾

Historically, beryl has been the primary mineral from which beryllium was extracted. However, this process requires hand-cobbing large crystals from pegmatite deposits. Because of high labor costs and limited domestic resources of large beryl crystals, the U.S. has been a large importer of foreign beryl. Domestic production of nonpegmatite beryllium deposits in Utah, however, has started to reverse this trend. While production figures are considered confidential, imports and apparent consumption figures indicate these nonpegmatite ores satisfy about half of total U.S. demand.

This nonpegmatite deposit, located at Spor Mountain in western Utah, is the only significant beryllium producing area in the U.S. Bertrandite is the primary beryllium ore mineral in the deposit and occurs as low grade, highly disseminated, supergene mineralization in water-laid volcanic tuff.

Beryllium deposits also occur as hydrothermal replacement of carbonate rocks in Alaska's Western Seward Peninsula. Although beryllium is not being produced there, the U.S.G.S. regard the deposits to be of economic significance.

DOMESTIC BERYLLIUM RESOURCES

Large amounts of beryl occur in domestic pegmatite deposits. The U.S.G.S. estimate as much as 800,000 tons of beryl are contained in North Carolina

pegmatite belt alone. However, because of low grade and small particle size, this beryl is not economically recoverable with the present state-of-the-art technology.⁽²⁾ Well zoned pegmatite deposits in New England, South Dakota, and Colorado are known to contain about 10,000 tons of large, readily recoverable beryl crystals. However, these deposits are also not classified as reserves because the high cost of hand sorting prohibits recovery.

Significant domestic reserves and resources occur as low grade nonpegmatite deposits in Western Utah and Seward Peninsula of Alaska.

Nonpegmatite Deposits

The only significant domestic production of beryllium is from deposits at Spor Mountain in West Central Utah. Bertrandite is the principal ore mineral in these deposits and occurs in localized units of water-laid volcanic tuff. Precise dimensions of the units are lacking; however, it is known that they vary from 2 to 20 feet in thickness and that some units are at least 200 feet in length. The U.S. Bureau of Mines places proved reserves within the deposits at 28,000 short tons of contained beryllium metal.⁽¹⁾

Domestic beryllium deposits also occur in economically significant quantities in the Central York Mountains, Seward Peninsula, Alaska. Here a suite of five beryllium minerals were formed by hydrothermal alteration where dikes cut marien carbonate rocks. Proved reserves of beryllium in the York Mountain deposits are at least 4,500 short tons of contained metal.⁽³⁾

Data is lacking on total domestic resource of beryllium. However, the Bureau of Mines estimates that proved reserves plus subeconomic resources would yield at least 80,000 tons of beryllium metal.⁽¹⁾ Deposits at Spor Mountain and the Seward Peninsula have been estimated to each contain approximately 35,000 tons of this metal. The remaining 10,000 tons occurs in small low grade nonpegmatite deposits at various locations in the U.S.

SPOR MOUNTAIN BERYLLIUM DEPOSITS

Location and Demography

The Spor Mountain beryllium deposits are located in The Thomas Mountain Range in the west central portion of Utah bordering Nevada. The nearest

populated area is the community of Delta, which lies 45 miles to the southeast, and has a population of approximately 2,000. The unincorporated communities of Callao and Trout Creek are located approximately 30 miles to the west of the deposits. Both of these communities have sparse rural populations. Because of the mine's remote location, expansion would not affect populated areas.

Topography

Spor Mountain lies at the margin of the Great Salt Lake Desert and displays typical basin and range topography. That is, featureless, arid plains are occasionally broken by low, barren hills. More than 80% of the total surface area has less than 8% slope grade while local elevation difference is less than 300 feet. Soils consist of volcanic alluvium and Lake Bonneville sediments, low in organic matter.

Meteorology

The Spor Mountain beryllium deposit is located in an arid semidesert environment. Average annual precipitation is less than 10 inches. Snowfall in the area is less than 10 inches per year. Because of the arid conditions, runoff averages less than 2 inches per year. Average annual pan evaporation ranges from 48 to 64 inches.

Winds in the area are moderate throughout the year and tend to come from the South. Wind and barometric data are given in Table B.1.

TABLE B.1 Wind and Barometric Data for the Spor Mountain Area

<u>Month</u>	<u>Wind Velocity</u>	<u>Direction</u>	<u>Barometric Pressure</u>
January	10 mi/hr	N.N.W	1022
April	4 mi/hr	N.N.E.	1013
July	3 mi/hr	N.W.	1011
October	5 mi/hr	N.W.	1016

The average afternoon mixing depth at which pollutants disperse above Spor Mountain ranges from 2.0 to 2.5 km. Average wind speed through this mixing depth ranges from 6 to 8 m/sec. Ventilation data for the area are given in Table B.2.

TABLE B.2 Afternoon Ventilation Data for the Spor Mountain Area
(000 m²/Sec)

<u>Season</u>	<u>Afternoon Ventilation</u>
Spring	12-18
Summer	over 24
Fall	12-18
Winter	6-12

Average yearly Fahrenheit temperature is in the high 40's with annual extremes ranging from minus 20 to over 100.

Ecology

Spor Mountain lies on the eastern margin of the Great Basin and is characterized by a Great Basin biome. The chief plants present are deciduous microphyll shrubs, sagebrush, saltbrush, greasewood and shadscale. Principal animals of the area are jack rabbits, kangaroo rats, kangaroo mice, pocket mice, grasshopper mice, and antelope ground squirrels.⁽⁴⁾ Influent species found in the region are given in Table B.3.

Aquatic life does not occur in the area surrounding Spor Mountain because of the lack of surface water.

Hydrology

Because of the arid climate and lack of drainage, no perennial surface water exists at Spor Mountain. Unconsolidated aquifers of alluvial sediments do, however, contain ground water in the surrounding area. Recharge to the aquifers is from precipitation and runoff from nearby mountains.

Geology and Mineralization

The regional geology of the Spor Mountain beryllium deposits is characterized by territory volcanics overlying early paleozoic carbonate rocks. Because the deposit lies within the Lake Bonneville Basin, lowlands are covered with lucustrine sediments.

Lindsey⁽⁵⁾ has identified three separate volcanic units overlying the paleozoic marine sediments. The oldest and middle volcanic units are composed

TABLE B.3 Wildlife Occurring in the Spor Mountain Region

Coyote	Prairie Falcon
Mule Deer	Swanson's Hawk
Kit Fox	Golden Eagle
Bobcat	Red-tailed Hawk
Horned Owl	Bald Eagle

of latite, andesite, basalt, and ash flow tuffs. The youngest volcanic unit is of greatest interest in that it contains the beryllium mineralization. This younger unit is composed of extrusive topaz-bearing alkali rhyolite flows interbedded with a water-laid tuff. The water-laid tuff contains the beryllium and occurs as numerous local units that grade into the rhyolite. The tuff unit also contains clasts of older volcanic rocks and marine carbonate that grade into breccia. Approximately 90% of the tuff is comprised of amorphous pumice and glass shards while the remaining 10% is crystalline. These crystals are quartz, sunidine, plasioclase, clinoptiolite, crystobalite and potassium feldspar.

The mineralization at Spor Mountain occurred in an epithermal environment, is supergene in nature, and shows evidence of being both stratigraphically and structurally controlled. More specifically, regional and local faults provided conduits for rising hydrothermal mineral solutions. In addition, these fluids were contained by impervious rhyolite units but spread laterally to mineralize the permeable water-laid tuff.

The ore mineral at Spor Mountain is bertrandite ($2 \text{ Be}_2\text{SiO}_4 \cdot \text{H}_2\text{O}$). There is general agreement that beryllium entered the tuff deposit as a complex fluoride ion. As the fluoride replaced the carbonate clasts, these complex ions were depleted in fluoride and broke down to allow precipitation of bertrandite.^(5,6)

Because the mineralized tuff is composed of localized units rather than a single stratigraphic unit, vein size varies. These beryllium rich layers range from 2 to 20 feet in thickness. The length of the layers along strike and dip is unknown; however, some have been traced for at least 200 feet.⁽²⁾

SEWARD PENINSULA BERYLLIUM DEPOSITS

Location and Demography

The Lost River beryllium deposits are located in the York Mountains on the western tip of the Seward Peninsula, Alaska. The nearest metropolitan center is Nome, 90 miles to the southeast with a population of 2,500. The small villages of Tin City and Wales lie approximately 30 miles to the west and have a combined population of less than 200. The mineralized area is unpopulated.

Topography

The topography of the Western Seward Peninsula is characterized by featureless tundra at near sea level elevation. The beryllium deposits occur in the Southern slopes of the York Mountains that trend Northeasterly through this lowland.

Total elevation within the mineralized area is generally less than 1,000 feet, while local elevation differences are generally less than 500 feet. In the region of the Lost River, mineralized zones border the upper limit of the tundra.

Soils in the area tend to be rocky with little decomposed plant material present. Also, a subsurface permafrost horizon is present.

Meteorology

The climate of the mineralized area is moderated somewhat by its close proximity to the Bering Sea. The mean annual temperature of the region is 24°F. Average monthly extremes range from 5°F to 50°F. Mean annual precipitation and snowfall ranges from 8 to 16 inches and 32 to 64 inches, respectively. Winds are generally to the Southwest and average from 10 to 25 miles per hour throughout the year.

Data on the mixing depth at which pollutants disperse in the atmosphere are not available.

Ecology

Beryllium deposits in the York Mountains, Alaska, are characterized by a tundra biome. A permafrost soil horizon supports a vegetation cover of lichen and cotton grass. Sparse willow, arctic heath, and birch shrub are also present.

Dominant animals within the mineralized area are grizzly bear, caribou, and arctic wolf. Influent animals are listed in Table B.4.

Significant amounts of surface water exist within the mineralized area at various times during the year. Lost River and Rapid River as well as numerous streams cross the mineral deposits. However, because the majority of surface water tends to be intermittent, aquatic life is minimal to non-existent.

Hydrology

The York Mountains lie within a region characterized by continuous permafrost. Because of this, the circulation of ground water is restricted and generally occurs as frost. However, due to a general lack of soil within the mineralized area, little ground water occurs in this form. Large amounts of water do, however, circulate through the porous limestone country rock. Faulting in the area also provides a conduit for subsurface water. In addition, local aquifers occur in unconsolidated valley alluvium.

Geology and Mineralization

Country rock of the Western Seward Peninsula, in the area of the beryllium deposits, is composed of early paleozoic argillaceous limestone. Subsequent faulting and the emplacement of granitic intrusive magma provided both the conduit and metal rich solutions by which to mineralize the area. Studies by Sainsbury^(3,7) indicate that generally the ore bodies are the result of hydrothermal replacement of limestone and dolomite, and occur locally where lamprophyre dikes intruded the shattered marine rocks along existing fault zones. Fluorite is the principal gangue mineral present and amounts to about 50% of total ore bulk. Five beryllium minerals have been identified in the ore. These minerals are listed in Table B.5.

TABLE B.4 Influent Animals That Occur on the Western Seward Peninsula

Arctic Fox	Snowy Owl
Arctic Hare	Weasel
Lemmings	Wolverine
Ptarmigan	Ground Squirrel

TABLE B.5 Beryllium Minerals Present in the York Mountain Deposits

Chrysoberyl	$(Al_2 Be O_4)$
Euclase	$(Al Be Si -_4 (OH))$
Beryl	$(Al_2 Be_3 Si_6 O_{18})$
Bertrandite	$(2 Be_2 SiO_4 H_2O)$
Helvite	$((Fe, Mn, An)_4 Be_3 Si_3 O_{12} S)$

The U.S.G.S. conducted spectrographic analysis on ore samples taken from 14 beryllium bearing areas within the York Mountain region.⁽⁷⁾ Table B.6

TABLE B.6 Analysis of York Mountain Beryllium Ore

<u>Element</u>	<u>Percent</u>
Ca	.-01
Sn	.017
Pb	.014
Mn	.200
Fe	.370
Cr	.001
V	.002
Ti	.025
Mg	.620
Sr	.030
Ba	.006
B	.328
Be	.234
F	27.770

summarizes these results as an average. The spectrographic analysis also indicates the beryllium ores have an average F:Be ratio of 133.6.

At least five ore bodies of economic significance have been identified by the U.S.G.S. within the York Mountains. These ore bodies occur up to several hundred feet in length and vary in width from a fraction of an inch to as much as 50 feet. The most significant ore body to be identified by the Geologic Survey occurs along the Rapid River fault zone, where continuous mineralization can be traced for several thousand feet.⁽³⁾

In addition, it is significant to note potential beryllium deposits in unexplored regions of the Lost River. While in the area performing initial exploratory drilling, John Mulligan of The Bureau of Mines, Alaska Field Operations, reported the occurrence of beryllium minerals in outcrops along a nine mile unexplored area. Should these deposits prove to be similar to those in explored regions, reserve figures would be greatly increased.

Mining

Currently, beryllium is not being produced from deposits on the Seward Peninsula. Tin, however, was being produced from underground mines near Lost River until the 1940's. Techniques and problems associated with this mine can be applied when designing conceptual mining methods to extract beryllium. Further, the consulting firm of Watts, Griffis and McQuat, Limited, has prepared a comprehensive feasibility study for the production of tin-fluorite-tungsten at the old mine site.⁽⁸⁾ To the extent that beryllium is associated with these deposits, the report could apply equally well for the extraction of beryllium.

Mining methods proposed by the feasibility study are open-pit rather than underground. Based on rock mechanics in the area, pit slopes were designed so as not to undercut shear zones of weakness. Thus, the mine was designed to have pit slopes ranging from 52° to 65°. Also, bench height would be 40 feet with berm intervals 80 feet and berm width a minimum of 40 feet. In addition, the slope between berms would range from 75° to vertical.

In an attempt to better define the ore body and control feedstock grade, the conceptual design calls for removal of all overburden prior to any ore

extraction. A large part of this overburden will be used to construct roads and a tailings dam across the Lost River Valley. Upon extraction, the ore would be sorted and stockpiled by grade to insure a uniform feedstock to the nearby mill. This overburden combined with ore gangue material, would be relocated to construct roads and a tailings dam across the Lost River Valley. Upon extraction, the ore would be sorted and stockpiled by grade to insure a uniform feedstock to the nearby mill.

The former underground mine operations had experienced serious flooding problems because of the close proximity to Cassiterite Creek. The open-pit mining operation would attempt to alleviate this problem by diverting the creek into Lost River Valley to the West. Moreover, fissure zones would be sealed with grout to prevent seepage. This action would eliminate a significant portion of liquid effluents that would otherwise be released from the pit.

Weather conditions also create problems at Lost River. Wet fractured ore left to stand will freeze and require reblasting. Also, melting permafrost and frost-heaving often make roads impassable. In addition, ore shipments are only possible between June and October because of ice in the Bering Sea.

Potential Mining Effluents

A draft environmental impact statement on the Lost River project has been prepared by the Corps of Engineers.^(a) This document identifies solid waste as the most significant effluent that would be released by an open-pit mine. Effluents of lesser importance are fugitive dust and mine water.

Design of the conceptual mine at Lost River requires an initial overburden removal of 750,000 tons. Further, steady-state mining operations is expected to yield waste to ore ratios averaging 2.5.⁽⁸⁾ Solid waste material not being used in construction of roads and the tailings dam will be heaped for future reclamation programs.

Fugitive dust levels were not estimated for the proposed mine. However, it can be assumed dust levels would be similar to other open-pit mines. Previous studies indicate fugitive dust produced from fracturing and transporting

(a) "Draft Environmental Impact Statement, Lost River Project, Lost River, Alaska." 1974. Alaska District, Corps of Engineers, Anchorage, AK.

ore ranges from .1% to .15% of total ore mined.^{(a)(9)} Included in this total amount, each surface mining vehicle will produce dust levels between 1.4 to 2.8 kg per vehicle km. Of this dust, 58% will be from tires while 42% is from wind erosion.

Because Cassiterite Creek would be diverted from the ore body, a minimum of mine water is expected. Also, local runoff is only expected to be significant during the Spring thaw. It is estimated that pumps with a total capacity of 500 gallons per minute will be sufficient to handle peak mine water levels.⁽⁸⁾ Design calls for decanting this liquid effluent into shafts of the old underground tin mine. Should subsurface fissure systems provide a sufficient conduit to accommodate the water, little would be released at the surface.

(a) "Draft Environmental Statement for Homestake Mining Company's Pitch Project." 1978. USDA-FS-R2-DES(ADM) FY-78-03, U.S. Forest Service, Delta, CO.

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APPENDIX C

NIOBIUM ASSESSMENT

THE CURRENT NIOBIUM INDUSTRY

Niobium is considered a strategic metal because of its desirable properties as a metal alloy. By controlling grain size, the niobium alloy can improve the strength-to-weight ratio of steel. Demand for this type of steel is increasing due to its use in pipelines, offshore drilling platforms, bridges, and similar applications. Niobium is also used to produce corrosive resistant superalloys and components, as an electrical superconductor at cryogenic temperatures, and more recently, in nuclear reactors.

Currently, there is no domestic production of niobium. Rather, the U.S. imports all niobium in the form of pyrochlore ore concentrate, niobium-tantalite concentrate, tin slag, and ferroniobium. Brazil is the chief source of niobium and accounted for 31% of total U.S. imports in 1977. Other significant exporters of niobium are West Germany and Canada, which supplied the U.S. with 24% and 20% of total 1977 imports, respectively. Nigeria and Thailand combined contributed 20% to total U.S. imports of niobium in 1977.⁽¹⁾

NIOBIUM RESERVES AND RESOURCES

At present, there is no domestic production of niobium. In addition, the U.S. has no deposits of niobium that can be classified as reserves.

The most recent data published by the Bureau of Mines establishes world proved reserves to be 23 billion pounds of contained niobium. Of this amount, Brazil has 77%, the African countries 10%, and Canada 6%. The remaining 7% lies in Eastern Europe and the U.S.S.R. An additional 10 billion pounds of contained niobium exist in the world but are considered to be subeconomic at current prices and technology. The majority of these resources lie within the U.S.S.R., which has 55% of the total. Africa and Brazil each have 16% of total world niobium resources while Canada has 7%. In addition, the U.S. and Europe each have 3% of world niobium resources.⁽²⁾ World reserves and resources of niobium are summarized in Table C.1.

TABLE C.1. World Niobium Reserves and Resources⁽²⁾
(Million Pounds Contained Niobium)

<u>Country</u>	<u>Reserves</u>	<u>Resources</u>
United States	--	300
Canada	1,300	700
Brazil	18,000	1,600
U.S.S.R.	1,500	5,500
Europe	200	300
Zaire	900	--
Kenya	--	1,000
Nigeria	700	--
Uganda	700	--
Other African	200	600
World Total	23,500	10,000

Domestic Niobium Resources

The principal domestic resource of niobium occurs as pyrochlore and perovskite in a dolomitic intrusive plug in Gunnison County, Colorado. The Bureau of Mines estimates 140 million pounds of niobium metal are contained in the 40 million tons of ore rock that averages at least 0.25% Nb₂O₅.

The second largest domestic occurrence of niobium is in titanium, and aluminum ore in Arkansas. Titanium ores in Hot Springs County are associated with a perovskite-bearing magnetite-pyroxenite and contains up to 0.04% niobium in places. In addition, titanium-bearing bauxite in Pulaski and Saline Counties contain approximately 0.05% niobium. These two deposits, combined with alumina plant wastes, are estimated by the Bureau of Mines to contain 118 million pounds of niobium.

Domestic niobium resources of lesser significance occur in placer gravels in Valley County, Idaho, and Kiowa County, Oklahoma. These two deposits have been estimated by the Bureau of Mines to each contain about 20 million pounds of niobium metal.⁽²⁾

COMMERCIAL FORMS OF NIOBIUM

Niobium and niobium ore is marketed in a variety of forms. The majority of niobium imported to the U.S. is in the form of pyrochlore and niobium tantalite ore concentrates and ferroniobium as well as tin slags. In addition, niobium metal, synthetic concentrates, and special alloys are imported. Domestic import figures for 1977, published by the Bureau of Mines, are reproduced in Table C.2.

Pyrochlore $((\text{Na}, \text{Ca})_2 (\text{Nb}, \text{Ta})_2 \text{O}_6 (\text{O}, \text{OH}, \text{F}))$ and niobium-tantalite $((\text{Fe}, \text{Mn}) (\text{Nb}, \text{Ta})_2 \text{O}_6)$ ores and concentrates are sold on the basis of percent total pentoxides of niobium and tantalum. Niobium concentrate generally has 65% total pentoxide with a Nb_2O_5 to Ta_2O_5 ratio that ranges from as low as 8 1/2:1 to a high of 10:1. Pyrochlore concentrate generally contains 50 to 55% Nb_2O_5 and less than 1% Ta_2O_5 .

Ferroniobium is a ferroalloy produced as an intermediate product and contains from 60 to 70% niobium. Grade depends on the percentage of tantalum present which can range from 0.5% to 6.0%.

Niobium pentoxide is another intermediate product whose grade also depends upon the percentage of tantalum present. Purity can be as low as 89.0% or as high as 99.7% with the Ta_2O_5 present ranging from 0.25 to 10.0%.

Niobium metal is marketed in many physical shapes and forms with purity ranging from 95 to 99.9%.

TABLE C.2 U. S. Imports of Niobium in 1977
(000 Pounds Contained Metal)

<u>Form</u>	<u>Amount</u>
Ore concentrate, tin slag, metal, special alloys	2,200
Ferroniobium	2,600

APPENDIX C: REFERENCES

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